IAEA-TECDOC-1000

Clearance of materials resulting from the use of radionuclides in medicine, industry and research

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The originating Section of this publication in the IAEA was:

Waste Safety Section
International Atomic Energy Agency
Wagramerstrasse 5
P.O. Box 100
A-1400 Vienna, Austria

CLEARANCE OF MATERIALS RESULTING FROM
THE USE OF RADIONUCLIDES IN MEDICINE, INDUSTRY AND RESEARCH
IAEA, VIENNA, 1998
IAEA-TECDOC-1000
ISSN 1011–4289

Printed by the IAEA in Austria
February 1998

FOREWORD

It has long been recognized that the use of sources of radiation should be regulated. The recommended regulatory approach is based on a system of notification and authorization exemplified by the IAEA International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources (BSS). However, some types of radiation source do not need to be subject to regulatory control because they can only give rise to trivial radiation hazards. Such sources may be exempted or cleared from the requirements of regulatory control. Exemption applies to radiation sources which do not enter the regulatory control regime, while clearance is relevant to radiation sources which were within the regulatory regime but can be released because they present an insignificant risk to health. It is the latter situation which is the subject of this publication. Guidance is provided on applying the internationally agreed principles for the exemption and clearance of radiation sources and practices from regulatory control to derive clearance levels relevant to materials resulting from the use of radionuclides in medicine, industry and research.

While the emphasis of the report is on providing guidance on the clearance process and on methods for determining clearance levels, taking account of site specific factors as appropriate, a set of conservatively derived generic clearance levels is also presented. These are expected to be of considerable help to users of small quantities of radionuclides in laboratories and hospitals and their regulators. They can be used, taking due account of the associated notes of caution, as reference levels for determining when clearance is the appropriate option for disposing of the materials.

The report was developed through a series of consultants, Advisory Group and Technical Committee meetings. Although a draft document was prepared in April 1992, further development of the document was postponed until the publication of a related report entitled "Clearance Levels for Radionuclides in Solid Materials: Application of Exemption Principles". With the publication of the latter as IAEA-TECDOC-855 in January 1996, this report was further developed and revised to take into account, amongst other things, the publication of the BSS and the outcome of the Specialists Meeting on the Application of the Concept of Exclusion, Exemption and Clearance: Implication for the Management of Radioactive Materials (May 1997).

The IAEA wishes to acknowledge the work of the experts who took part in the development of the report, in particular, the contribution made by L. Baekelandt of Belgium and J. Simmonds of the United Kingdom. A list of contributors to drafting and review can be found at the end of this report. R. Rastogi and G. Linsley of the Division of Radiation and Waste Safety were the responsible officers at the IAEA.

EDITORIAL NOTE

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Many uses of radioactive materials are beneficial to mankind, for example in industry and research, 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, as described in the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) [1].

Some types of exposure to ionizing radiation are not subject to regulatory control because they are not amenable to such control (e.g. cosmic rays) and are therefore excluded from the regulatory process; some radiation sources present a very low risk to health and for these it is normally considered that control by regulatory process would be a waste of resources.

For sources which present such low risks, two situations can be distinguished:

(1) the sources do not enter the regulatory control regime, i.e. control is never imposed;
(2) the sources are released from regulatory control, i.e. control is no longer exercised.

The first situation is referred to as 'exemption' and the sources are said to be exempted from regulatory control. Exempted sources typically include sources of low activity 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.

The second situation, where sources of radiation are removed from the regulatory control regime, is referred to as 'clearance'. The corresponding levels of activity or activity concentration are called clearance levels. The cleared sources include wastes and materials that are a candidate for recycling and reuse. Cleared materials should not become subject of regulatory control again and, therefore, clearance levels will never exceed exemption levels.

It must be kept in mind that, when the predicted exposure from released materials is not certain to be trivial, their disposal must be authorized. Under such authorized disposal, appropriate regulatory requirements are maintained, for as long as necessary, to ensure safety.

The general principles and criteria for exemption and clearance have been detailed in the BSS. They are based on the principles for exemption that were agreed upon in 1988 [2] and on the 1990 Recommendations of the ICRP, which recognize "that the exemption of sources is an important component of the regulatory functions" [3] and iterate the basic criteria, namely that the source gives rise to small individual and collective doses in both normal and accidental conditions and that no reasonable control procedures can achieve significant reductions in individual and collective doses.

The clearance levels presented in this report are based on the same radiological principles and criteria [2] as those that were used to derive clearance levels for solid material [4] and for the recycling and reuse of materials [5].

1.2. OBJECTIVE

The objective of this report is to provide guidance on translating the principles of exemption from regulatory control into practice in the context of the waste arising from medical, research and industrial uses of radionuclides. It gives guidance on regulatory considerations in granting clearances and on the nature and scope of radiation dose calculations which must be performed in deriving clearance levels.

A further objective is to develop generic clearance levels for application to these types of waste. These are values, expressed in terms of release rates of radionuclides to the environment or activity concentrations in solid materials, below which there is no need for further regulatory control.

1.3. SCOPE

This report deals with the clearance from regulatory control of airborne, liquid, and solid wastes containing very low levels of unsealed radioactive materials originating from hospitals, research institutes and industrial installations that are not linked to the nuclear fuel cycle. Spent sealed sources and wastes arising from activities/practices where naturally occurring radionuclides are present (either in natural or enhanced concentrations) are not considered in this report.

1.4. STRUCTURE

Following the Introduction, the nature of the wastes from medical, research and industrial applications of radionuclides and the potential disposal routes are reviewed in Section 2. In Section 3 the relevant radiation protection principles and their applications are presented and regulatory issues related to the granting of clearances and their verification are discussed. In Section 4 the exposure scenarios which must be analyzed and the dose assessment modelling are reviewed and discussed. Section 5 deals with the derivation of generic clearance levels for airborne and aquatic releases and solid wastes.

The report is supplemented by appendices describing: averaging procedures for waste packages; the estimation of generic clearance values for airborne and liquid discharges and solid waste disposal; and the estimation of collective doses associated with clearance levels for airborne and liquid discharges.

2. NATURE OF THE RADIOACTIVE WASTE AND DISPOSAL PRACTICES

2.1. WASTE TYPES

The use of unsealed radioactive materials in hospitals for diagnosis and treatment as well as for research in the medical field results in the generation of various kinds of radioactive waste. These include items which have been contaminated with radioactive materials, such as paper, plastic gloves and covers, counting tubes, glassware, washing liquids and excreta from patients who have had radioisotopes administered for treatment or diagnosis.

A wide variety of radioisotopes are used in industrial applications and for research in the fields of agriculture, biology, physics, chemistry, etc. The wastes generated from such applications range from more easily handled materials such as contaminated aqueous liquids and combustible materials, to less easily handled contaminated materials such as organic liquids (e.g. scintillation fluids), large animal carcasses and large contaminated equipment.

Many of the radionuclides used in hospitals and research establishments have relatively short half-lives. Where appropriate, they should be disposed of following storage to allow for decay to harmless levels. Therefore, they are generally separated from longer lived radionuclides.

Much of the solid radioactive waste arising in hospitals, research establishments and industry consists of contaminated paper and plastics. These wastes may be disposed of by incineration or by direct disposal in landfills. At some establishments a proportion of solid waste consists of animal carcasses. These may be treated as combustible waste, or macerated and disposed with liquid waste, or collected for burial.

Liquid waste is generally segregated into aqueous and organic liquids. Organic scintillation liquids constitute a significant proportion of the organic liquids.

Airborne waste from hospitals and research establishments is generally less significant than solid or liquid waste. The potential for inhalation may, however, be significant and needs to be considered.

Table I gives an overview of some important radionuclides present in the waste arising from medical, research and industrial uses of radionuclides.

2.2. WASTE DISPOSAL

Routes commonly used for the disposal of waste from hospitals, industry and research institutes are as follows:

(a) airborne effluents:
discharge into the atmosphere either directly or through filtration systems;

(b) aqueous effluents:
discharge directly to sewer systems, or to septic tanks, or collection ponds, or to various water bodies such as rivers, lakes and the marine environment;

TABLE I. OVERVIEW OF SOME IMPORTANT RADIONUCLIDES USED IN MEDICINE, INDUSTRY AND RESEARCH

Radio-nuclide	Physical half-life	Principal application	Waste characteristics	Mode of disposal
H-3	12.3 a	clinical measurements, research, labelling, luminating devices	solid, liquid organic solvents HTO, HT	burial dispersal incineration
C-14	5.73×10^3 a	research, labelling, clinical measurements	solid, liquid solvent exhaled CO_2	burial dispersal incineration
Na-22	2.6 a	clinical measurements, research	liquid	dispersal
Na-24	15 0 h	clinical measurements	liquid	dispersal delay and decay
P-32	14 3 d	clinical therapy, research, labelling	solid, liquid	burial dispersal delay and decay
S-35	87.4 d	research	solid, liquid	burial dispersal delay and decay
Cl-36	3.01×10^5 a	research	solid, liquid	dilute and disperse as per regulatory authorization
K-42	12.4 h	biological research	mainly solid	delay and decay
Ca-45	163 d	research	mainly solid some liquid	burial dispersal delay and decay
Ca-47	4 53 d	clinical measurements	mainly solid some liquid	burial dispersal delay and decay
Cr-51	27 7 d	clinical measurements, research	mainly liquid	dispersal
Fe-59	44 5 d	clinical measurements, research	mainly liquid	dispersal delay and decay
Co-57	271 d	clinical measurements, research	solid, liquid	burial dispersal delay and decay
Co-58	70 8 d	clinical measurements, research	solid, liquid	burial dispersal delay and decay
Ga-67	3 26 h	clinical measurements,	liquid	dispersal delay and decay
Se-75	120 d	clinical measurements, research	mainly liquid	dispersal delay and decay

TABLE I (cont)

Radio-nuclide	Physical half-life	Principal application	Waste characteristics	Mode of disposal
Sr-89	50 5 d	clinical therapy	mainly liquid	burial dispersal delay and decay
Y-90	2 67 d	clinical therapy	mainly liquid	burial dispersal
Mo-99	2 75 d	extraction of Tc-99m	mainly liquid	delay and decay
Tc-99m	6 02 h	clinical measurements, research	solid, liquid	delay and decay
In-111	2 83 d	clinical measurements, research	solid, liquid	burial dispersal
I-123	13 2 h	clinical measurements	solid, liquid	delay and decay
I-125	60 1 d	clinical measurements, research, labelling	solid, liquid occasionally vapour	burial dispersal delay and decay adsorption on C
I-131	8 04 d	clinical measurements research clinical therapy	solid, liquid occasionally vapour	burial dispersal delay and decay adsorption on C
Xe-127	36 4 d	clinical measurements	gas, liquid	adsorption on C atmospheric dispersion
Xe-133	5 24 d	clinical measurements	gas, liquid	adsorption on C atmospheric dispersion
Pm-147	2 62 a	luminating devices	solid, liquid	burial, dispersal incineration
Er-169	9 4 d	biological research	mainly solid	delay and decay
Au-198	2 69 d	clinical therapy	liquid	delay and decay
Hg-197	2 67 d	clinical measurements	liquid, solid	delay and decay
Hg-203	46 6 d	biological research	solid, liquid	burial, dispersal delay and decay
Tl-201	3 04 d	clinical measurements	mainly liquid	burial delay and decay dispersal
Th-232	$1\,40 \times 10^{10}$ a	incandescent gas mantles	solid, liquid	burial dispersal

a years, d days, h hours

(c) organic liquids (including scintillation fluids):
incineration resulting in atmospheric releases, usually without off-gas cleaning; scintillation liquids are usually incinerated in their plastic vials, although sometimes, if the vials are made of glass, the liquids are decanted before being incinerated;

(d) solid waste:
solid waste such as paper, plastic and wood and glass vials, is usually disposed of with normal refuse in landfills; the combustible waste may be incinerated with normal refuse resulting in gaseous and particulate effluents and the ash which is disposed of in landfills.

The disposal of patient excreta containing radionuclides sometimes needs special consideration, for example when ^{131}I is used in the treatment of cancer and thyroid disorders. For the large activities typically applied during therapy (up to 10 GBq), patients, in general, should occupy specially equipped rooms which should have separated sanitary facilities for collecting the excreta in tanks until the radionuclides have decayed to an appropriate level, after which it may be released for dispersion in the sewer system. Alternatively, on the basis of radiological assessment, excreta may be discharged directly to the sewer if a specific authorization is given. The collection and decay storage of waste may involve manual operations which could result in the exposure of hospital staff to radiological and other hazards.

A significant proportion of the radioactive waste typically produced in hospitals and research institutes is of very low specific activity and may therefore be considered as candidate for clearance from regulatory control. All of the disposal routes listed above can be used for the disposal of the cleared waste. For short lived radioactive waste in liquid or in solid forms, storage to allow for decay below clearance levels is a common practice. Solid radioactive waste that is neither cleared for disposal with normal refuse nor incinerated is usually buried at an authorized radioactive waste repository or other special disposal facility.

3. EXEMPTION AND CLEARANCE PRINCIPLES AND THEIR APPLICATION

3.1. THE SYSTEM OF REGULATORY CONTROL

Sources and practices involving the exposure of people to ionizing radiation are normally controlled by a system of notification and authorization as exemplified in the BSS [1]. In general, prior to the use of radioactive materials, the operator is required to notify the Regulatory Authority of his intentions and to apply for an authorization in the form of a registration or a licence. Such authorizations normally include conditions related to the disposal of radioactive waste and recycle and reuse of radioactive material to ensure that doses to the members of the public are kept as low as reasonably achievable and below the appropriate dose limits and constraints. These authorizations usually stipulate disposal routes for various waste forms and limits in terms of activity concentration and/or total activity as conditions that the operator must comply with.

In the case of authorization for the controlled recycle of radioactive material or disposal of solid waste, conditions may be attached to the authorization in terms of quantities and/or further use or destination (e.g. controlled landfill, incinerator, recycling route, etc.). Control is carried out when the material is transferred from the originating facility, either at the originating facility or at the final destination.

In the case of discharge of liquid and gaseous effluents, the authorization may include discharge limits (i.e. total activity, concentration) and provisions to control the discharges (e.g. release rate). Control is carried out at the point of discharge and surveillance is performed in the environment. The implementation of the BSS in the context of discharges of radionuclides to the environment is described in Ref. [6].

3.2. RELEASE FROM REGULATORY CONTROL — THE CLEARANCE CONCEPT

Sources and practices may be removed from the system of regulatory control provided the radiological impact of these practices/sources after removal from the system is sufficiently low as not to warrant any further control. Such removal of sources and practices from regulatory control is called 'clearance'.

The basic criteria for determining, whether sources and practices within regulatory control system should no longer be subject to regulatory control are identical to the exemption criteria set out in the BSS; they are as follows:

(a) the radiation risks to individuals caused by the practice or source be sufficiently low as to be of no regulatory concern;

(b) the collective radiological impact of the practice or source be sufficiently low as not to warrant regulatory control under the prevailing circumstances; and

(c) the practices and sources be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in (a) and (b).

A practice or a source within a practice may be exempted (or cleared) without further consideration provided that the following criteria are met in all feasible situations:

(i) the effective dose expected to be incurred by any member of the public due to the practice or source is of the order of 10 μSv or less in a year, and

(ii) either the collective effective dose committed by one year of the performance of the practice is no more than about 1 man·Sv or an assessment for the optimization of protection shows that exemption is the optimum option.

It must be kept in mind that, when the predicted exposure from released materials is not certain to be trivial, their disposal, discharge, or use must be authorized. Under such authorized actions appropriate regulatory requirements are maintained, for as long as necessary, to ensure safety.

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[1] limits do not apply directly to potential exposures; ideally they should be replaced by a risk

[1]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

equivalent which takes 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 ICRP 60 [3] that a simpler approach can be taken provided that the doses are less than the dose limits even if the event occurs. In such a case it is adequate 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 in cases of scenarios, which have low probability of occurrence but 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.

A more pragmatic approach is to select only those scenarios which have a reasonable likelihood of occurring and to assign a probability of unity to them. With this approach dose limits can be applied as in the case of normal scenarios. This approach will usually be conservative but relies on good judgment in the selection of scenarios for analysis.

3.3. THE NEED FOR DERIVED QUANTITIES

The guiding radiological criteria for clearance are expressed in terms of dose and for practical application it is necessary to convert them into quantities derived in terms of mass activity concentration (Bq g^{-1}), surface contamination (Bq cm^{-2}), total activity per unit time (Bq a^{-1}) or total mass per unit time (t a^{-1}). 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 cleared 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 g^{-1}, Bq cm^{-2}, Bq a^{-1}, t a^{-1}, or some combination of them, which will satisfy the clearance criteria.

3.4. REGULATORY CONSIDERATIONS

It is important to define the terms of the clearance so that both regulatory authorities and applicants have a common understanding of the scope of the cleared practices. This is also needed in order to carry out an optimization analysis. A 'practice'[2] was defined in IAEA Safety Series No. 89 [2] in general terms that would be applicable to all types of exemptions from regulatory control. In the context of waste disposal from hospitals, industry and research institutes, it is not possible to further specify a definition of 'practice' that would be appropriate for all possible situations. However, the definition of a practice in the context of clearance can be illustrated by examples which represent common situations. In the case of airborne or liquid waste, the practice can often be defined as the discharge of low level radioactive effluents into the atmosphere or the aquatic environment from a given site. If more than one installation is discharging effluents into the same environment (from the same general geographic area), the combination of the discharges from these installations should also be taken into account. In general, as the geographical area under consideration increases, the collective dose will also increase due to the combined impact of several facilities in the same area. In a similar example for solid waste, a practice would be the disposal of cleared waste from a facility or facilities.

In the context of this report, the process of disposal begins when the operator relinquishes control of the waste. In most situations, this will be when the waste leaves the premises of the operator, unless, for some reason, the operator retains control of the waste

[2]Normally, the scope of this practice is smaller than the scope of the practices referred to in the context of the principle of justification.

after that time. For example, disposal of solid waste could begin when a dustbin or waste container passes into the control of other persons, or the operator's employee delivers it to a disposal facility. For liquid waste, disposal could be considered to begin when the waste is poured down the sink or drain on the operator's premises. For airborne waste, disposal could be considered to begin at the fumehood.

A radiological impact assessment needs to be performed before granting clearance. Such radiological impact assessment should justify and support clearance. The assessment should cover all the likely pathways of exposure situations that could arise from a practice being considered for clearance. In some circumstances the assessment may simply be a review to determine that the conditions of a reference evaluation, such as presented in this report, remain valid and that therefore the generic clearance levels can be applied. In general, the assessment should provide estimates of doses to workers and to members of the public who may be exposed after the point of disposal. Both normal and accidental exposure circumstances should be covered. The competent authorities should exercise judgment in considering exposure situations with a low probability of occurrence, such as the possible misuse of waste from the cleared practice which could have consequences serious enough to preclude clearance.

If the clearance levels are exceeded, it may be that clearance is still appropriate on the basis of a more detailed optimization analysis or, where justified, for reasons other than those concerned with radiation protection. On the other hand, even if both criteria are met, there may be other non-radiological reasons for not granting a clearance, for example if an appropriate quality assurance system is not in place.

3.5. GRANTING OF CLEARANCE

The clearance may be granted for specific activities or for all activities undertaken by a particular organisation. The clearance may also be granted for specific materials (e.g. concrete, steel, plastic). More generally, it may be possible to release material on condition that it is in a form which ensures, a priori, that the dose criteria for clearance will not be exceeded taking into account all feasible subsequent uses of the material. Such removal from regulatory control is called conditional clearance. It must be noted that, if subsequent verification or monitoring actions would be required to confirm compliance with the clearance criteria, for example verification of the final destination, the material may not be released through conditional clearance; it must be dealt with through authorized disposal.

The formulation of a clearance should not allow for circumvention of controls which would otherwise be applicable, by such means as dilution or fractionation of the waste. Although dilution in the environment is recognized as an important factor in reducing doses to members of the public, it is not appropriate to take relatively high specific activity materials and deliberately dilute them in order to meet the clearance criteria. One way to avoid problems of this type is to limit the total activity for all kinds of waste that are disposable under a clearance, rather than to rely solely on a concentration limit.

Competent authorities should periodically review the clearances to confirm that the radiological considerations and analysis continue to be valid, i.e. that the relevant parameters determining the conditions of a clearance have not changed significantly. This review should also determine whether the basic radiological criteria for clearance continue to be observed. A means should also be established to verify that the operator continues to comply with the

conditions of the clearance, normally by an appropriate national programme of inspection and the requirement to maintain records.

Competent authorities should ensure that clearances do not relieve the operator's responsibility for complying with other pertinent legal requirements covering the disposal of waste with other non-radiological but hazardous characteristics.

3.6. VERIFICATION OF CLEARANCE LEVELS

It is necessary to verify that the cleared material conforms to the applicable clearance levels. In general, all the activities required to verify compliance with the established levels for clearance should be performed within the framework of a suitable quality control system set up in accordance with pertinent recognized quality assurance requirements. Such a system should take into account the potential amount of cleared material, its final destination and the complexity of the practice. The system should cover, as appropriate, the keeping of an inventory of radioactive material, the keeping of records on quantities of cleared materials, related activity concentrations, arrangements for training and maintaining checks on disposal procedures.

Verification of clearance levels can be done by direct measurement on the material to be cleared, by laboratory measurements on representative samples, by use of properly derived relationships or by other means which are accepted by the competent national authority.

In industrial and other routine practices, in which there are economic and practical considerations, the choice of measurement strategy and appropriate measuring instruments are important factors. Depending on the radionuclides present, it may be necessary to supplement direct measurements on the material with laboratory analysis of suitably selected samples.

When deciding on a measurement strategy, the following steps should be considered:

- 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 pertinent 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.

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. Since the destination of the released material is unknown, it is possible that break-up of the released material would result in pieces which have a significantly higher activity concentration than the derived levels. The probability of this occurrence can be reduced by averaging over small quantities. Guidance on how such averaging can be carried out is provided in Appendix A.

4. RADIATION DOSE ASSESSMENT

The aim of this section is to indicate the nature and scope of radiological assessments which might be required to support or justify the clearance of waste from regulatory control.

Ideally, assessing the radiological impact of disposal would involve the use of environmental monitoring information as well as mathematical modelling. However, such monitoring information is not normally available because, inter alia, the quantities and concentrations of radionuclides in the environment are likely to be too low to be measured.

A wide range of methods and models exist for the assessment of the effect of releases of radionuclides to the environment. The models vary considerably in their complexity. Simple models are easy to apply and to explain to third parties but may not include all features which may be important. It is possible to make conservative assumptions rather than consider all processes in detail and to design the model such that the resultant radiation doses are more likely to be overestimated than underestimated.

For producing generic clearance levels for application in a wide range of circumstances, simple, robust assumptions will be required. A sophisticated model containing many detailed assumptions may not be helpful because the detailed assumptions may not be generally applicable. However, it should be recognized that a good understanding of relevant factors associated with the proposed disposal is required in order to demonstrate that the simple model is indeed robust (i.e., stands up to close examination).

It is important to begin by having a clear idea of the nature of the proposed disposal, to be sure exactly what is to be cleared and where regulatory control would cease. If the criteria for clearance are not met and the planned disposal has to be subject to regulatory control then more detailed models for radiological assessment may be required along with effluent and environmental monitoring [7]. Further justification that the resulting higher dose is as low as reasonably achievable must be made to the regulatory authority. Such models are outside the scope of this report.

4.1. RADIOLOGICAL CHARACTERISTICS OF THE WASTE

The first step in carrying out a radiological impact assessment is to characterize the radioactive waste involved in the proposed disposal. Information is required on the quantities, characteristics and types of radionuclides in the waste to be disposed of, the mode of disposal and the disposal locations.

Radiological and environmental characteristics vary over a large range, so it is necessary to determine clearance levels for each nuclide. Some basic radiological data (half life, dose coefficients for ingestion and inhalation, external irradiation dose coefficients for point and infinite plane source) are given in Table II for some relevant radionuclides. Dose coefficients for intake depend on the chemical form and may vary considerably. It is generally difficult to predict the chemical form of the nuclides after processing and especially when they are in the environment. It is recommended in such situations to adopt, for assessment purposes, the chemical form that gives the highest dose coefficient; this was done for the values presented in Table II. Two radionuclides of interest are tritium and C-14. Both are used extensively, often in relatively mobile forms and both are relatively long lived. In the particular case of tritium, it is of interest to distinguish between tritium gas and tritiated water because pathways, behaviour and radiological properties are very different in the two

TABLE II BASIC RADIOLOGICAL DATA FOR RADIONUCLIDES OF INTEREST

Radionuclides	Physical half-life	Dose per unit intake (a) Sv Bq^{-1}	
		Inhalation	Ingestion
H-3	12.3 a	2.6×10^{-10} S	1.8×10^{-11}
C-14	5.73×10^{3} a	5.8×10^{-9} S	5.8×10^{-10}
Na-22	2.60 a	1.3×10^{-9} F	3.2×10^{-9}
Na-24	15.0 h	2.7×10^{-10} F	4.3×10^{-10}
P-32	14.3 d	3.4×10^{-9} M	2.4×10^{-9}
S-35	87.4 d	1.9×10^{-9} S	1.3×10^{-10}
Cl-36	3.01×10^{5} a	7.3×10^{-9} M	9.3×10^{-10}
K-42	12.4 h	1.2×10^{-10} F	4.3×10^{-10}
Ca-45	163 d	3.7×10^{-9} S	7.1×10^{-10}
Ca-47	4.53 d	2.1×10^{-9} S	1.6×10^{-9}
Cr-51	27.7 d	3.7×10^{-11} S	3.8×10^{-11}
Fe-59	44.5 d	4.0×10^{-9} S	1.8×10^{-9}
Co-57	271 d	1.0×10^{-9} S	2.1×10^{-10}
Co-58	70.8 d	2.1×10^{-9} S	7.4×10^{-10}
Ga-67	3.26 d	2.4×10^{-10} M	1.9×10^{-10}
Se-75	120 d	1.3×10^{-9} S	2.6×10^{-9}
Sr-85	64.8 d	8.1×10^{-10} S	5.6×10^{-10}
Sr-89	50.5 d	7.9×10^{-9} S	2.6×10^{-9}
Y-90	2.67 d	1.5×10^{-9} S	2.7×10^{-9}
Mo-99	2.75 d	9.9×10^{-10} S	6.0×10^{-10}
Tc-99	2.13×10^{5} a	1.3×10^{-8} S	6.4×10^{-10}
Tc-99m	6.02 h	2.0×10^{-11} S	2.2×10^{-11}
In-111	2.83 d	2.3×10^{-10} M	2.9×10^{-10}
I-123	13.2 h	7.4×10^{-11} F	2.1×10^{-10}
I-125	60.1 d	5.1×10^{-9} F	1.5×10^{-8}
I-131	8.04 d	7.4×10^{-9} F	2.2×10^{-8}
Xe-127	5.24 d	(b)	(b)
Xe-133	36.4 d	(b)	(b)
Pm-147	2.62 a	5.0×10^{-9} M	2.6×10^{-10}
Er-169	9.30 d	1.0×10^{-9} M	3.7×10^{-10}
Au-198	2.69 d	8.6×10^{-10} S	1.0×10^{-9}
Hg-197 (inorganic)	2.67 d	3.0×10^{-10} M	2.3×10^{-10}
Hg-203 (organic)	46.6 d	5.6×10^{-10} F	1.9×10^{-9}
Tl-201	3.04 d	4.4×10^{-11} F	9.5×10^{-11}
Ra-226	1.60×10^{3} a	9.5×10^{-6} S	2.8×10^{-7}
Th-232	1.40×10^{10} a	1.1×10^{-4} F	2.3×10^{-7}

a years, d days, h hours

(a) Dose here means committed effective dose equivalent Committed effective dose per unit intake via inhalation/ingestion to members of the public (age > 17a) are for the chemical form giving the highest dose

(b) Uptake of noble gases into the body is very limited and so doses per unit intake are not evaluated The major hazard is from external radiation

Note: F, M and S denote fast, moderate and slow absorption from the lung, respectively

cases. The transformation of tritium gas to tritiated water is slow in the environment. This requires consideration of both forms for airborne release. The dose coefficients for inhalation and ingestion and for the external irradiation given in Table II have been taken from the IAEA Basic Safety Standards [1] and Refs [8, 9] respectively. The geometrical conditions for which these dose coefficients apply have to be taken into account.

4.2. ASSESSMENT OF RADIATION EXPOSURE

An outline of the major ways in which people can be exposed to radiation is given below for each of the waste types identified in Section 2. While it is not possible to include all possible exposure pathways and events which can lead to exposure, the associated uncertainty can be lessened with knowledge of the relevant local and regional conditions and of the environment relating to the disposal practice. The exposure pathways and scenarios of importance for the waste disposal options considered here have been well studied over many years. In some cases it has been possible to carry out validation studies in which predictions are compared with environmental measurements. This experience gives confidence in the results for the cases that are considered here.

Whatever the waste disposal option and type of waste, it is necessary to determine the ways in which radioactivity might transfer through the environment to people. Account needs to be taken of the possible accumulation of radionuclides in environmental media, as well as possible dilution and dispersion. Potential exposures, such as those due to accidents, also need to be taken into account with the probability of occurrence as well as the consequences being considered. The main concern is the assessment of doses to members of the critical group [6] but collective doses may also be required. Different models for assessing the radiation doses may be required depending on the different circumstances.

The IAEA has published information on methods and data to assess the radiological impact of planned releases of radionuclides to atmosphere and to water bodies [7]. This methodology is intended to be a relatively simple assessment tool that enables rapid evaluation of planned discharges to see whether a given practice is in compliance with relevant dose limiting criteria. It is considered sufficiently robust for use in an initial assessment of clearance levels for discharges.

4.2.1. Airborne releases

There is usually little or no delay between the production and the release of airborne radionuclides which leads to a continuous mixing in the atmosphere. This is in contrast to other waste types which are usually collected and then released after appropriate control. For this reason materials which generate airborne releases are usually controlled by limiting the quantities being handled to ensure that the amount released meets the appropriate criteria. Airborne releases to atmosphere are assumed to be through stacks or vents. The possible radiation exposures due to the disposal of filters may need to be considered. Exposures from the disposal of filters are of the same nature as those from any other solid waste (see Section 4.2.3).

Radionuclides released to atmosphere will be dispersed depending on the local meteorological conditions. They will give rise to exposure due to inhalation of, and external irradiation from, the passing plume. During dispersal radionuclides may be deposited on to the ground, depending on their physical form. The main exposure pathways of concern then are external irradiation from the deposited material; internal irradiation following the transfer

of radionuclides through the terrestrial foodchains; inhalation of radionuclides transferred back into the atmosphere by the action of the wind or mechanical disturbance (resuspension). The build up of deposits in the environment needs to be considered for longer lived nuclides.

If discharges are continuous or frequent, then average meteorological conditions are relevant for estimating doses. If discharges were to be infrequent, then it may be appropriate to consider the implications of a large fraction of the release occurring during the least favourable dispersion conditions. Consideration should also be given to the possibility of any discharge being strongly affected by a building, or being drawn into air intakes of a building, or being affected by any other circumstances that might limit dispersion.

The IAEA Safety Series No. 57 (currently under revision) [7] provides models for four different dispersion situations: release from high stacks; dispersion from short stacks under the influence of a building; dispersion where the source and receptor are on the same building surface; and dispersion where the receptor is very close to, but not on the building, and is under the influence of building wake effects. Models are also included for the transfer of radionuclides through terrestrial foodchains. Resuspension has been found to be a relatively minor exposure pathway following routine releases of radionuclides to atmosphere and so is not considered in Ref. [7]. It is important only where there are no further atmospheric releases and for those radionuclides, such as the actinides, where doses due to external radiation and foodchain transfer are relatively unimportant. If required, models for wind-driven resuspension and that due to man-made disturbances given in Ref. [10] may be referred to.

The models from Ref. [7] have been used to derive generic clearance levels for airborne discharges as outlined in Section 5.

4.2.2. Liquid releases

This section deals with releases of aqueous liquid waste. Non-aqueous liquid waste, such as organic waste and contaminated liquid scintillant are normally disposed of by incineration which would give rise to exposures as described in Sections 4.2.1 and 4.2.3.

A frequently used route for the disposal of very low level liquid waste is via the ordinary sewer system. In this case the exposure of sewage system workers needs to be evaluated. Also treatment of sewage may result in a contaminated sludge and a contaminated liquid, which is discharged to a water body. The sludge might be incinerated giving rise to contaminated airborne effluent and solid residue. Alternatively, the sludge might be treated and used as fertilizer, giving rise to exposures similar to those following deposition of radionuclides after atmospheric release. Yet another possibility is that the sludge is disposed of as solid waste.

The liquid effluent, or the sewage stream itself , may be discharged directly into a surface water body: a river, a lake, an estuary or the marine environment. Generally this results in considerable dilution which can be taken into account in assessing clearance levels. However, it is important to be sure that no circumstances could arise where this dilution does not occur.

Following release of liquid effluent to a river, the effluent is carried downstream and consideration has to be given to all the potential uses of the river water which might give rise to radiation exposure. The pathways of interest include: ingestion of drinking water; ingestion of fish taken from the river; and ingestion of foods derived from land irrigated with river

water. In addition, some radionuclides may become attached to sediments and fall to the river bed. These sediments might then act as a source of external irradiation or give rise to intake via inhalation. Sediments are also applied to agricultural land in some cases and so may give rise to the transfer of radionuclides through the terrestrial foodchain. The possible build-up of long-lived radionuclides in sediment and subsequent exposure pathways, therefore, also has to be considered. For very short lived radionuclides, the assessment should concentrate on the drinking water pathway.

Similar considerations also apply to releases to other water bodies. Another possibility for liquid waste disposal is discharge to a seepage pit. In this case radionuclides would be retained in the pit, or be carried away with the leachate. On closure of such a pit, radioactive material would be buried. Both the buried material and the leachate can be assessed in a similar fashion to solid waste disposal as discussed in Section 4.2.3.

Models for assessing the release of radionuclides to water bodies and to sewers, together with the subsequent exposure pathways are given in Ref. [7]. These models have been used to derive the generic clearance levels for liquid discharges as outlined in Section 5.

4.2.3. Solid waste

Combustible solids, as well as organic liquids and animal carcasses, may be incinerated giving rise to airborne discharges (see Section 4.2.1). Non-combustible solids, including ash from incinerators, might be disposed of to some form of municipal landfill. These facilities can vary considerably, both in their design and in the degree to which operations are controlled. At one extreme, such disposal might consist of little more than loose tipping on to a ground surface with final cover only being applied after some delay. At the other extreme, operations may be carefully controlled, with prompt covering of disposed material, preventing any human contact with the waste except by those workers directly concerned with the disposal operation. For such workers, exposure due to direct irradiation from the waste and as a result of the inhalation of resuspended radionuclides will be the main exposure pathways. In the first case, however, consideration should be given to the possibility of scavenging as an exposure scenario.

Once in the ground, radionuclides could find their way back to people by a number of mechanisms. Leaching by infiltrating groundwater is likely to be one of the most important mechanisms. Contaminated leachate may find its way to a surface water body (see Section 4.2.2), but there may be considerable delay in the ground before this occurs. Solubility of the radionuclides and the potential for sorption in the ground are important factors; both are element specific and vary from site to site according to the nature of the ground materials. The ground water flow system is also important. Again, this is site specific, and a full analysis must take account of factors such as rainfall and infiltration, level of the water table relative to the buried material, and the engineered features of the facility designed to minimize leaching and/or to control leachate movement.

For some radionuclides, generation of gas in the landfill may be a relevant process, e.g. tritium and ^{14}C may be incorporated into methane generated in the landfill. However, it may be of more significance in dryer climates than in humid climates.

Some radionuclides are not very mobile and will tend to stay in the landfill. Radiation exposure might occur only if the landfill is re-excavated after closure. In some cases former landfill sites are controlled for a certain period after closure, to stop any kind of post-disposal

intrusion, while in other cases little or no control is exercised and intrusion could occur soon after the site is closed. The probability that such intrusion will occur depends on many factors, some of which are site and country specific. This is only relevant for longer lived nuclides.

Solid waste in the form of sewage sludge might also be spread on agricultural land as a fertilizer. The possible transfer of radionuclides through foodchains is a route which should then be considered.

Some contaminated solids may also be suitable for recycling or reuse. While recycling of the materials considered in this report is not likely, reuse of instruments and equipment may occur. Clearance of these materials is addressed in Ref. [4] which also applies to large quantities of material. It must be noted, however, that the IAEA Regulations for the Safe Transport of Radioactive Material [11] define limits in terms of surface contamination. Objects with surface contamination above these levels should be transported in accordance with the requirements of Ref. [11].

5. DERIVATION OF CLEARANCE LEVELS

In this section a set of generic clearance levels is derived. For solid materials, the clearance levels are given in terms of activity concentration and are based on the BSS exemption levels [1]. However clearance levels for airborne and liquid releases need a separate derivation since the relevant discharge scenarios were not included when deriving the BSS values [12]. The methods used in this report to derive such levels have been taken from Ref. [7].

The generic values for releases to atmosphere, freshwater and as solid waste are presented in Tables III, IV and V respectively and are discussed in the following sections. The applicability and need for generic values is also discussed. The values in these Tables are to be considered as order of magnitude, reflecting a level of uncertainty associated with this type of generic assessment. In deriving clearance levels for atmospheric and liquid discharges, annual averages have been used for some parameters. Therefore, they are not appropriate for short term releases.

The values in Tables III, IV and V were derived with the intention of assuring that if complied with, annual doses to individual members of the public arising from any single cleared practice will not exceed 10 μSv. A conservative approach has, therefore, been adopted throughout in the assumptions and in the choice of scenarios and model data. However, as with all such assessments the results must be treated with caution. In particular, the possibility that the waste is not diluted as assumed after release or that exposure pathways other than those assumed are of significance should be considered. This is discussed further in Section 5.4.

5.1. GENERIC CLEARANCE LEVELS FOR RELEASE TO ATMOSPHERE

The set of generic clearance levels for releases to atmosphere presented in Table III was derived as outlined in Appendix B. It was necessary to make a number of assumptions in deriving these values and these assumptions were intended to give a cautious estimate of the clearance levels. In particular, it was assumed that the release occurs from a vent in the

TABLE III. DERIVED GENERIC CLEARANCE LEVELS FOR AIRBORNE RELEASES

Radionuclide	Annual release rate (Bq a^{-1})	Main exposure pathways and limiting age group
H-3	1×10^{11}	Ingestion
C-14	1×10^{10}	Ingestion
Na-22	1×10^{6}	External from deposit (Adults and Infants)
Na-24	1×10^{9}	External from deposit (Adults and Infants)
P-32	1×10^{8}	Ingestion (Infants)
S-35	1×10^{8}	Ingestion (Infants)
Cl-36	1×10^{7}	Ingestion (Infants)
K-42	1×10^{10}	External from deposit (Adults and Infants)
Ca-45	1×10^{8}	Ingestion (Infants)
Ca-47	1×10^{9}	External from deposit and Ingestion (Adults and Infants)
Cr-51	1×10^{9}	External from deposit (Infants)
Fe-59	1×10^{8}	External from deposit (Adults and Infants)
Co-57	1×10^{9}	Ingestion (Infants)
Co-58	1×10^{9}	Ingestion (Infants)
Ga-67	1×10^{10}	External from deposit (Adults and Infants)
Se-75	1×10^{8}	External from deposit (Adults and Infants)
Sr-85	1×10^{8}	External from deposit (Adults and Infants
Sr-89	1×10^{8}	Ingestion (infants)

Table III. (cont.)

Radionuclide	Annual release rate ($Bq\ a^{-1}$)	Main exposure pathways and limiting age group
Y-90	1×10^{10}	Inhalation and Ingestion (Infants)
Mo-99	1×10^{9}	External from deposit (Adults and Infants)
Tc-99	1×10^{7}	Ingestion (Infants)
Tc-99m	1×10^{11}	External from deposit (Adults and Infants)
In-111	1×10^{9}	External from deposit (Adults and Infants)
I-123	1×10^{10}	External from deposit (Adults and Infants)
I-125	1×10^{8}	Ingestion (Infants)
I-131	1×10^{8}	Ingestion (Infants)
Xe-127	1×10^{11}	External from cloud (Adults and Infants)
Xe-133	1×10^{12}	External from cloud (Adults and Infants)
Pm-147	1×10^{10}	Inhalation (Adults and Infants)
Er-169	1×10^{10}	Inhalation and Ingestion (Infants)
Au-198	1×10^{9}	External from deposit (Adults and Infants)
Hg-197	1×10^{10}	External from deposit (Adults and Infants)
Hg-203	1×10^{8}	External from deposit and Ingestion (Infants)
Tl-201	1×10^{10}	External from deposit (Adults and Infants)
Ra-226	1×10^{6}	Inhalation and Ingestion (Adults and Infants)
Th-232	1×10^{5}	Inhalation (Adults)

Notes:

(a) The calculations on which these values are based assume releases from a building vent or window. The closest individual is located 20 m from the release point and gets his food, 100 and 800 m from the release point. Doses are evaluated via inhalation, ingestion and external exposure routes.

(b) Significant differences in these values are possible for different source to receptor distances (see Appendix B).

TABLE IV. DERIVED GENERIC CLEARANCE LEVELS FOR LIQUID RELEASES

Radionuclide	Annual release rate (Bq a^{-1})	Main exposure pathways
H-3	1×10^{12}	River - Ingestion
C-14	1×10^{10}	River - Ingestion
Na-22	1×10^{5}	Sewage - External
Na-24	1×10^{8}	Sewage - External
P-32	1×10^{6}	River - Ingestion fish
S-35	1×10^{9}	River - Ingestion fish
Cl-36	1×10^{10}	River - Ingestion fish and water
K-42	1×10^{9}	Sewage - External
Ca-45	1×10^{10}	River - Ingestion fish and water
Ca-47	1×10^{8}	Sewage - External
Cr-51	1×10^{8}	Sewage - External
Fe-59	1×10^{6}	Sewage - External
Co-57	1×10^{9}	Sewage - External
Co-58	1×10^{8}	Sewage - External
Ga-67	1×10^{8}	Sewage - External
Se-75	1×10^{6}	Sewage - External
Sr-85	1×10^{6}	Sewage - External
Sr-89	1×10^{9}	River - Ingestion fish and water
Y-90	1×10^{10}	River - Ingestion fish and water
Mo-99	1×10^{8}	Sewage - External
Tc-99	1×10^{10}	River - Ingestion fish and water
Tc-99m	1×10^{9}	Sewage - External
In-111	1×10^{8}	Sewage - External
I-123	1×10^{9}	Sewage - External
I-125	1×10^{8}	Sewage - External
I-131	1×10^{7}	Sewage - External
Xe-127	Not applicable	
Xe-133	Not applicable	

TABLE IV. (cont.)

Radionuclide	Annual release rate ($Bq\ a^{-1}$)	Main exposure pathways
Pm-147	1×10^{10}	Sewage - External and River - Ingestion fish and water
Er-169	1×10^{10}	River - Ingestion fish and water
Au-198	1×10^{8}	Sewage - External
Hg-197	1×10^{9}	Sewage - External
Hg-203	1×10^{7}	Sewage - External
Tl-201	1×10^{8}	Sewage - External
Ra-226	1×10^{6}	Sewage - External
Th-232	1×10^{6}	Sewage - External

Note: The values are the most restrictive of those calculated following discharge to a river or discharge to a sewer (see Appendix B).

side of a building and that someone lives only 20 m from this point. This is intended to account for the relatively extreme conditions that could occur, for example, for hospitals in cities. However, considerably reduced doses would be obtained if the distance between the source and the receptor were greater, particularly if the release occurred from a stack. This is illustrated in Appendix B. In contrast, significantly higher doses would be obtained for the unlikely situation where the intake of air into a building was on the same building face as the vent. This possibility should be considered and, if applicable, revised clearance levels should be estimated.

The generic clearance values take account of the ingestion of crops produced 100 m from the release point and animal products produced 800 m from the release point; it was assumed that the food intake rates were typical of critical groups. The models and data used were developed for application in temperate European and North American conditions. The clearance values may therefore need to be reviewed for countries with significantly different types of diet, agriculture and style of living. Table III indicates which exposure pathways are most important in determining the clearance levels for releases to atmosphere.

5.2. GENERIC CLEARANCE LEVELS FOR AQUATIC RELEASES

The generic clearance values for liquid releases to sewers or freshwater bodies are given in Table IV and their derivation is outlined in Appendix B. For discharge to sewers two extreme possible scenarios were considered: assuming that no radioactive material is retained in sewage sludge but it is all discharged to the water body in liquid form; and assuming that all of the radioactive material discharged is retained in the sewage sludge at the sewage treatment works. Radiation doses were calculated for both cases and the most restrictive level used to give the generic values in Table IV. If the waste is discharged directly to a river and the sewage scenario is the most restrictive, then the doses will be overestimated. The release is assumed to occur into a relatively small river with a low flow

rate and a receptor location 1 km downstream of the release point. Changes in these assumptions, particularly changing the river size and flow rate can have a significant effect on the resultant doses. Releases to an estuary or to the marine environment will result in lower doses. However, releases to a lake may result in higher doses and need to be considered separately.

For radionuclides in sewage sludge, two exposure pathways were considered: external irradiation and inhalation of resuspended material. For releases to a river, the pathways considered were: ingestion of drinking water; ingestion of fish; and external irradiation from contaminated sediments. The transfer of radionuclides to the terrestrial foodchains due to irrigation or treatment with sewage sludge has not been considered in these calculations. In particular circumstances, these pathways may be important and should be considered. Table IV indicates which exposure pathways are most important in determining the clearance levels for aquatic releases.

TABLE V. GENERIC CLEARANCE LEVELS FOR SOLID WASTE (Bq/g)

Radio-nuclide	Clearance level for moderate quantities (a)	Radio-nuclide	Clearance level for moderate quantities (a)
H-3	1×10^{6}	Sr-89	1×10^{3}
C-14	1×10^{4}	Y-90	1×10^{3}
Na-22	1×10^{1}	Mo-99	1×10^{2}
Na-24	1×10^{1}	Tc-99	1×10^{4}
P-32	1×10^{3}	Tc-99m	1×10^{2}
S-35	1×10^{5}	In-111	1×10^{2}
Cl-36	1×10^{4}	I-123	1×10^{2}
K-42	1×10^{2}	I-125	1×10^{3}
Ca-45	1×10^{4}	I-131	1×10^{2}
Ca-47	1×10^{1}	Pm-147	1×10^{4}
Cr-51	1×10^{3}	Er-169	1×10^{4}
Fe-59	1×10^{1}	Au-198	1×10^{2}
Co-57	1×10^{2}	Hg-197	1×10^{2}
Co-58	1×10^{1}	Hg-203	1×10^{2}
Ga-67	1×10^{2}	Tl-201	1×10^{2}
Se-75	1×10^{2}	Ra-226	1×10^{1}
Sr-85	1×10^{2}	Th-232	1×10^{0}

(a) Moderate quantity means less than 3 tonnes per year and per facility. For larger quantities the clearance level is one tenth of the levels in Table V (see Section 5.3).

Note: The clearance levels for moderate quantities are identical to the BSS [1] exemption levels.

5.3. GENERIC CLEARANCE LEVELS FOR SOLID WASTE

Schedule I of the the Basic Safety Standards [1] contains exemption levels for a wide range of radionuclides. Sources whose activity or activity concentration is below the specified exemption level may be automatically exempted without further consideration from the requirements of the Standards. These exemption levels were derived using a conservative model for a series of limiting use and disposal scenarios [12]. The values represent the order of magnitude of the lowest values calculated in any scenario. They apply to practices involving small scale usage of activity, where only moderate quantities are involved (of the order of a tonne or less). In cases where quantities of the order of one tonne are involved, the exemption values of the BSS may be used as clearance levels. However, if larger quantities of material are involved, these levels might no longer be appropriate and would need some modifying factor to take account of the quantity involved.

In 1996, the IAEA published an interim report on clearance levels for radionuclides in solid materials [4]. Its aim was to provide a set of nuclide specific clearance levels which can apply to solid materials irrespective of the use to which they are put or of their destination after radiological control has been relinquished. These values are generally applicable to large amounts of material. The values were derived independently of the BSS exemption levels [1] but it has been observed that there is a fairly good agreement between these clearance levels and one tenth of the BSS exemption levels. In fact for the majority of radionuclides, they are within the same order of magnitude [13].

In the interests of promoting consistency and uniformity in recommended international values for exemption and clearance[3], the generic clearance levels for solid wastes recommended in this report are numerically equal to the exemption values in Schedule I of the BSS. However, they should be adjusted using the modifying factor of 1/10 when the amounts for clearance become large (Table V).

It is to be noted that the amount referred to in Table V is the quantity of material per year of practice, i.e. the amount of waste generated per year in a single facility. The term "of the order of 1 tonne" refers to the range between 0.3 and 3 tonnes, 1 tonne being the geometrical mean of the upper bound and the lower bound of that range. In the context of this report, the term "moderate quantity" is equivalent to "of the order of 1 tonne or less" and is therefore interpreted as "any quantity below 3 tonnes".

5.4. APPLICABILITY OF THE GENERIC CLEARANCE VALUES

Ideally, clearance levels should be derived using assumptions and model parameters that are appropriate to the particular practice and situation of interest. This would increase the degree of confidence associated with the use of such levels compared to those derived based on generic assumptions. Without appropriate site specific information the uncertainty associated with the dose estimate is likely to be higher and conservative assumptions have to be made to compensate for this uncertainty.

The generic clearance values presented here have been derived using conservative assumptions. They may be useful for an initial assessment of whether a proposed disposal could be cleared from regulatory control. However, they do not preclude the use of other

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[3]The BSS exemptional levels have also been adopted in the 1996 edition of the IAEA Regulations for the Safe Transport of Radioactive Material [11].

clearance values derived using more appropriate local or regional information. For a particular source, it may be appropriate to use site specific information and more realistic assumptions. However, if clearance levels are intended for general application to a large number of similar facilities within a particular country or region, then a site specific approach is not applicable and a generic, cautious approach of the type outlined here may be appropriate.

The scenarios, models and data used in the derivation of the generic values are all derived from experience in Europe and North America. The cautious nature of the assessment procedure will be expected to compensate for some differences between countries, for example in foodchain transfer and eating habits. However, there may be circumstances or exposure pathways that are outside the range of possibilities covered in this analysis. For example, it has been assumed that waste actually reaches the disposal facility. If it did not, then a range of possibilities exist for early exposure. However, this has been taken care of to some extent in the derivation of the clearance levels for solid waste. A judgement should be made by the competent authority as to whether the methodology outlined in this report is suited to local conditions. If it is felt that a potentially important scenario or exposure pathway has been omitted then a specific analysis should be undertaken.

5.5. COLLECTIVE DOSES

An important principle for exemption and clearance is that the radiation protection be optimized, which involves consideration of the collective dose associated with a practice. If it can be shown that the collective dose commitment is less than about 1 man·Sv per year of practice, then no further action is required but otherwise a more thorough optimization exercise may be needed.

The evaluation of collective dose requires information on the disposal practice as well as site specific data on the population at risk, exposure routes etc.. However, it is possible to explore, in a general way, the likely order of magnitudes of collective doses which could result from disposals of radioactive waste. The IAEA has published generic estimates of collective dose for unit discharge of a number of radionuclides to atmosphere and to water bodies [7]. These are order of magnitude estimates intended for screening purposes. They could be used to determine if the collective doses corresponding to the clearance levels for airborne or liquid discharges meet the 1 man·Sv criterion. Appendix C gives the collective doses per unit discharge for a range of radionuclides taken from Ref. [7] together with the estimated collective doses corresponding to the generic clearance levels. From Appendix C it is seen that in all cases the collective doses are below the 1 man·Sv value and so do not need to be considered further.

Similarly, for solid waste it has been discussed in Ref. [4] that collective doses are unlikely to exceed the 1 man·Sv criterion. Similar arguments should apply in the case of clearance levels based on the exemption levels of Ref. [1] for small material quantities. It is, therefore, likely that collective dose will not be a significant factor for the practices considered in this report and considering individual doses alone will usually be adequate for determining clearance levels.

5.6. CONSIDERATION OF MIXTURES OF RADIONUCLIDES

To determine if a mixture of radionuclides meets the criteria for clearance, a simple ratio expression can be used. This expression is:

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

where: C_i = the proposed release rate, Bq a^{-1} or concentration Bq g^{-1} of radionuclide i in the waste; C_{Li} = the limiting clearance release rate, Bq a^{-1} or concentration Bq g^{-1} of radionuclide in the waste; and n = the number of radionuclides in the mixture.

In the above expression, the ratio of the proposed release rate or concentration of each radionuclide to its clearance release rate or concentration is summed over all radionuclides in the mixture. If this sum is less than or equal to one, the proposed release rate or concentration for the mixture can be regarded as meeting the criteria. This type of relationship may be used by national authorities with their specific guidance on clearance release rates or concentrations to account for situations where multiple radionuclides are present in waste mixtures.

Appendix A

AVERAGING PROCEDURE

For solid waste, clearance levels have been derived in terms of activity concentrations. These values are meant to be averages over a moderate quantity of material. It is also assumed that the radionuclides are more or less uniformly distributed throughout the material.

It is prudent to establish a method by which the significance of the estimated average activity and the distribution of the radionuclides can be judged.

A simple method for assessing the average activity involves dividing the volume occupied by the material in a package (e.g. a 200 L drum, a 1 m^3 box) into defined portions and then assessing the activity concentration of each of these portions and comparing it with the appropriate clearance level.

The minimum number of portions is suggested to be ten and the maximum volume of each portion is suggested to be 20 L, whichever is more limiting in terms of portion volume. The activity of each volume should then be assessed, either through measurements, calculations or combinations thereof.

It is suggested that the activity concentration in each portion does not exceed ten times the clearance level, while the activity concentration, averaged over the package, does not exceed the appropriate clearance level.

Appendix B

ESTIMATION OF GENERIC CLEARANCE VALUES FOR AIRBORNE AND LIQUID DISCHARGES

This appendix outlines the calculations carried out to obtain the generic clearance values for discharge given in Tables III and IV of the main text. The basic assumptions are given and detailed calculations described for a single radionuclide as an example. In addition, the effect of changing some of the assumptions is demonstrated with some additional calculations. The models and data used in the calculations are all taken from Ref. [7].

B.1. AIRBORNE DISCHARGES

B.1.1. Calculation of dispersion and ground deposition

For airborne discharges it is assumed that the radionuclides of interest are released to atmosphere from a vent in the side of a building. The diameter of the vent pipe is 0.5 m and the cross-sectional area of the building is 500 m^2. An individual is assumed to live close to the vent in a separate building, 20 m high. The distance from the source to the receptor is about 20 m. The situation here is dispersion in the lee of a building inside a cavity zone where the height of release is less than 2.5 times the building height and the distance to the receptor point is less than 2.5 times the square root of the surface area of the appropriate wall of the building of concern. The following equation is used to calculate the concentration of radionuclides in air [14]:

$$C_A = (P_p Q_i) / (\Pi u_a H_B K) \quad \text{(B.1)}$$

where: C_A is the concentration of radionuclides in air in Bq m^{-3}
P_p is the fraction of the time per year that the wind blows towards the receptor of interest in sector p
Q_i is the annual average release rate for radionuclide i (Bq s^{-1})
u_a is the annual geometric mean wind speed at the height of the release (m s^{-1})
H_B is the building height (m)
K is a constant (value 1 m).

Equation B.1 represents a box model with a vertical dimension equal to the building height and a lateral dimension equal to 1 m. This model is an empirical formulation that yields cautious predictions of concentrations in air when compared with about 40 sets of field data from tracer experiments around nuclear reactor structures [7].

In the case considered, Equation B.1 applies for all source to receptor distances up to 56 m (2.5 × √500). From Ref. [7] the default values of u_a of 2 m/s and for P_p of 0.25 are believed appropriate and sufficiently conservative for screening calculations. For a release of 1 Bq/s over a year the predicted concentration in air is, therefore:

$$C_A = (0.25 \times 1) / (3.14 \times 2 \times 20 \times 1)$$
$$= 2 \times 10^{-3} \ Bq\ m^{-3}$$

From Ref. [7] the ground deposition rate is given by:

$$C_{gr} = (V_d + V_W) C_A \quad \text{(B.2)}$$

where: C_{gr} is the total daily average deposition rate on the ground of a given radionuclide from both dry and wet deposition processes (Bq m^{-2} d^{-1})
V_d is the dry deposition coefficient for a given radionuclide (m d^{-1})
V_W is the wet deposition coefficient for a given radionuclide (m d^{-1}).

In Ref. [7] it is suggested that a total deposition coefficient ($V_d + V_W$) of 1000 m d^{-1} be used for screening purposes for depositing aerosols and radioactive gases [15]. For tritium, carbon-14 and non-reactive gases such as krypton, it should be assumed that the total deposition coefficient is 0 [7]. In the case being considered for clearance, the ground deposition at the location of the individual is $1000 \times 2 \times 10^{-3}$, which is 2 Bq m^{-2} d^{-1}.

The air concentration and ground deposition given above is appropriate for the location of the individual but they could not obtain all of their food from such a location. It is, therefore, assumed that they obtain all vegetables and other crops 100 m from the source, while meat and milk are obtained 800 m from the source. These locations are such that dispersion in the lee of a building inside the wake zone is appropriate. Here, the concentration of the radionuclide in air at the location of interest is given by:

$$C_A = (P_p \times B \times Q_i) / u_a \qquad \text{(B.3)}$$

where: C_A , P_p , u_a and Q_i are as defined above and B is a parameter related to the downwind distance of interest, the surface area of the appropriate wall of the building of interest and the vertical diffusion parameter [7]. A Table of values of B for various downwind distances and cross sectional areas appropriate for neutral atmospheric stratification is also given in Ref. [7]. From this Table for a surface area of 401 to 800 m^2 and distances of 100 m and 800 m, the values of B are 1×10^{-3} and 6×10^{-5} m^{-2}, respectively. The air concentrations at 100 m is, therefore, estimated to be:

$$C_A = (0.25 \times 1 \times 10^{-3})/2 = 1.3 \times 10^{-4} \text{ Bq m}^{-3} \text{ for a release of 1 Bq s}^{-1}$$

and at 800 m

$$C_A = (0.25 \times 6 \times 10^{-5}) / 2 = 7.5 \times 10^{-6} \text{ Bq m}^{-3} \text{ for a release of 1 Bq s}^{-1}$$

The ground depositions at these points are calculated as above and is $1000 \times C_A$ i.e., 1.3×10^{-1} Bq m^{-2} d^{-1} at 100 m and 7.5×10^{-3} Bq m^{-2} d^{-1} at 800 m.

The relative locations of the discharge point and the receptor point can make a significant difference in the extent of the atmospheric dispersion and the resulting air concentrations and ground depositions. For example, if it is assumed that the receptor point is on the same face of the building as the discharge point and 10 m away (from Ref. [7]), the predicted air concentration would be 0.15 Bq m^{-3}. This is two orders of magnitude higher than the case considered here. In contrast, if the receptor point was 400 m from the discharge point then from Ref. [7] the predicted air concentration would be 2.5×10^{-5} Bq m^{-3}; two orders of magnitude lower than assumed in the reference clearance level calculations. Similarly, the distance assumed for food production affects significantly the relevant deposition rates.

B.1.2. Pathways of exposure

For radionuclides in the air the two exposure routes of concern are inhalation and external irradiation. The annual dose from inhalation E_{inh} (Sv a^{-1}) is given by:

$$E_{inh} = C_A \cdot R_{inh} \cdot DF_{inh} \quad \text{(B.4)}$$

where: C_A is the annual average concentration of the radionuclide in air at the point of interest (Bq m^{-3})
R_{inh} is the inhalation rate (m^3 a^{-1})
DF_{inh} is the dose coefficient for intake by inhalation (Sv Bq^{-1}).

The value of C_A for this case is 2×10^{-3} Bq m^{-3} from above. The value of R_{inh} is 1400 m^3 a^{-1} for infants and 8400 m^3 a^{-1} for adults [7]. The value of DF_{inh} is radionuclide dependent, for example for ^{131}I, DF_{inh} is 7.2×10^{-8} Sv Bq^{-1} for infants (1 to 2 years) and 7.4×10^{-9} Sv Bq^{-1} for adults [1].

Therefore for ^{131}I, the dose from inhalation is 2×10^{-7} Sv a^{-1} for infants and 1.2×10^{-7} Sv a^{-1} for adults.

The dose from external irradiation from radionuclides in the cloud, E_{im} (Sv a^{-1}) is given by:

$$E_{im} = C_A \cdot DF_{im} \cdot O_f \quad \text{(B.5)}$$

where: DF_{im} is the immersion dose conversion factor (Sv a^{-1} per Bq m^{-3})
O_f is the fraction of the time of exposure per year.

In this case O_f is considered to be 1.0, i.e., the individuals concerned spend all of their time at the location of interest. For ^{131}I the value of DF_{im} is 4.7×10^{-7} Sv a^{-1} per Bq m^{-3} for all ages [7].

Therefore, for ^{131}I, $E_{im} = 9.4 \times 10^{-10}$ Sv a^{-1}

For radionuclides deposited on the ground there are two additional exposure pathways of concern, external irradiation from the deposited material and ingestion of radionuclides in food.

The dose from external irradiation from deposited material, E_{gr} (Sv a^{-1}) is given by:

$$E_{gr} = C_{gr} \cdot DF_{gr} \cdot O_f \quad \text{(B.6)}$$

where: C_{gr} is the annual average deposition on the ground (Bq m^{-2} d^{-1})
DF_{gr} is the dose conversion factor for exposure to contaminated ground (Sv a^{-1} per Bq m^{-2} d^{-1}).

For the location of interest, O_f is again taken as 1.0 and C_{gr} is given above as 2 Bq m^{-2} d^{-1} for all ages. The value of DF_{gr} is radionuclide dependent; for ^{131}I the value is 1.2×10^{-7} Sv a^{-1} per Bq m^{-2} d^{-1} for all ages. It should be noted that no allowance is made for the shielding effects of buildings.

Therefore, $E_{gr} = 2.4 \times 10^{-7}$ Sv a^{-1}

The dose from ingestion of radionuclides in food E_{ing-p} (Sv a^{-1}) is given by:

$$E_{ing\text{-}p} = C_{p\text{-}i} \cdot H_p \cdot DF_{ing,i} \quad \text{(B.7)}$$

where: $C_{p\text{-}i}$ is the concentration of radionuclide i in food p at the time of intake (Bq kg^{-1})
H_p is the consumption rate for food p (kg a^{-1})
$DF_{ing,i}$ is the dose coefficient for ingestion of radionuclide i (Sv Bq^{-1})

and $$C_{p,i} = C_{gr} \cdot CU_{p,i} \quad \text{(B.8)}$$

where: C_{gr} is the annual average deposition on the ground (Bq m^{-2} d^{-1})
$CU_{p,i}$ is the concentration of radionuclide i in food p per unit deposit (Bq kg^{-1} per Bq m^{-2} d^{-1}).

From above it is assumed that crops are obtained at 100 m while milk and meat are obtained at 800 m from the discharge point where the ground dispositions are 1.3×10^{-1} and 7.5×10^{-3} Bq m^{-2} d^{-1} respectively. The foods consumed can be split into three groups: crops; milk; meat [7], where crops includes fruit, vegetables and grain. The intake rates for each food are given in Table B.I for infants (1 to 2 years) and adults. The food concentration per unit deposition rate is radionuclide dependent. From Ref. [7], the values for ^{131}I are:

6.6×10^{-1}	Bq kg^{-1} per Bq m^{-2} d^{-1} for crops
2.2	Bq kg^{-1} per Bq m^{-2} d^{-1} for milk
1.6	Bq kg^{-1} per Bq m^{-2} d^{-1} for meat.

The dose coefficients for ingestion are also radionuclide dependent and can be obtained from Ref. [1]. For ^{131}I, the values are 1.8×10^{-7} Sv Bq^{-1} for infants (1 to 2 years) and 2.2×10^{-8} Sv Bq^{-1} for adults.

Therefore, for ^{131}I the doses from ingestion of different foods are:

For infants:

For crops $E_{ing} = 6.6 \times 10^{-1} \times 1.3 \times 10^{-1} \times 150 \times 1.8 \times 10^{-7}$
$= 2.3 \times 10^{-6}$ Sv a^{-1}

For milk $E_{ing} = 2.2 \times 7.5 \times 10^{-3} \times 300 \times 1.8 \times 10^{-7}$
$= 8.9 \times 10^{-7}$ Sv a^{-1}

For meat $E_{ing} = 1.6 \times 7.5 \times 10^{-3} \times 40 \times 1.8 \times 10^{-7}$
$= 8.6 \times 10^{-8}$ Sv a^{-1}

Therefore, the total dose from ingestion for infants is 3.2×10^{-6} Sv a^{-1}.

For adults:

For crops $E_{ing} = 6.6 \times 10^{-1} \times 1.3 \times 10^{-1} \times 410 \times 2.2 \times 10^{-8}$
$= 7.7 \times 10^{-7}$ Sv a^{-1}

For milk $E_{ing} = 2.2 \times 7.5 \times 10^{-3} \times 250 \times 2.2 \times 10^{-8}$
$= 9.1 \times 10^{-8}$ Sv a^{-1}

For meat $E_{ing} = 1.6 \times 7.5 \times 10^{-3} \times 100 \times 2.2 \times 10^{-8}$
$= 2.6 \times 10^{-8}$ Sv a^{-1}

Therefore, the total dose from ingestion for adults is 8.9×10^{-7} Sv a^{-1}.

TABLE B.I. REPRESENTATIVE CRITICAL GROUP CONSUMPTION RATES OF DIFFERENT FOODS

Food	Consumption rate	
	Infants (1 to 2 a)	Adults
Crops (kg a^{-1})	150	410
Milk (l a^{-1})	300	250
Meat (kg a^{-1})	40	100
Water and beverages (m^3 a^{-1})	0.26	0.6
Freshwater fish (kg a^{-1})	15	30

The total individual dose, E_{TOT}, is obtained by summing the doses from each pathway. The results are:

For infants $E_{TOT} = 3.6 \times 10^{-6}$ Sv a^{-1} and
for adults $E_{TOT} = 1.2 \times 10^{-6}$ Sv a^{-1} for a discharge of 1 Bq s^{-1} for a year.

B.1.3. Estimation of generic clearance levels

From the previous section, an infant is estimated to receive a dose of 3.6×10^{-6} Sv a^{-1} for a release of 1 Bq s^{-1} for a year. The generic clearance level for discharge to the atmosphere is obtained as:

$$RC_a = \frac{Cc}{E_{TOT}} \times 3.15 \times 10^7 \qquad \text{(B.9)}$$

where: RC_a is the generic clearance level (Bq a^{-1})
Cc is the criteria for clearance (Sv a^{-1})
E_{TOT} is the total dose from unit discharge rate (Sv a^{-1} per Bq s^{-1})
3.15×10^7 is the conversion from Bq per second to Bq per year discharge.

In this case Cc is 10 μSv a^{-1} (i.e. 10^{-5} Sv a^{-1}) and for ^{131}I, E_{TOT} is 3.6×10^{-6} Sv a^{-1}. The generic clearance level for discharge to the atmosphere is therefore 8.75×10^7 Bq a^{-1}. This is rounded to 1×10^8 Bq a^{-1} in Table III.

Similar calculations can be carried out for other radionuclides and this has been done using data taken from Refs [7] and [1] supplemented by data from Ref. [16]. The radionuclide specific parameter values used in the calculation of the generic clearance levels are given in Tables B.II and B.III. The important exposure pathways for each radionuclide are indicated on Table III of the main text. Carbon-14 and tritium are treated separately (see Section B.3). In each case the generic value given in Table III is obtained by rounding to the nearest order of magnitude.

TABLE B.II. EXTERNAL AND INTERNAL DOSE COEFFICIENTS USED IN THE DERIVATION OF CLEARANCE LEVELS

Radionuclide	External dose from cloud (a), Sv per a/Bq m^{-3}	External dose from deposit (a), Sv per a/Bq m^{-2} d^{-1}	Dose coefficient for intake by inhalation (b) adults, Sv per Bq
H-3* (c)	1.04E-11	0.00E+00	4.50E-11
C-14* (c)	7.06E-12	5.08E-13	2.00E-11
Na-22	2.80E-06	7.00E-05	1.30E-09
Na-24	6.00E-06	7.80E-08	2.70E-10
P-32*	3.12E-09	9.18E-11	3.40E-09
S-35*	7.66E-12	5.30E-13	1.40E-09
Cl-36*	7.03E-10	2.12E-11	7.30E-09
K-42*	4.60E-07	8.39E-09	1.20E-10
Ca-45*	2.72E-11	1.45E-12	2.70E-09
Ca-47*	1.69E-06	3.15E-08	1.90E-09
Cr-51	4.00E-08	3.30E-08	3.70E-11
Fe-59	1.50E-06	1.70E-06	3.70E-09
Co-57*	1.77E-07	3.63E-09	1.00E-09
Co-58	1.20E-06	3.00E-08	2.10E-09
Ga-67	1.80E-07	1.90E-08	2.40E-10
Se-75	5.00E-07	1.80E-06	1.10E-09
Sr-85	6.30E-07	1.20E-06	8.10E-10
Sr-89	1.80E-10	2.40E-10	7.90E-09
Y-90*	5.99E-09	1.68E-10	1.50E-09
Mo-99	2.00E-07	2.90E-08	9.90E-10
Tc-99	6.70E-13	1.50E-10	1.30E-08
Tc-99m	1.60E-07	1.30E-09	2.00E-11
In-111	5.00E-07	4.50E-08	2.30E-10
I-123	2.00E-07	3.70E-09	7.40E-11
I-125	1.40E-08	5.20E-08	5.10E-09
I-131	4.70E-07	1.20E-07	7.40E-09
Xe-127*	3.94E-07	0.00E+00	0.00E+00
Xe-133*	4.92E-08	0.00E+00	0.00E+00
Pm-147*	1.71E-08	3.56E-10	7.30E-10
Er-169*	5.49E-11	2.55E-12	1.00E-09
Au-198	5.00E-07	4.10E-08	8.60E-10
Tl-201	1.10E-07	1.20E-08	4.40E-11
Hg-197	7.70E-08	7.70E-09	3.00E-10
Hg-203	2.80E-07	4.00E-07	2.40E-09
Ra-226	8.40E-09	3.60E-06	9.50E-06
Th-232	2.30E-10	3.60E-06	1.10E-04

TABLE B.II (cont.)

Radionuclide	Dose coefficient for intake by inhalation (b) infants, Sv per Bq	Dose coefficient for intake by ingestion (b) adults, Sv per Bq	Dose coefficient for intake by ingestion (b) infants, Sv per Bq
H-3 (c)	2.70E-10	1.80E-11	4.80E-11
C-14 (c)	6.60E-09	5.80E-10	1.60E-09
Na-22	7.30E-09	3.20E-09	1.50E-08
Na-24	1.80E-09	4.30E-10	2.30E-09
P-32	1.50E-08	2.40E-09	1.90E-08
S-35	4.50E-09	7.70E-10	5.40E-09
Cl-36	2.60E-08	9.30E-10	6.30E-09
K-42	1.00E-09	4.30E-10	3.00E-09
Ca-45	8.80E-09	7.10E-10	4.90E-09
Ca-47	7.70SE-09	1.60E-09	9.30E-09
Cr-51	2.10E-10	3.80E-11	2.30E-10
Fe-59	1.30E-09	1.80E-09	1.30E-08
Co-57	3.70E-09	2.10E-10	1.60E-09
Co-58	7.50E-09	7.40E-10	4.40E-09
Ga-67	1.00E-09	1.90E-10	1.20E-09
Se-75	4.50E-09	2.60E-09	1.30E-08
Sr-85	3.70E-09	5.60E-10	3.10E-09
Sr-89	3.00E-08	2.60E-09	1.80E-08
Y-90	8.80E-09	2.70E-09	2.00E-08
Mo-99	4.80E-09	6.00E-10	3.50E-09
Tc-99	3.70E-08	6.40E-10	4.80E-09
Tc-99m	1.00E-10	2.20E-11	1.30E-10
In-111	1.20E-09	2.90E-10	1.70E-09
I-123	7.90E-10	2.10E-10	1.90E-09
I-125	2.30E-08	1.50E-08	5.70E-08
I-131	7.20E-08	2.20E-08	1.80E-07
Xe-127	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00
Pm-147	3.60E-09	9.90E-10	7.40E-09
Er-169	3.50E-09	3.70E-10	2.80E-09
Au-198	4.40E-09	1.00E-09	7.20E-09
Tl-201	3.30E-10	9.50E-11	5.50E-10
Hg-197	1.20E-09	2.30E-10	1.60E-09
Hg-203	7.90E-09	1.90E-09	1.10E-08
Ra-226	2.90E-05	2.80E-07	9.60E-07
Th-232	2.20E-04	2.30E-07	4.50E-07

(a) All external doses were taken from Ref. [7] except radionuclides marked * where external doses from cloud and deposited activity were taken from Ref. [16].

(b) All dose coefficients for inhalation and ingestion were taken from Ref. [1] and are the most restrictive value given.

(c) For H-3 and C-14 a specific activity approach is adopted.

TABLE B III PARAMETERS USED TO CALCULATE TERRESTRIAL FOOD CONCENTRATIONS

Radionuclide	Concentration factor for root uptake into crops (a)	Concentration factor for root uptake into pasture (b)	Transfer to milk d per litre	Transfer to meat d per kg
H-3 (c)	0 00E+00	0	0 00E+00	0 00E+00
C-14 (c)	0 00E+00	0	0 00E+00	0 00E+00
Na-22	5 00E-02	0 6	2 50E-01	8 00E-01
Na-24	5 00E-02	0 6	2 50E-01	8 00E-01
P-32	1 00E+00	10	2 00E-02	5 00E-02
S-35	6 00E-01	6	2 00E-02	2 00E-01
Cl-36 (d)	5 00E+00	80	1 00E-03	1 00E-03
K-42 (d)	4 00E-02	1	1 00E-02	5 00E-02
Ca-45 (d)	3 00E-01	10	3 00E-03	1 00E-02
Ca-47 (d)	3 00E-01	10	3 00E-03	1 00E-02
Cr-51	1 00E-03	0 1	2 00E-04	9 00E-02
Fe-59	1 00E-03	0 1	3 00E-04	5 00E-02
Co-57	8 00E-02	2	1 00E-02	7 00E-02
Co-58	8 00E-02	2	1 00E-02	7 00E-02
Ga-67	3 00E-03	0 1	1 00E-05	3 00E-04
Se-75	1 00E-01	1	1 00E-03	1 00E-01
Sr-85	3 00E-01	10	3 00E-03	1 00E-02
Sr-89	3 00E-01	10	3 00E-03	1 00E-02
Y-90	3 00E-03	0 1	6 00E-05	1 00E-02
Mo-99	2 00E-01	1	5 00E-03	1 00E-02
Tc-99	5 00E+00	80	1 00E-03	1 00E-03
Tc-99m	5 00E+00	0	1 00E-03	1 00E-03
In-111	3 00E-03	0 1	2 00E-04	4 00E-03
I-123	2 00E-02	0 1	1 00E-02	5 00E-02
I-125	2 00E-02	0 1	1 00E-02	5 00E-02
I-131	2 00E-02	0 1	1 00E-02	5 00E-02
Pm-147	2 00E-03	0 1	6 00E-05	2 00E-03
Er-169 (d)	5 00E-02	0 1	3 00E-04	2 00E-04
Au-198	1 00E-01	0 4	1 00E-05	5 00E-03
Tl-201	2 00E+00	2	3 00E-03	1 00E-02
Hg-197	3 00E-01	3	5 00E-04	
Hg-203	3 00E-01	3	5 00E-04	1 00E-02
Ra-226	4 00E-02	0 4	4 00E-02	1 00E-02
Th-232	1 00E-03	0 1	1 00E-03	5 00E-03
				1 00E-04

(a) Bq/kg plant fresh wt per Bq/kg soil dry wt
(b) Bq/kg plant dry weight per Bq/kg soil dry wt
(c) Tritium and carbon-14 are modelled separately (see text)
(d) No values are given in Ref [7], therefore
Cl assumed to behave as Tc
Ca assumed to behave as Sr
K assumed to behave as Cs
Er assumed to behave as Ce

Note: All values are taken from Ref [7] unless indicated.

B.2. LIQUID DISCHARGES

B.2.1. Release to a sewer and retained in sewage sludge

As discussed in Section 5, two approaches are adopted for releases of radionuclides to a sewer: firstly, assuming that all of the material is retained in the sewage sludge at the sewage treatment works; secondly, assuming that all the activity is discharged to a water body. The doses due to the former scenario are considered here, while the latter case is dealt with in the following section.

If it is assumed that all of the radionuclides discharged are transferred to the sewage sludge, then the concentration in the sludge is given by Ref. [7]:

$$C_{sludge} = Q_i / S \qquad \text{(B.10)}$$

where: C_{sludge} is the annual average concentration of the radionuclide in the sludge (Bq kg^{-1})
Q_i is the annual discharge of the radionuclide i (Bq a^{-1})
S is the annual sewage sludge production at the relevant sewage treatment works (kg a^{-1}).

The annual sewage production at the relevant treatment plant will vary depending on the size of the treatment plant, which reflects the size of the population served by the plant. Reference [1] suggests a default annual sewage production of 400 tonnes a^{-1} (dry weight) based on the plant serving 20 000 people, each producing 20 kg a^{-1} (dry weight) of sewage.

For unit discharge per year, the concentration in sludge is therefore:

$$C_{sludge} = \frac{1}{4 \times 10^5} = 2.5 \times 10^{-6}\ Bq\ kg^{-1}\ per\ Bq\ a^{-1}\ released$$

There are two exposure pathways of concern here [7]: external irradiation from radionuclides in the sludge and inhalation of radionuclides resuspended into the air from the sludge. Both are for workers at the sewage treatment plant.

The external irradiation dose is estimated by Ref. [7].

$$E_{si} = C'_{sludge} \cdot DF_{gr} \cdot O_f \qquad \text{(B.11)}$$

where: E_{si} is the radiation dose in Sv a^{-1}
C'_{sludge} is the concentration of the radionuclide in sewage sludge in Bq m^{-2}
DF_{gr} is the dose conversion factor for ground contamination in Sv per Bq m^{-2}
O_f is the fraction of time of exposure per year (a^{-1}).

C'_{sludge} is the surface concentration of the radionuclide in unit area of the sewage sludge. It can be obtained from the concentration of radionuclides in unit mass of the sludge calculated from equation B.10, by assuming that the density of sewage sludge is 1.0×10^3 kg m^{-3} and that the depth of sewage sludge container is 1 m. Then $C'_{sludge} = 1000 \times C_{sludge}$ [7].

The fraction of time of exposure per year can be taken as 0.228 based on a working time of 2000 h a^{-1}. The value of DF_{gr} is radionuclide dependent and as used in Section B.1.2.

For ^{131}I , the value of DF_{gr} = 1.2 × 10^{-7} Sv a^{-1} per Bq m^{-2} d^{-1}. But for this calculation it is required in units of Sv per Bq m^{-2} and therefore this value is divided by 365 to give 3.3 × 10^{-10} Sv per Bq m^{-2}.

Therefore, the external dose from ^{131}I in sewage sludge is:

= 1000 × 2.5 × 10^{-6} × 3.3 × 10^{-10} × 0.228
= 1.9 × 10^{-13} Sv a^{-1} for 1 Bq a^{-1} discharge

The dose due to inhalation of resuspended material is given by Ref. [7]:

$$E_{res} = C_{sludge} \cdot R_{inh} \cdot DF_{inh} \cdot O_f \cdot DL \qquad (B.12)$$

where: E_{res} is annual dose from inhaling resuspended sewage sludge (Sv a^{-1})
C_{sludge} is the concentration of the radionuclide in sewage sludge in Bq kg^{-1}
R_{inh} is the annual inhalation rate in m^3 a^{-1}
DF_{inh} is the dose coefficient for intake by inhalation (Sv Bq^{-1})
O_f is the fraction of the year that exposure occurs
DL is the dust loading due to resuspension in kg m^{-3}.

The concentration in sewage sludge is in dry weight and as given above is 2.5 × 10^{-6} Bq kg^{-1} for a unit annual release. The annual inhalation rate is 8400 m^3 a^{-1} [7] and O_f can be taken as 0.228 as above. The ambient dust loading at the sewage works will depend on local conditions, in particular the extent to which the sewage sludge has dried out and the weather conditions. A default value of 0.1 mg m^{-3} is suggested in Ref. [7] based on measurements made around a sewage plant [17]; this corresponds to 1 × 10^{-7} kg m^{-3}. The dose coefficient is radionuclide dependent and for ^{131}I is 7.4 × 10^{-9} Sv Bq^{-1} for adults [1].

Therefore, the dose due to inhalation of ^{131}I in sewage sludge is:

= 2.5 × 10^{-6} × 8400 × 7.4 × 10^{-9} × 0.228 × 1.0 × 10^{-7}
= 3.5 × 10^{-18} Sv a^{-1} for 1 Bq a^{-1} discharge which is about 5 orders of magnitude less than the external dose.

The total dose to a sewage worker, therefore, from discharge of 1 Bq a^{-1} of ^{131}I to a sewer is obtained by summing these two doses and is 1.9 × 10^{-13} Sv a^{-1}.

B.2.2. Release to river

It is assumed that the radionuclide is released to a small river with a flow of 0.1 m^3 s^{-1} and that people live 500 m downstream from the outfall on the same side of the river.

From Ref. [7], the river flow corresponds to a river of width 3.47 m and depth 0.058 m.

The river flow velocity, U (m s^{-1}) can be obtained by:

$$U = \frac{q_w}{B \cdot D} \qquad (B.13)$$

Where: q_w is the flow rate in $m^3\ s^{-1}$
B is the width of the river in m
D is the depth of the river in m.

Therefore:

$$U = \frac{0.1}{3.47 \times 0.058} = 0.5\ m\ s^{-1}$$

For the distance to the receptor of interest, complete vertical mixing can be assumed [7] and the next stage is to calculate the partial mixing index A (dimensionless) of the river by:

$$A = \frac{1.5\ D\ x}{B^2} \qquad \text{(B.14)}$$

where: x is the distance from the release point to the receptor location (m)
D and B are as before.

For x = 500 m:

$$A = \frac{1.5 \times 0.058 \times 500}{3.47^2} = 3.6$$

The partial mixing coefficient, P_r is next obtained from Ref. [7] and for A = 3.6 is 1.6 (dimensionless). The radionuclide concentration at downstream distance x, C (Bq m^{-3}) is then from Ref. [7]:

$$C(x) = \frac{Q}{q_w} \exp \frac{(-\lambda_i\ x)}{U} \cdot Pr \qquad \text{(B.15)}$$

where: Q is the annual average release rate of the radionuclide (Bq s^{-1})
q_w is the annual average flow rate ($m^3\ s^{-1}$)
λ_i is the radionuclide decay constant (s^{-1})
x is the downstream distance (m)
U is the river flow velocity (m s^{-1})
Pr is the partial mixing coefficient (dimensionless).

For this case,

$$C(x) = \frac{1}{0.1} \times \exp\left(-\frac{9.98\times10^{-7} \times 500}{0.5}\right) \times 1.6 = 16\ Bq\ m^{-3}$$

This is the total concentration in the water and is the sum of the dissolved and suspended sediment sorbed radionuclides. The dissolved (filtered) radionuclide concentration $C_{W,S}$ (Bq m^{-3}) is given by Ref. [7]:

(B.16)

$$C_{w,s} = C \cdot \frac{1}{1 + k_d S_s}$$

where: k_d is the distribution coefficient (m^3 kg^{-1})
S_s is the suspended sediment concentration (kg m^{-3})
C is the total concentration (Bq m^{-3}).

Values of k_d can be obtained from Ref. [7] and depend on the radionuclide and the characteristics of the water body. For iodine a value of $k_d = 5 \times 10^{-3}$ m^3 kg^{-1} is given for freshwater in Ref. [7]. The suspended sediment concentration, S_s, will depend upon the characteristics of the particular river. A default value of 5×10^{-2} kg m^{-3} is used here.

Therefore:

$$C_{W,S} = 16 \times \frac{1}{1 + (5\times10^{-3} \times 5\times10^{-2})}$$

$$= 16 \text{ Bq m}^{-3}$$

When surface water is used for drinking, the suspended sediment is removed by water treatment processes. It can, therefore, be assumed that the dissolved radionuclide concentration is appropriate for estimating radiation doses from drinking water. However, removal efficiency varies depending on the radionuclide concerned and the specific water treatment used.

Radionuclides will also be deposited on shore and beach sediment. From Ref. [7] the surface contamination of shore/beach sediment is given by:

$$C_s = \frac{C \times k_d \times 60 \times 0.1}{1 + S_s k_d} \cdot \frac{1 - e^{-\lambda i\, Te}}{\lambda i\, Te} \qquad \text{(B.17)}$$

where: C_s is the surface concentration in shore/beach sediment (Bq m^{-2})
C is the concentration in water (Bq m^{-3})
k_d is the distribution coefficient (m^3 kg^{-1})
λi is the radioactive decay constant (d^{-1})
Te is the effective accumulation time in days.

The factor 60 (kg m^{-2}) takes into account the top layer (5.0 cm) and bulk density of sediments. The 0.1 is because the k_d for bottom and shore/beach sediment is assumed to be 1/10 of that for suspended sediment. Te allows for build up in sediments and is conservatively assumed to be 365 days.

Therefore, for ^{131}I the concentration on beach sediment is:

$$\frac{16 \times 5\times10^{-3} \times 60 \times 0.1}{1 + (5\times10^{-2} \times 5\times10^{-3})} \times \frac{(1-e^{-8.7\times10^{-2} \times 365})}{8.7\times10^{-2} \times 365} = 1.51\times10^{-2}\ Bq\ m^{-2}$$

The exposure pathways of concern here are:

- ingestion of radionuclides in drinking water;
- ingestion of freshwater fish; and
- external irradiation from radionuclides in shore/beach sediment.

The dose rate from ingestion of radionuclides in drinking water E_{DW} (Sv a^{-1}) is given by:

$$E_{DW} = C_{W,S} \times H_W \times DF_{ing,i} \qquad (B.18)$$

where: $C_{W,S}$ is the dissolved (filtered) radionuclide concentration (Bq m^{-3})
H_w is the consumption rate of drinking water (m^3 a^{-1})
$DF_{ing,i}$ is the dose coefficient for ingestion of radionuclide i (Sv Bq^{-1}).

$C_{W,S}$ is obtained from equation (B.17). From Ref. [7], the consumption rate of drinking water is 0.6 m^3 a^{-1} for an adult. For ^{131}I the dose conversion factor for ingestion is 2.2×10^{-8} Sv Bq^{-1} for adults.

Therefore, $E_{DW} = 16 \times 0.6 \times 2.2 \times 10^{-8}$
$E_{DW} = 2.1 \times 10^{-7}$ Sv a^{-1} for unit discharge to a river.

The dose from the ingestion of radionuclides in fish, $E_{ing,f}$ (Sv a^{-1}) is given by:

$$E_{ing,f} = C_{f,i} \times H_f \times DF_{ing,i} \qquad (B.19)$$

where: $C_{f,i}$ is the concentration of radionuclide i in fish (Bq kg^{-1})
H_f is the consumption rate of fish (kg a^{-1})

$DF_{ing,i}$ is the dose coefficient for ingestion of radionuclide i (Sv Bq^{-1})

and $$C_{f,i} = C_{W,S} \times Bp \qquad (B.20)$$

where: $C_{W,S}$ is the dissolved (filtered) radionuclide concentration (Bq m^{-3})
Bp is the equilibrium ratio of the concentration of radionuclide i in fish to its dissolved concentration in water (m^3 kg^{-1}), known as the bioaccumulation factor.

The dissolved radionuclide concentration is obtained from equation (B.17). Bp is radionuclide dependent and varies widely depending on the particular circumstances [7]. For ^{131}I a value of Bp of 40 Bq kg^{-1} per Bq l^{-1} is given in Ref. [7], this is equivalent to 4×10^{-2} Bq kg^{-1} per Bq m^{-3}.

Therefore, $C_{f,i} = 16 \times 4 \times 10^{-2}$
$= 6.4 \times 10^{-1}$ Bq kg^{-1}

The consumption rate of fresh water fish is taken as 30 kg a^{-1} [7] and $DF_{ing,i}$ is for ^{131}I as given above.

Therefore, $E_{ing,f} = 6.4 \times 10^{-1} \times 30 \times 2.2 \times 10^{-8}$
$= 4.2 \times 10^{-7}$ Sv a^{-1} for unit discharge to a river.

The external irradiation dose from radionuclides in sediment, $E_{m,i}$ (Sv a^{-1}) is given by:

$$E_{m,i} = C_S \times DF_{gr} \times O_f \quad (B.21)$$

where: C_S is the surface concentration in shore/beach sediment (Bq m^{-2})
DF_{gr} is the dose conversion factor for ground contamination (Sv a^{-1} per Bq m^{-2})
O_f is the fraction of time of exposure per year for the particular exposure pathway.

For ^{131}I from above, $C_S = 1.5 \times 10^{-2}$ Bq m^{-2}, and

$DF_{gr} = 3.3 \times 10^{-10}$ Sv per Bq m^{-2}.

From Ref. [7] a default value for O_f of 0.18 is given for adults.

Therefore, $E_{m,i} = 1.5 \times 10^{-2} \times 3.3 \times 10^{-10} \times 0.18$
$= 8.9 \times 10^{-13}$ Sv a^{-1}

The total dose due to unit discharge of ^{131}I to a river is obtained by summing the doses from all three exposure pathways and is 6.3×10^{-7} Sv a^{-1} per Bq s^{-1} released or 2.0×10^{-14} Sv a^{-1} per Bq a^{-1} released.

B.2.3. Estimation of generic clearance levels for aquatic releases

From the previous sections the dose to an adult for a release of 1 Bq a^{-1} of ^{131}I for a year is found to be 1.9×10^{-13} Sv a^{-1}, if it is assumed that all of the activity is retained in sewage sludge. If it is assumed that all of the activity is discharged to a river, the resulting dose is 2.0×10^{-14} Sv a^{-1}. The former case is therefore more restrictive for ^{131}I and so is used to derive the generic clearance levels.

The generic clearance level for liquid discharge is obtained as:

$$RC_e = \frac{Cc}{E_{TOT}} \quad (B.22)$$

where: RC_e is the generic clearance level (Bq a^{-1})
Cc is the criteria for clearance (Sv a^{-1})
E_{TOT} is the total dose from unit discharge rate (Sv a^{-1} per Bq a^{-1}).

In this case Cc is 10^{-5} Sv a^{-1} and for ^{131}I, E_{TOT} is 1.9×10^{-13} Sv a^{-1}. The generic clearance level for liquid discharge is therefore 5.3×10^{7} Bq a^{-1}. This is rounded to 1×10^{7} Bq a^{-1} in Table IV.

Similar calculations can be carried out for other radionuclides and this has been done using data taken from Refs [1], [4], [5] and [7]. The radionuclide specific parameter values

used in the calculations of the generic clearance values are given in Tables B.II and B.IV. The important exposure pathways for each radionuclide are indicated on Table IV of the main text. Tritium is treated separately (see Section B.3). In each case the generic value given in Table IV is obtained by rounding to the nearest order of magnitude.

B.3. CALCULATION OF GENERIC CLEARANCE VALUES FOR TRITIUM AND CARBON-14

Due to their behaviour in the environment the specific activity approach was adopted to calculate generic clearance values for tritium and carbon-14 as recommended in Ref. [7]. Using this approach it is assumed that a steady-state equilibrium has been attained between the environment and the exposed individual so that the ratio between the radionuclides and its stable counterpart is fixed.

B.3.1. Tritium

For tritium it is assumed that the nuclide is transferred through the environment and incorporated into the organism through its association with water molecules. The concentration of tritium in humans is then derived from its equilibrium concentration in the water vapour in the atmosphere resulting from airborne releases and in the water of aquatic environments receiving liquid releases. From Ref. [7] for airborne releases:

$$D_T = C_A \times DF_T \qquad \text{(B.23)}$$

where: D_T is the annual dose received by the person at the point of interest from ^3H
DF_T is the dose rate conversion factor (Sv a^{-1} per Bq litre $^{-1}$ of human water content)
C_A is the steady state concentration of ^3H in atmospheric water vapour (Bq l^{-1}) at the location of interest.

The location of interest is taken to be the point where the atmospheric concentration will contribute most significantly to the dose received. In this case it is assumed to be 100 m, the distance where crops are assumed to be grown, to take account of the important intake by ingestion. For tritium the dose rate factor is 2.6×10^{-8} Sv a^{-1} per Bq l^{-1}. The concentration of tritium at the point of interest is given by:

$$C_A = \frac{X}{H} \qquad \text{(B.24)}$$

where: X is the concentration of ^3H in air at the location of interest (Bq m^{-3}).
H is the absolute humidity of the atmosphere, assumed as a default to be 6×10^{-3} l m^{-3} of air.

TABLE B.IV. PARAMETERS USED FOR RELEASES TO RIVERS

Radionuclide	Distribution coefficient, Kd $l\ kg^{-1}$	Bioaccumulation factor, fish $Bq\ kg^{-1}/Bq\ l^{-1}$
C-14	10	1
Na-22 (a)	1000	20
Na-24 (a)	1000	20
P-32	30	50000
S-35 (b)	30	1000
Cl-36 (c)	10	50
K-42 (a)	1000	2000
Ca-45 (d)	100	75
Ca-47 (d)	100	75
Cr-51	100	200
Fe-59	1000	200
Co-57	1000	300
Co-58	1000	300
Ga-67 (b)	100	400
Se-75	30	200
Sr-85	100	75
Sr-89	100	75
Y-90 (b)	4000	30
Mo-99 (b)	30	10
Tc-99	10	50
Tc-99m	10	50
In-111	100	10000
I-123	5	40
I-125	5	40
I-131	5	40
Pm-147 (e)	1000	30
Er-169 (e)	1000	30
Au-198 (b)	200	35
Tl-201 (b)	100	1000
Hg-197 (d)	100	1000
Hg-203 (d)	100	1000
Ra-226	100	50
Th-232	1.00E+05	100

(a) Assumed to behave as caesium
(b) Kd value taken from Ref. [10] using analogous elements if necessary
(c) Assumed to behave as technetium
(d) Assumed to behave as strontium
(e) Assumed to behave as cerium.

Note: All values taken from Ref. [7] unless indicated.

From Section B.1.1, the value of X at 100 m from unit discharge to atmosphere is 1.3×10^{-4} Bq m^{-3}, therefore at 100 m:

$$C_A = \frac{1.3 \times 10^{-4}}{6.0 \times 10^{-3}}$$

$$= 2.2\ 10^{-2} \quad \text{Bq } l^{-1}$$

and

$$D_A = 2.2 \times 10^{-2} \times 2.6 \times 10^{-8}$$

$$= 5.7 \times 10^{-10} \quad \text{Sv } a^{-1} \text{ for a release of 1 Bq } s^{-1}$$

The clearance level is then:

$$= 10^{-5} \times 3.15 \times 10^{7} / 5.7 \times 10^{-10}$$

$$= 5.5 \times 10^{11} \text{ Bq } a^{-1}$$

For liquid disharges of tritium a similar approach is adopted where C_A is replaced by C_W the steady state concentration of tritium in water at the location of interest. From Section B.2.2 the value of C_W is 16 Bq m^{-3} per Bq s^{-1} release, which is $1.6\ 10^{-2}$ Bq l^{-1}. Therefore, the dose per unit discharge is:

$$D_W = 1.6 \times 10^{-2} \times 2.6 \times 10^{-8} \quad \text{Sv } a^{-1} \text{ per Bq } s^{-1}$$

$$= 4.2 \times 10^{-10}$$

and the clearance level is:

$$= 10^{-5} \times 3.15 \times 10^{7} / 4.2 \times 10^{-10}$$

$$= 7.5 \times 10^{11} \text{ Bq } a^{-1}$$

B.3.2. Carbon-14

The specific activity approach is adopted for atmospheric releases only and for liquid releases the same approach is used as for other radionuclides. It is assumed that the carbon-14 release is associated with CO_2 molecules and is subsequently fixed within plant tissues during photosynthesis. Also, the resulting organic molecules are assumed to be transported along with stable carbon through food chains and into the human body. Therefore, the dose at equilibrium will be directly proportional to the concentration of ^{14}C in air relative to the concentration of stable carbon at a given location. Therefore, from Ref. [7]:

$$D = A \cdot f_c \cdot DF_c \qquad \text{(B.25)}$$

where: D is the annual effective dose
A is the specific activity (Bq ^{14}C/ g C) to which food products at the location of interest will be chronically exposed.

DF_c is the effective dose rate factor that relates the annual dose rate (Sv a^{-1}) to the concentration of ^{14}C per g carbon in people (Bq g^{-1}).

The dose rate factor recommended in Ref. [1] is 5.6×10^{-5} Sv a^{-1} per Bq g^{-1}. Values of A are calculated as follows:

$$A = X / C_c \qquad (B.26)$$

where: X is concentration of ^{14}C in air at the location of interest
C_c is the concentration of airborne carbon, assumed to be 1.8×10^{-1} g m^{-3}.

As for tritium the location of interest is taken as 100 m and for this distance from Section B.1.1, $X = 1.3 \times 10^{-4}$ Bq m^{-3}.

Therefore:

$$D = 1.3 \times 10^{-4} \times 5.6 \times 10^{-5} / 1.8 \times 10^{-1}$$
$$= 4 \times 10^{-8} \text{ Sv } a^{-1} \text{ per Bq } s^{-1} \text{ released}$$

and the clearance level is

$$= 10^{-5} \times 3.15 \times 10^{7} / 4 \times 10^{-8}$$
$$= 7.9 \times 10^{9} \text{ Bq } a^{-1}$$

Appendix C

ESTIMATION OF COLLECTIVE DOSES CORRESPONDING TO CLEARANCE VALUES FOR AIRBORNE AND LIQUID DISCHARGES

The determination of collective dose requires information on the disposal practice together with site specific data on the population at risk, exposure routes, etc. However, generic estimates of collective dose have been made by the IAEA [7] for unit discharges of radionuclides to the atmosphere and to water bodies. These estimates are based on the results of models applied to locations in western Europe [18] and the results of a simple model developed by UNSCEAR [19]. These collective doses are order-of-magnitude estimates only and are for use for screening purposes. However, they are used here to estimate the collective dose associated with the clearance values for airborne and liquid discharges. The resulting collective doses can then be compared with the 1 man·Sv criterion.

TABLE C.I. COLLECTIVE DOSES FOR RELEASES OF RADIONUCLIDES TO ATMOSPHERE

Radionuclide	Clearance level (Bq a^{-1})	Collective dose	
		per unit discharge man·Sv per Bq a^{-1}	for clearance levels man·Sv
H-3	1×10^{11}	10^{-15}	10^{-4}
C-14	1×10^{10}	10^{-11}	10^{-1}
P-32	1×10^{8}	10^{-13}	10^{-5}
S-35	1×10^{8}	10^{-12}	10^{-4}
Cr-51	1×10^{9}	10^{-14}	10^{-5}
Fe-59	1×10^{8}	10^{-13}	10^{-5}
Co-58	1×10^{9}	10^{-13}	10^{-4}
Sr-89	1×10^{8}	10^{-14}	10^{-6}
I-131	1×10^{8}	10^{-12}	10^{-4}
Xe-133	1×10^{12}	10^{-17}	10^{-5}
Th-232	1×10^{5}	10^{-10}	10^{-5}

Note: The values provided here are order-of-magnitude estimates for screening purposes only and should be used with caution.

TABLE C.II. COLLECTIVE DOSES FOR RELEASES OF RADIONUCLIDES TO WATER BODIES

Radionuclide	Clearance level (Bq a^{-1})	Collective dose	
		per unit discharge man-Sv per Bq a^{-1}	for clearance levels man-Sv
H-3	1×10^{12}	10^{-14}	10^{-2}
C-14	1×10^{10}	10^{-11}	10^{-1}
P-32	1×10^{6}	10^{-12}	10^{-6}
S-35	1×10^{9}	10^{-14}	10^{-5}
Cr-51	1×10^{8}	10^{-16}	10^{-8}
Fe-59	1×10^{6}	10^{-13}	10^{-7}
Co-58	1×10^{8}	10^{-13}	10^{-5}
Sr-89	1×10^{9}	10^{-16}	10^{-7}
I-131	1×10^{7}	10^{-15}	10^{-8}
Th-232	1×10^{6}	10^{-10}	10^{-4}

Note: The values provided here are order-of-magnitude estimates for screening purposes only and should be used with caution.

Tables C.I and C.II give the collective doses per unit discharges taken from Ref. [1] and those corresponding to the clearance level for releases to atmosphere and to water bodies, respectively. Collective doses per unit discharge are only available for some of the radionuclides considered in this report. However, similar results are expected for other radionuclides. In all cases, the collective doses corresponding to the clearance levels are significantly below the 1 man·Sv criterion. Only for carbon-14 does the collective dose approaches the criterion due to its long half-life and global circulation. Therefore, for carbon-14 it may be necessary to consider collective doses if discharges are close to clearance level.

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