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Derivation of activity limits for the disposal of radioactive waste in near surface disposal facilities

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FOREWORD

Low and intermediate level waste is generated from a wide range of activities, such as the operation and decommissioning of nuclear fuel cycle facilities and the use of various sealed and unsealed radiation sources in a broad range of medical, industrial, research and other activities. The waste from these different activities often has different characteristics, which can influence its acceptability for near surface disposal facilities. Criteria for limiting the acceptance of waste in near surface facilities are needed to ensure protection of the workers and public during operation of the facilities and following their closure. Waste acceptance criteria can encompass both quantitative and qualitative elements, for example they can include radioactivity limits in terms of concentration and of total activity per package or per disposal site. They can also require limits on the amount of free liquids, an absence of particular toxic, inflammable or corrosive materials in the waste and a specification of the required strength of the waste packaging materials. The criteria are determined through an analysis of the importance for safety of the different elements within the waste disposal system.

The IAEA has been working on the development of a safety assessment methodology for near surface facilities and its applications for a number of years and, specifically, through the IAEA co-ordinated research project entitled **I**mprovement of **S**afety **A**ssessment **M**ethodologies for Near Surface Disposal Facilities (ISAM) (1997–2000). The ISAM project methodology provides a logical, transparent and systematic means for the evaluation of the operational and post-closure impacts of near surface radioactive waste disposal. This publication describes the application of the ISAM project methodology for the purpose of deriving radioactivity limits for low and intermediate level waste in near surface disposal facilities and provides illustrative values that can be used for reference purposes, for example at the preliminary planning stage of a disposal facility development.

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1. INTRODUCTION

1.1. BACKGROUND

Radioactive waste must be managed safely, consistent with internationally agreed safety standards [1–5]. The disposal method chosen for the waste should be commensurate with the hazard and longevity of the waste. Near surface disposal is an option used by many countries for the disposal of radioactive waste containing mainly short lived radionuclides and low concentrations of long lived radionuclides [3, 6]. The term “near surface disposal” encompasses a wide range of design options [7], including disposal in engineered structures at or just below ground level, disposal in simple earthen trenches a few metres deep, disposal in engineered concrete vaults, and disposal in rock caverns several tens of metres below the surface.

The use of a near surface disposal option requires design and operational measures to provide for the protection of human health and the environment, both during operation of the disposal facility[1] and following its closure.

Potential radiological impacts during operation of the facility may arise from routine and non-routine operations, which could give rise to radiation exposure to workers or members of the public [8]. Exposures during the normal operation of the facility could arise due to direct exposure to radiation from the waste/waste packages and from any radioactive releases from the packages or the disposal facility. Such exposures can be controlled using standard radiation protection measures, such as limiting the time of exposure, controlling the accessible distance to the source, providing shielding of the source and ensuring containment of the waste material. Exposures could result from abnormal operations at the facility (i.e. unplanned incidents) and could arise, for example, from physical damage to waste packages, leakage, fire or explosions. They may result in the unplanned release of radionuclides from the waste package into the surrounding environment. Such impacts can be limited by controlling the form and contents of the waste package and establishing suitable facility operating procedures. Requirements for safety are set out in the IAEA Safety Requirements for the Near Surface Disposal of Radioactive Waste [3]. The Safety Requirements specify radiation dose and risk limits and constraints for workers and members of the public and are applicable to the assessment of releases from both routine and non-routine operations.

Potential radiological impacts following closure of the disposal facility may arise due to gradual (continuous or episodic) processes, such as natural degradation of barriers or from disruptive events including human intrusion, which may affect the containment and isolation of the waste from the accessible environment. The potential for inadvertent human intrusion can be assumed to be negligible while active institutional controls are effective, but it will increase following the cessation of such controls. Requirements for safety in the post-closure period are set out in Reference [3]. They are expressed in terms of radiation dose or risk constraints and are intended to be applicable to the assessment of releases from both gradual processes and disruptive events.

To ensure the safety of both workers and the public (both in the short term and the long term), the operator is required to design a comprehensive waste management system for the safe operation and closure of a near surface disposal facility. Part of such a system is to establish

[1] “Disposal” refers to the emplacement of waste in a facility without the intention of retrieval [3]. The term “disposal facility” refers to the emplaced waste plus the engineered barriers and structures.

criteria for accepting waste for disposal at the facility. The purpose of the criteria is to limit the consequences of events which could lead to radiation exposures and in addition, to prevent or limit hazards, which could arise from non-radiological causes. Waste acceptance criteria include limits on radionuclide content concentration in waste materials, and radionuclide amounts in packages and in the repository as a whole. They also include limits on quantity of free liquids, requirements for exclusion of chelating agents and pyrophoric materials, and specifications of the characteristics of the waste containers. Largely as a result of problems encountered at some disposal facilities operated in the past, in 1985 the IAEA published guidance on generic acceptance criteria for disposal of radioactive wastes to near surface facilities [9]. These criteria are qualitative in nature and, for example, they do not address limitations on radionuclide content of waste, waste packages or the facility as a whole.

1.2. OBJECTIVE

The purpose of this publication is to present an approach for establishing radiological waste acceptance criteria using a safety assessment methodology and to illustrate its application in establishing limits on the total activity and the activity concentrations of radioactive waste to be disposed in near surface disposal facilities. The approach makes use of accepted methods and computational schemes currently used in assessing the safety of near surface disposal facilities both during the operational and post-closure periods.

1.3. SCOPE

The scope of this publication covers the use of safety assessment methodology to calculate total and specific activities limits for radioactive waste in near surface disposal facilities. It is used to evaluate the potential operational and post-closure radiological impact of solid and solidified radioactive waste in near surface facilities. The radioactive waste types used to illustrate the approach range from waste containing radionuclides used for medical, industrial and research purposes to waste arising from nuclear fuel cycle activities. They also include waste arising from the decommissioning of nuclear facilities is also included.

The following are not included in the scope of this publication:

— the operational radiological consequences of pre-disposal waste management (such as waste conditioning and storage);

— the disposal of radioactive waste in deep geological repositories, in boreholes and mine/mill tailings;

— the disposal of disused sealed sources; and

— the non-radiological hazards associated with the disposal of the radioactive waste.

Nonetheless, a similar approach to the one presented in this publicaton could be applied to this wider range of waste types and waste management practices, if appropriate modifications were made.

It is also important to recognize that the values derived are only illustrative of the approach for the disposal systems[2] considered in this publication. They should not be seen as IAEA

[2] Disposal system refers to the disposal facility and the geosphere/biosphere of the disposal site.

recommended limits to be applied to any disposal system. Nevertheless the derivation process and the results can be used as a benchmark against which to compare disposal system specific values. However, such comparisons should be undertaken with care. When undertaking such comparisons, checks must be made to assess the relevance and consistency of the assessment context, site and facility characteristics, scenarios, conceptual and mathematical models, and the associated data.

Finally, it should be emphasized that consideration is not given in this publication to all the broader elements that contribute to disposal system safety (safety strategy, iterative assessments, safety culture, robustness of design and defence-in-depth considerations, quality assurance, etc.) and to all the other limitations, controls and conditions important to safety (physical, chemical and biological properties, fire resistance, compatibility with handling equipment, identification and traceability requirements, etc.). The focus of the publication is on using of safety assessment methodology in derivation of quantitative radioactivity limits.

1.4. STRUCTURE

This report deals with the role of activity limits in disposal system safety (Section 2), the relevant radiation protection criteria (Section 3), the approach to derive activity limits (Section 4), illustrations of the application of this approach (Section 5), and guidance on the use of the approach (Section 6).

2. ACTIVITY LIMITS AND DISPOSAL SYSTEM SAFETY

In evaluating the safety of a disposal system it is important to identify the components of the system that provide or contribute to its safety (i.e. the disposal system concept), and the elements that are needed to assess the safety of the system (i.e. the assessment capability). This is illustrated in Table I.

The first column in Table I represents the physical and organizational aspects, which contribute to the safety of the disposal system, and includes system characteristics, such as sub-surface hydrogeology of the host site, characteristics of the engineered barrier materials, specification of construction materials, the quality of construction work, and operational practices such as waste package inspection and acceptance.

TABLE I. FRAMEWORK FOR CATEGORISING ELEMENTS OF DISPOSAL SYSTEM SAFETY.

Disposal System Concept	**Assessment Capability**
• Characteristics of site	• Quality of data and arguments (e.g., radionuclide inventory, permeabilities, radionuclide sorption coefficients). • Quality of assessment methods and models.
• Design of facility	
• Construction (e.g., specification of materials for structures and engineered barriers, and construction of facility).	
• Operational practices (e.g., inspection and acceptance of waste packages, emplacement of waste packages).	
↓	↓
Confidence in safety	

The second column in Table I presents additional aspects relevant to safety assessment related to the quality of safety assessment. The confidence that can be developed in the results obtained from the safety assessment depends directly on the quality of data, the methods adopted and the models used. For quantitative assessments, mathematical models and parameter values for these models are required. But also qualitative aspects (such as a systematic analysis of features, events and processes, traceability of information and decisions, and the use of multiple lines of reasoning) are essential components in building confidence in the safety case and its supporting safety assessment.

Control over the acceptance of radioactive waste in near surface facilities is therefore one of many elements that contribute directly to the safety of the disposal system. Waste acceptance requirements have to be specified in order to ensure that the waste packages can be safely disposed of and that they will not compromise the safe confinement of the waste by the various engineered and natural barriers. As noted in Section 1.3 both quantitative and qualitative waste acceptance requirements have to be established. Quantitative limits on radionuclide content in individual waste packages and in the facility are particularly important safety requirements in the case of disposal to near surface facilities, due to their proximity to the biosphere and to the possibility of human intrusion.

In the case of an actual near surface facility, appropriate safety assessment methods must be used to establish limits on radionuclide content of both individual packages and the disposal facility itself [3]. It is important to adopt an iterative approach to the design of a waste disposal system and the manner in which it will be operated. It is equally so in the assessment of its safety and in the establishment of limitations and controls on its construction and operation. Starting from a projected radionuclide inventory for the anticipated or actual waste streams, radionuclides that make an appreciable contribution to the total radiological impact can be identified and then more closely evaluated or measured as part of the evaluation process. For post-closure safety assessments, radionuclide activity limits will be determined at the initial stage of the facility development programme, following which the scenarios, models and parameters can be re-examined in greater detail to further improve the assessment as necessary. For example, sensitivity analysis can be performed to identify critical parameters enabling appropriate site specific data to be collected or measured and the assessment calculations repeated. There are also a number of management options that can be employed to enhance the long term safety of the system. For example, additional waste conditioning and engineered barriers, or longer institutional control periods can be considered.

An iterative approach is equally important in respect of operational safety, which can be managed in a more active and direct manner than post-closure safety, by modifications to the design or to operating procedures. For example, additional radiation shielding can reduce external exposure or the likelihood of accidental exposure reduced by improving waste handling procedures. Many operational safety issues can be addressed using engineering or administrative solutions rather than by limiting the activity content in a package or a facility. Good engineering and operating practice must be employed and the cost–benefit implications of using such engineering solutions would need to be addressed. It is also important to recognize that operational practices can affect not only operational safety, but also post-closure safety.

Thus the derivation of activity limits is part of the iterative process of developing a specific disposal system and is strongly influenced by many factors, such as operational practices, disposal facility design, and site and waste characteristics. Due to the hypothetical nature of the illustrative cases in Section 5 of this report, it was not considered to be practicable to demonstrate the iterative process.

3. RADIATION PROTECTION CRITERIA

3.1. GENERAL

All human activities involving actual or potential exposure to ionising radiation, including radioactive waste management, require implementation of measures for protection of humans and the environment. The IAEA Safety Fundamentals publication sets out nine principles that apply to all radioactive waste management activities, including disposal, which are necessary to meet the basic safety objective, i.e.:

The objective of radioactive waste management is to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations.

Disposal facilities must be developed (i.e. sited, designed, constructed, operated and closed) such that human health and the environment are protected both now and in the future. The prime concern being the potential radiological hazard that the waste will present in the long term.

3.2. RADIOLOGICAL PROTECTION DURING THE OPERATIONAL PERIOD

The objective and criteria for radiological protection during the operational period of a disposal facility are the same as for any nuclear facility, and are as required by the Basic Safety Standards [4] as summarized below.

Objective

The radiation doses to workers and members of the public exposed as a result of operations at the disposal facility shall be as low as reasonably achievable, social and economic factors being taken into account, and the exposures of individuals shall be kept within applicable dose limits and constraints.

Criteria

Radiation dose limits and constraints for workers and of members of the public are set out in Schedule II of the Basic Safety Standards [4]:

(a) the occupational exposure of any worker shall be controlled so that the following limits are not exceeded:

– an effective dose of 20 mSv per year averaged over five consecutive years,

– an effective dose of 50 mSv in any single year.

(b) the estimated average doses to the relevant critical groups of members of the public from all practices shall not exceed the following limit:

– an effective dose of 1 mSv in a year.

Members of the public could receive exposure from a number of sources. To comply with the above limit, a facility such as a radioactive waste disposal facility (which constitutes a single source) shall be designed so that the estimated average dose to the relevant critical groups of members of the public, who may be exposed as a result of the facility and its operation, satisfies a dose constraint of not more than 0.3 mSv per year. This corresponds to a risk of the order of 10^{-5} per year.In radiological protection terms, the source (e.g. disposal facility) is under control, releases can be verified and exposures to workers and the public controlled. The engineering and practical means of achieving protection are well known, and provision must be made for their application in disposal facilities.

Optimization of protection must be considered at every stage of the development of the disposal facility (e.g. design, planning of operations) in order to ensure that radiation doses to workers will be as low as reasonably achievable. Important and relevant considerations from a safety point of view include the separation of construction activities from waste emplacement activities, the use of remote handling and shielded equipment for waste emplacement, the control of working environments, reducing the potential for accidents and their consequences, and the minimization of maintenance requirements in radiation and contamination areas.

During normal operation of a disposal facility only very minor releases of radioactive material, if any (e.g. releases of gaseous radionuclides), are expected so that significant doses to the workers and members of the public are not anticipated. Even in the event of operational accidents involving a breach of packaging, releases are likely to be mainly contained within the facility and doses to workers are not likely to be significant. Relevant considerations in this regard include the waste packaging, the form and content of the waste, control of the content and contamination on waste packages and equipment and monitoring of the disposal facility ventilation exhaust air and drainage water.

Doses and risks associated with the transport of radioactive waste to the disposal facility should be managed in the same way as those associated with the transport of other radioactive materials. The safety of transporting waste to the disposal facility is achieved by complying with the IAEA Regulations for the Safe Transport of Radioactive Material [10].

3.3. RADIOLOGICAL PROTECTION IN THE LONG TERM

The primary design goal of a radioactive waste disposal facility is to provide for the protection of human health and the environment in the long term, after the facility is closed and until the time when the associated radiological hazard will reach an insignificant level. In this period, the migration of radionuclides to the biosphere and consequent exposure of humans may occur due to slow degradation of barriers, slow natural processes and also following discreet events that may alter the disposal system barriers or lead to short term release of radionuclides. Radiation protection criteria relevant to the post-closure phase of near surface disposal facilities are set out in the relevant IAEA Requirements [3] and in a recent ICRP publication [11].

3.3.1. Objective

Radioactive waste disposal facilities shall be sited, designed, constructed, operated and closed so that protection in the long term is optimized, social and economic factors being taken into account, and a reasonable assurance provided that doses or risks in the long term will not exceed the dose or risk constraints for members of the public.

3.3.2. Criteria

The dose limit for members of the public from all sources is an effective dose of 1 mSv in a year, and this or its risk equivalent should be considered as criteria not to be exceeded in the future. To comply with this limit, a waste disposal facility (which constitutes a single source) shall be designed so that the estimated average dose or risk to members of the public, who may be exposed as a result of the disposal facilities in the future, shall not exceed a dose constraint of 0.3 mSv in a year or a risk constraint of the order of 10^{-5} per year.

These criteria are applicable to exposures resulting from gradual processes, which occur as a result of the expected evolution of the disposal system. Situations in which exposure could arise as a result of the occurrence of unlikely events that affect the repository, i.e. events, which low associated probabilities, should also be considered. The regulatory body should decide whether outcome of unlikely events should be compared with risk constraint or whether the probability of occurrence and the resulting dose should be considered separately [3].

The ICRP has given guidance on radiological criteria applied to human intrusion [11]. It suggests that the dose constraint of 0.3 mSv per year is not applicable in evaluating the significance of human intrusion because human intrusion bypasses the barriers that were established to provide the radiation protection of the facility. Nevertheless, it considers that a measure of the significance of human intrusion is necessary at the design stage of the facility development. In circumstances where human intrusion could lead to doses to those living around the site sufficiently high that intervention on current criteria would almost always be justified, reasonable efforts should be made to reduce the probability of human intrusion or to limit its consequences. In this respect the ICRP has advised that an existing annual dose of around 10 mSv may be used as a generic reference level below which intervention is not likely to be justifiable and that an existing annual dose of around 100 mSv may be used as a generic reference level above which intervention should be considered always justifiable.

Constrained optimization is recommended as the central approach to ensure the radiological safety of a waste disposal facility. In this context, optimization of protection is a judgmental process with social and economic factors being taken into account and should be conducted in a structured, but essentially qualitative way, supported by quantitative analysis as appropriate.

In general, protection can be considered optimized if:

— due attention has been paid to the long term safety implications of various design options at each step during development of the disposal system;

— the assessed doses and/or risks resulting from the generally expected range of natural evolution of the disposal system do not exceed the appropriate constraint;

— the probability of unlikely events that might disturb the performance such as to give rise to higher doses or risks has been reasonably reduced by siting, design, or institutional control; and

— the design, construction and operational programmes have been subjected to a quality management programme.

3.4. USE OF RADIATION PROTECTION CRITERIA

The IAEA Requirements [3] requires an assessment of possible radiological doses to the public to be carried out by reference to the exposed group of individuals in the population receiving the highest dose (the critical group). For any given exposure mechanism leading to a dose into the future, the critical group will be somewhat hypothetical because human habits may change significantly, even over a short period of time. Exposure scenarios for the critical group should be postulated on the basis of an appropriately conservative analysis of events and processes that will not lead to doses to the exposed individuals being underestimated.

For normal exposures, doses for individuals are expressed in effective dose in a year for external radiation and in effective dose committed in a year for intake of radionuclides. The addition of external annual effective dose plus effective dose committed annually is the relevant quantity to be compared with the established dose constraint.

Potential exposures must be taken into account in evaluating the overall safety of a disposal facility. Such exposures do not occur with certainty, but have only a potential to occur. The control of risk from potential exposures can be achieved by two ways: (a) by increasing protection in a manner that decreases the probabilities of occurrence of the events; or (b) by increasing protection in a manner that decrease the consequences, e.g. doses from the events if they occur (this is called mitigation). For coherence of all the protection efforts, it is usually recommended [12] that protection against potential exposures should have the same level of ambition than protection against normal exposures (same risk of health effects).

As previously indicated the constrained optimization process is clearly facility, site and programme specific, it is impracticable to introduce it in the illustrative cases given in Section 5 of this publication, especially since the pertinent economic and social factors can not be defined for the hypothetical cases.

4. APPROACH TO DERIVING ACTIVITY LIMITS

4.1. GENERAL

In previous studies that have been undertaken to derive such limits [13, 14], the safety assessment approach has been used. The Safety Guide on Safety Assessment for Near Surface Disposal notes [8] that the "results of safety assessments are an important means for determining inventory and/or concentration limits for specific radionuclides in the waste and provide one way for developing waste acceptance requirements for the near surface repository". Such an approach can be used for both operational and post-closure periods of a disposal facility. The main criterion underlying the derivation of activity limits for near surface disposal facilities is that the consequential radiation doses to workers and to members of the public from the chosen exposure scenarios are compatible with the system of radiological protection criteria discussed in Section 3.

Safety assessment methods have been developed and applied in a variety of ways for the assessment of near surface facilities [7]. The IAEA Co-ordinated Research Project for the Improvement of Long Term Safety Assessment Methodologies for Near Surface Radioactive Waste Disposal Facilities (ISAM) provided a critical evaluation of the safety assessment approaches. As part of the preparatory work for ISAM, the key components of a safety assessment approach were identified and synthesized (Fig. 1). A version of Fig. 1, revised for use in this study, is shown in Fig. 2. Key components are:

— the specification of the assessment context (Step 1);
— the description of the disposal system (Step 2);
— the development and justification of scenarios (Step 3);
— the formulation and implementation of models (Step 4); and
— the calculation and derivation of illustrative activity limits (Steps 5 and 6).

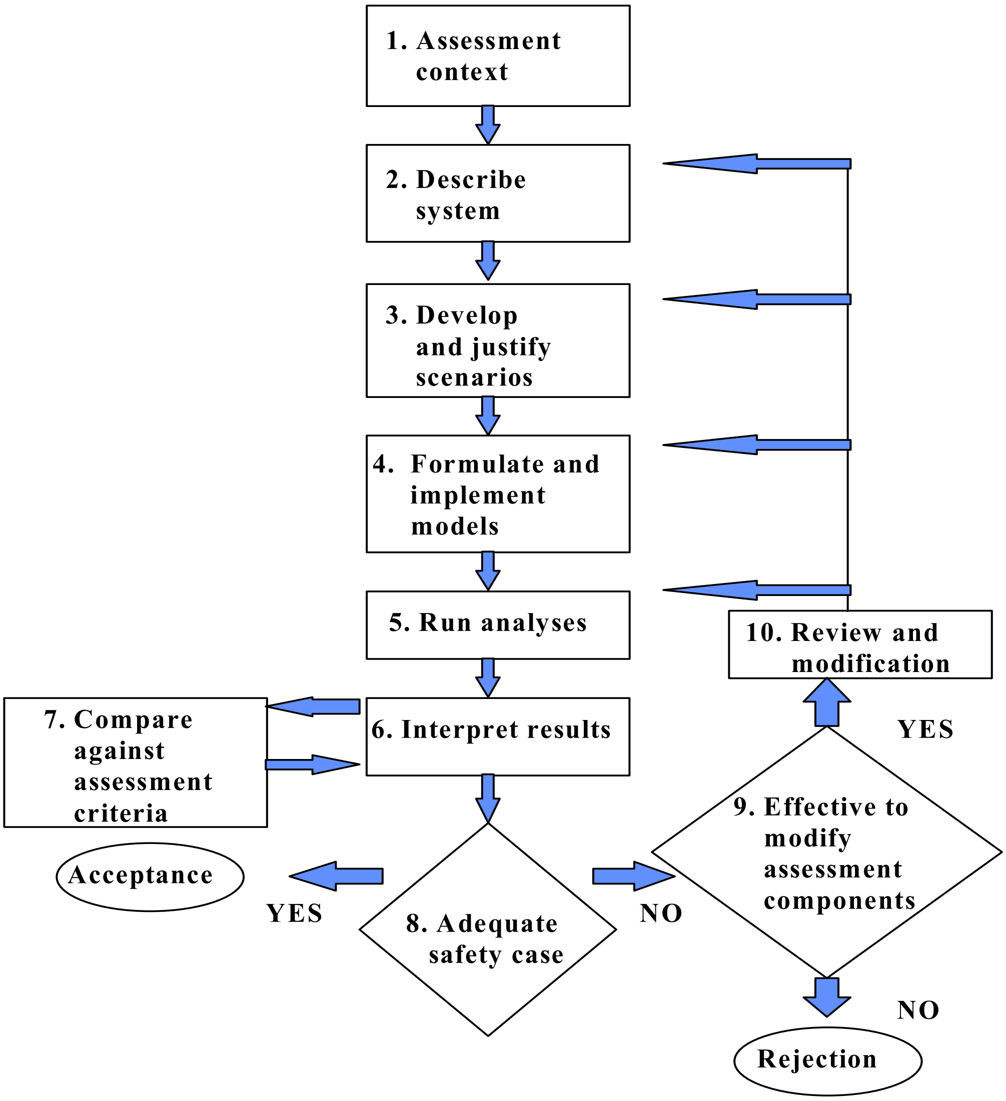

FIG. 1. The safety assessment process.

Each of these steps is briefly outlined in the following sub-sections (Sections 4.2–4.6), with emphasis being given to their use in the derivation of activity limits. In the programme for development of an actual disposal system, iterations between Step 6 and the previous steps is performed when comparing the derived activity limits with the expected or actual waste characteristics. In the light of such iterations and the analysis of associated results, modifications to operational procedures, engineering and/or other control measures are considered. As mentioned in Section 2, demonstration of the iterative aspect is beyond the scope of the illustrative cases provided in Section 5 of this publication.

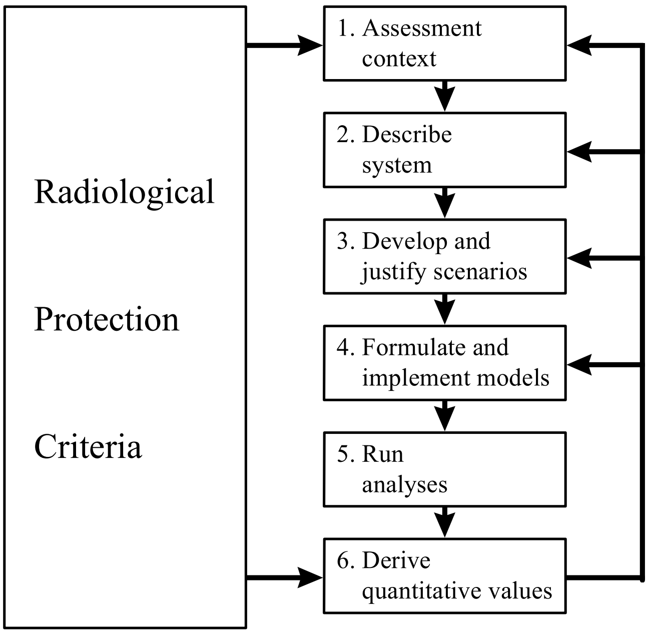

FIG. 2. The approach used for deriving activity limits for operation and post-closure periods.

4.2. ASSESSMENT CONTEXT

The assessment context establishes the framework (e.g. regulatory requirements, safety criteria) and general boundary conditions for the performance of safety assessment in an explicit and documented manner. In particular, it provides information concerning the following key aspects of the safety assessment:

— the purpose;
— the radiological protection criteria;
— the calculation end points;
— the assessment philosophy;
— the timeframes.

The general disposal system description, which can be seen as part of the assessment context, is discussed as part of the more detailed system description outlined in Section 4.3.

Each of these key aspects that need to be considered when specifying the assessment context is briefly discussed below, with particular reference to the current study.

The *purpose* of the study is demonstration of application of a safety assessment approach to derive illustrative operational and post-closure activity limits for the disposal of radioactive waste to operational and proposed near surface facilities.

In order to calculate the activity limits, it is necessary to specify the *radiological protection criteria* of interest, which are discussed in Section 3.

In most safety assessments some measure of impact on humans and/or the environment is the *calculation endpoint* by applying the safety assessment methodology in an iterative manner. The waste activity concentrations and total activity levels (i.e. the facility inventory) are usually the starting points of the assessment. In contrast, in this study, the calculation end points are the waste activity limits, which can be both concentration and total activity values, and the measures of impact, as specified by the radiological protection criteria, can be seen as the starting point of this calculation. However, in practice, the calculation of activity limits first requires a unit inventory to be assumed for which the appropriate measure of impact (dose, risk) is calculated (see Section 4.6). Assuming a linear relationship between the inventory and the impact, an activity limit is then derived for each radionuclide that meets the appropriate radiological protection criteria.

It is noted that the reason for including *the assessment philosophy* of the assessment context arises from the apparent different approaches that can be applied to the assessment of specific end points. While the nature of the end point may have been clearly defined, the nature of the assumptions used in assessment of the end point also need to be made clear. A range of approaches can be considered. The first is termed "cautious' (a cautious assumption is an assumption that will not result in the calculation end point(s) being underestimated); the second is termed "equitable" (an equitable assumption is an assumption that is physically possible and quite likely to occur). These two terms should not be considered as opposites. Indeed, in an assessment certain assumptions might be cautious, whilst other might necessarily equitable. The key issue is to document and justify the nature of each assumption in the assessment (be it cautious or equitable).

When undertaking a safety assessment to derive activity limits for the disposal of radioactive waste to a near surface disposal facility, three *timescales* need to be considered.

— The duration of the operational period (i.e. the period over which wastes are disposed of). This duration will vary from facility to facility, depending upon waste volumes and arising, the rate of waste disposals and the capacity of the site.

— The duration of the institutional control period after closure of the disposal facility. It is typically considered that a duration of between 100 and 300 years is reasonable for near surface disposal facilities. During this period, it is assumed that inadvertent human intrusion into the facility cannot occur.

— The time period for which post-closure impact calculations are undertaken. Calculations should be undertaken out to a time when it can be shown that the radiological hazard associated with the waste has reached an insignificant level. This time will vary depending upon the half-life of the radionuclides to be disposed and their activity levels. Various time periods have been used for calculations undertaken for post-closure impact assessments of near surface disposal facilities ranging up to and beyond 100 000 years. When analysing the results, it should be recognized that the reliability of quantitative safety indicators, such as dose, decreases with time due to the inherent uncertainties associated with the characterization of future environmental conditions and human habits [15, 16].

4.3. DISPOSAL SYSTEM DESCRIPTION

The disposal system can be divided into three components:

— the waste and its associated form;

— the engineered barriers of the disposal facility; and

— the geosphere and the biosphere (or the disposal site).

Radioactive waste can be classified according to its source and activity [6]. Two major categories have been identified: nuclear fuel cycle wastes; and non-nuclear fuel cycle waste. The former category is often the more important in terms of volumes of waste and associated activity concentrations. Both categories can be divided further. It is the waste derived from these sources, which are disposed of to near surface disposal facilities, and either have short lived radionuclides or low concentrations of long lived radionuclides that are of interest to this study. Since this study considers both the operational and post-closure radiological consequences of solid and solidified radioactive waste disposals, both short and long-lived radionuclides need to be considered. As indicated in Section 1, consideration of disused sealed sources is specifically excluded from this report; it is being considered in other IAEA studies.

The waste can be disposed in unconditioned or conditioned form. Different types of conditioning can be used depending upon the nature of the waste and the disposal facility (examples include: placing the unconditioned waste in bags, drums or metal boxes; compacting the waste into drums or metal boxes; grouting the waste into drums, metal boxes, or concrete containers). Conditioning can be undertaken at the disposal facility or off-site.

A near surface disposal *facility* can be constructed with differing levels of engineering. Options include above ground engineered structures, simple earthen trenches or engineered vaults a few meters deep and rock caverns several tens of meters below the earth surface. Different types of engineered barriers can be put in place according to the type of facility and to the chemical and physical properties of the wastes.

The *site characteristics* are important in the definition of appropriate engineering for the disposal facilities, in their further evolution and in the resulting radiological impact. The characteristics considered as relevant are [3]: geology, hydrogeology, geochemistry, tectonics and seismicity, surface processes, meteorology, climate, and the impact of human activities.

4.4. SCENARIO DEVELOPMENT AND JUSTIFICATION

As stated in the Safety Guide on Safety Assessment for Near Surface Disposal [8], scenarios depend on the environment and system characteristics, and on events and processes which could either initiate release of radionuclides from waste or influence their fate and transport to humans and the environment. The choice of appropriate scenarios and associated conceptual models is very important and strongly influences subsequent analysis of the waste disposal system. In some countries scenarios are specified by the regulator, although the operator may also choose to consider others. In other countries, the operator may select the scenarios and be required to justify the selection to the regulator.

The first step in identifying which of the many phenomena are relevant to the safety assessment should be to establish a checklist of events/activities that might initiate a scenario. It may be helpful, in developing a suitable list of scenarios, to consider the following initiating events/activities:

— natural processes and events;
— waste and disposal facility characteristics and operational activities;
— non-operational human activities.

Within the scope of this report, it is not appropriate to study each potential initiating event/activity as generic disposal systems have been considered. Indeed, some of them can be considered as not relevant to scoping calculations because of their low probability, or due to the site specificity of their magnitude and consequences. Furthermore too extensive screening can be avoided by considering the final consequences of the events/activities on the system, rather than the details of their individual features. Hence, for the purposes of this study, the states of the disposal system components are discussed, assuming that a given state can be attributed to one or several initiating events.

Several techniques have been used in the past to generate sets of scenarios relevant to waste disposal [17]. Indeed, techniques relevant to near surface disposal have been reviewed in the ISAM project. They include methodologies such as expert judgement (as used in this study given its scoping nature), fault tree and event tree analysis. Even if expert judgement has been extensively used, it is increasingly recognized that systematic techniques are helpful, in particular because they develop a justified and documented audit trail thereby enhancing the transparency and defensibility of the assessment. It should be noted that the conclusions reached by using different techniques are often in fact very similar; the output is a selection of a few scenarios encompassing most of the possibilities in terms of potential impact.

4.5. MODEL FORMULATION AND IMPLEMENTATION

According to the approach outlined in Section 4.1, once the scenarios have been developed their consequences in terms of the assessment context (Section 4.2) must be analysed. The scenarios must be organized into a form that is amenable to mathematical representation. A set of model–level assumptions (about dimensionality, boundary conditions, features, events, processes, etc.) is needed for each of these scenarios. These assumptions comprise the conceptual model.

A variety of approaches can be used to develop the conceptual models. Indeed, the generation of conceptual models using formal, defensible and transparent approaches is one of the issues which has been addressed in the ISAM project. Given the nature and scope of this study, experience gained from previous assessments (for example [14, 18–22]) has been used to generate the conceptual models.

The conceptual model for each scenario is then expressed in mathematical form as a group of algebraic and differential equations, which then need to be solved. These equations may be empirically and/or physically based, depending upon the level of understanding and information concerning the processes represented. More than one mathematical formulation might be appropriate for the conceptual models considered. These equations and their associated parameters form the basis of the mathematical models. Solution of the mathematical models is usually achieved by implementing one or more computer tools using analytic and/or numerical techniques. In order to allow the computer tools to be run, data for their input parameters need to be specified.

4.6. DERIVATION OF ACTIVITY LIMITS

The resulting peak doses for each scenario for a unit activity (concentration or total amount) of each radionuclide have to be compared to derive the limiting scenario for each radionuclide, i.e. the scenario potentially leading to the highest dose. To derive the activity limits, the highest dose for each radionuclide has to be compared with the dose limit, i.e. the selected radiological protection criteria. For some scenarios only the activity concentration might be relevant, whilst for others the total activity is more relevant.

For any given scenario, the activity concentration limit of each radionuclide in the waste ($Bq.kg^{-1}$ of waste) can be calculated using:

$$Conc_i = \frac{Dose_{\lim} \cdot C_{iu}}{Dose_{iu}} \tag{4.1}$$

where

$Conc_i$ is the activity concentration limit of radionuclide i for the scenario [$Bq.kg^{-1}$ of waste]

$Dose_{lim}$ is the relevant dose limit for the scenario [$Sv.y^{-1}$]

$Dose_{iu}$ is the dose resulting from the initial activity of radionuclide i in the waste contributing to the radiological impact of the scenario [$Sv.y^{-1}$]

$C_{iu} = A_{iu} / \rho_{bd} V_w$ is the initial activity concentration of radionuclide i in the waste contributing to the radiological impact of the scenario [$Bq.kg^{-1}$]

where

A_{iu} is the initial activity of radionuclide i in the waste in a waste contributing to the radiological impact of the scenario [Bq]

ρ_{bd} is the dry bulk density of the waste [$kg.m^{-3}$]

V_w is the volume of the waste contributing to the radiological impact of the scenario [m^3]

The total activity limit of each radionuclide in the waste (Bq) can be calculated using:

$$Amount_i = \frac{Dose_{\lim} \cdot A_i}{Dose_i} \tag{4.2}$$

where

$Amount_i$ is the total activity limit of radionuclide i for the scenario [Bq]

$Dose_{lim}$ is the relevant dose limit for the scenario [$Sv.y^{-1}$]

A_i is the initial activity of radionuclide i in the total quantity of waste intended for disposal [Bq]

$Dose_i$ is the total dose resulting from the initial activity A_i [$Sv.y^{-1}$]

Once activity limits for each radionuclide in the disposal facility have been established, another basic criterion to be met is that the combined doses of all radionuclides still remains under the relevant dose limit or established dose constraint if required. This is achieved by the following limiting condition or summation rule:

$$\sum_i \frac{Q}{Q_{i,l}} \leq 1 \tag{4.3}$$

where

Q_i is the actual activity of radionuclide i to be disposed [Bq or $Bq.kg^{-1}$]; and

$Q_{i,l}$ is the activity limit for radionuclide i from the most restrictive scenario, assuming radionuclide i is the only radionuclide to be disposed [Bq or $Bq.kg^{-1}$].

This is very cautious. It would be more appropriate to recognize the time dependent nature of this problem (i.e. peak doses for the different radionuclides do not occur all at the same time).

As the influence of the duration of the operational period and the institutional control period on the activity limits for short lived radionuclides is very large, it might be necessary to do the above calculations for different durations of these periods.

5. ILLUSTRATIVE APPLICATION OF THE APPROACH

In this section, the approach outlined in Section 4 for the derivation of activity limits is used to derive illustrative activity limits for two example near surface disposal systems with differing characteristics based on the evaluation of their operational and post-closure safety. Detailed information on these systems and on the site characteristics is provided in the following sub-sections.

5.1. ASSESSMENT CONTEXT

5.1.1. Purpose

The purpose is to apply the safety assessment approach outlined in Section 4 to derive illustrative operational and post-closure activity limits for the disposal of radioactive waste to two example near surface disposal facilities. The illustrative limits can be used as a benchmark against which to compare limits for specific disposal systems. Since the use of the assessment approach allows the derivation of limits in a clear and well documented manner, the reasons for any differences between these illustrative limits and system specific limits can be identified, provided that the derivation of the system specific limits has been clearly documented.

5.1.2. Radiological protection criteria

For the purposes of this illustration, dose limits of 20 $mSv.y^{-1}$ for workers and 1 $mSv.y^{-1}$ for members of the public are used.

5.1.3. Calculation end points

The calculation endpoints for this illustration are radionuclide activity concentration limits and total activity limits that correspond to an exposure of 20 $mSv.y^{-1}$ (dose limit for workers), and 1 $mSv.y^{-1}$ (dose limit) for members of the public. A probability of unity is assumed for all scenarios assessed.

5.1.4. Assessment philosophy

A generally cautious assessment philosophy is adopted that is unlikely to result in an over-estimation of activity limits. However, an attempt is made to avoid the compounding of pessimisms resulting from the adoption of cautious assumptions at every stage of the assessment.

5.1.5. Time frames

Consistent with the discussion in Section 4.2, the following key timeframes are considered for the current illustration:

— an operational period of 50 years;
— subsequent representative institutional control periods of 30, 100 and 300 years; and
— a time period for post-institutional control calculations that allows the demonstration that the peak dose has been reached for each scenario assessed (recognising that the reliability of dose calculations decreases with time – see Section 4.2).

5.2. DISPOSAL SYSTEM DESCRIPTION

General operations at a disposal facility may include some waste processing and storage activities. However, for the purposes of the illustration, it is assumed that the waste is immediately disposed of upon arrival at the facility.

5.2.1. Waste characteristics

Through consideration of the results of a number of safety assessments (for example [7, 14, 18–21, 23, 24]), the radionuclides considered in the majority of them during the operational and post-closure periods, and over the timescales of concern in this study, have been identified. These are given in Table II for the operational period and in Table III for the post-closure period. In assessing the impact of these radionuclides, it might be necessary to consider the in-growth of associated daughters. It is recognized that for any specific disposal facility, the list of radionuclides to be considered will depend upon factors such as the source and nature of the waste streams, the disposal system characteristics and the duration of the institutional control period. Therefore, the radionuclides in Tables II and III are addressed for the illustrative purpose of the study.

TABLE II. DISPOSED RADIONUCLIDES CONSIDERED IN THE ILLUSTRATIVE OPERATIONAL SAFETY ASSESSMENT OF NEAR SURFACE DISPOSAL FACILITIES

^{3}H	^{90}Sr	^{137}Cs	^{232}Th
^{10}Be	^{93}Zr	^{144}Ce	^{234}U
^{14}C	^{94}Nb	^{147}Pm	^{235}U
^{22}Na	^{99}Tc	^{151}Sm	^{238}U
^{41}Ca	^{106}Ru	^{152}Eu	^{237}Np
^{54}Mn	^{110m}Ag	^{154}Eu	^{238}Pu
^{55}Fe	^{121m}Sn	^{204}Tl	^{239}Pu
^{59}Ni	^{125}Sb	^{210}Pb	^{240}Pu
^{63}Ni	^{126}Sn	^{226}Ra	^{241}Pu
^{60}Co	^{129}I	^{228}Ra	^{241}Am
^{65}Zn	^{134}Cs	^{227}Ac	

TABLE III. DISPOSED RADIONUCLIDES CONSIDERED IN THE ILLUSTRATIVE POST-CLOSURE SAFETY ASSESSMENT OF NEAR SURFACE DISPOSAL FACILITIES

^{3}H	^{90}Sr	^{151}Sm	^{237}Np
^{14}C	^{93}Zr	^{226}Ra	^{238}Pu
^{41}Ca	^{94}Nb	^{228}Ra	^{239}Pu
^{55}Fe	^{99}Tc	^{232}Th	^{240}Pu
^{59}Ni	^{129}I	^{234}U	^{241}Pu
^{63}Ni	^{134}Cs	^{235}U	^{241}Am
^{60}Co	^{137}Cs	^{238}U	

The waste can be disposed in unconditioned or conditioned form. Different types of conditioning can be used depending upon the nature of the waste and the disposal facility. The nature of the waste conditioning for the illustrative assessment is described in Section 5.2.2.

Note that for safety assessments considering post-closure scenarios, certain radionuclides could be excluded on the basis of their short half-life or their impact on post-closure safety. Certain radionuclides listed in Table 5.1 are excluded from Table 5.2 because, following the minimum 30 year institutional control period (Section 5.1.5), the associated activity is assumed to have decayed to insignificant amounts, whilst others were considered important in previous safety assessments. In practice, performing screening calculations can substantiate this assumption. In this illustration no such screening calculations are undertaken.

5.2.2. Engineered barrier characteristics

For the purposes of this illustration, the disposal facility is assumed to consist of ten disposal units (two rows of five units) with a site boundary during the operational period and institutional control period located 70 m from the edge of the disposal units (Fig. 3). Two categories of disposal units are considered which represent the simple and highly engineered disposal facilities.

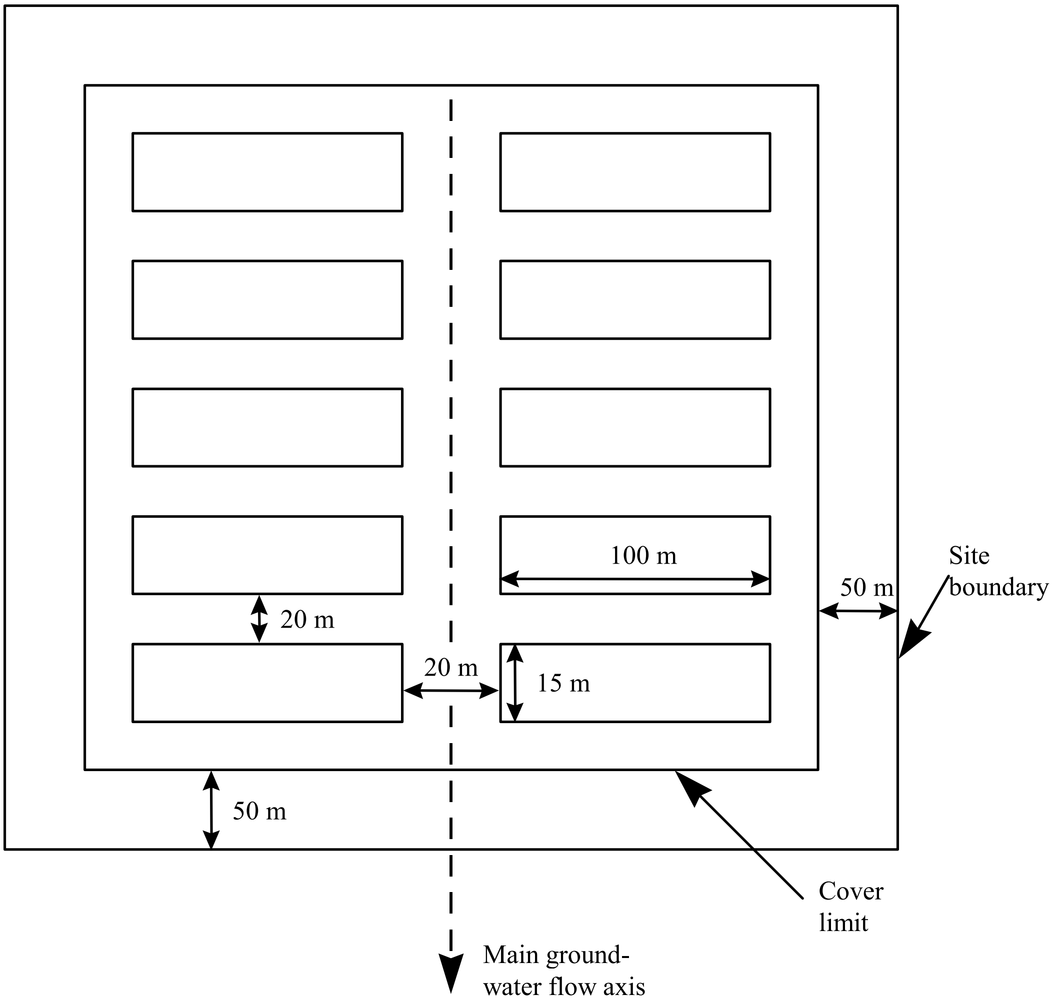

FIG. 3. Illustrative near surface disposal facility layout.

At one end, there is the *minimally engineered disposal unit (trench)* (Figs 4, 5). The characteristics of the trench are adapted from the trench used in the Test Case 2C of the Co-ordinated Programme on the Safety Assessment of Near Surface Radioactive Waste Disposal Facilities (NSARS). Relevant parameters and their associated data values are given in Appendix I. The trench is excavated into the ground and contains loose tipped waste. It is noted that the loose tipping practices considered in this illustration might not be appropriate for all waste types that are suitable for disposal in near surface facilities. Nevertheless, its consideration is indicative of a conservative approach to assessment of operational safety.

It is assumed that during the operational period unconditioned waste is driven by a truck to the tipping face of the currently open trench. It is then tipped into the trench from the back of the lorry. As each load of waste is tipped, the tipping face moves towards the end of the trench by this tumble tipping process (Fig. 4). Large items, such as those arising from the decommissioning of nuclear installations, are lowered into the trench by a mobile crane. The upper surface of waste is covered by 1 m of uncontaminated soil/fill material, a geotextile layer, stones and finally ashes/small aggregate in order to ensure a stable and contamination free interface between the waste and the tipping vehicles (Fig. 4). It is assumed that the tipping face remains uncovered because of the practical difficulties in achieving a uniform thickness of cover, and the associated problem of wasted disposal volume. During the filling of the trench, it is conservatively assumed that infiltrating rainwater is collected by a drainage system that discharges directly to a local water body. Upon completion of filling, the trench is closed by placing a further 1 m of uncontaminated soil/fill on top of the existing 1 m cover, giving a total cap thickness of 2 m (Fig. 5), and the drainage system for the trench is decommissioned.

At the other end of the spectrum, there is the *heavily engineered disposal unit (vault)* (Figs 6, 7). The characteristics of the vault are adapted from the vault used in the NSARS Test Case 1 exercise [25]. Relevant parameters and their associated data values are given in Appendix I. The vault is excavated into the ground and lined with concrete.

It is assumed that during the operational period conditioned waste packages (metallic drums and concrete containers) are driven by truck into the facility and lifted into the currently operating vault. The waste is assumed to be grouted into the drums and concrete containers at an off-site installation. For the purposes of filling, the vault is divided into four sub-vaults by internal concrete walls (Fig. 6). The truck stops in the sub-vault adjacent to the one currently accepting waste packages. A crane operator is situated in a cabin directly above the sub-vault in which the truck is unloaded. Each waste package is picked up from the truck by the crane and lowered remotely into the disposal sub-vault. The waste packages are disposed layer by layer, each layer is immobilized in grout, which is pumped into the sub-vault. Once the sub-vault has been filled, concrete shielding is poured on top of the sub-vault. During the filling of the vault, it is cautiously assumed that infiltrating rainwater is collected by a drainage system and monitored for activity prior to release to a local water body. Upon completion of filling, the vault is closed by placing 3 m of uncontaminated material (included a 1 m layer of rolled clay (Fig. 7)). Once all the ten vaults have been completed it is assumed that the drainage system for the vaults is decommissioned.

It is recognized that the vault concept is potentially more relevant than the trench concept for many countries. However, the trench concept might be of relevance for certain wastes with very low levels of activity and it can be seen as a starting point for an iterative assessment in which additional engineered barriers and operational controls can be progressively introduced until the desired degree of safety is achieved.

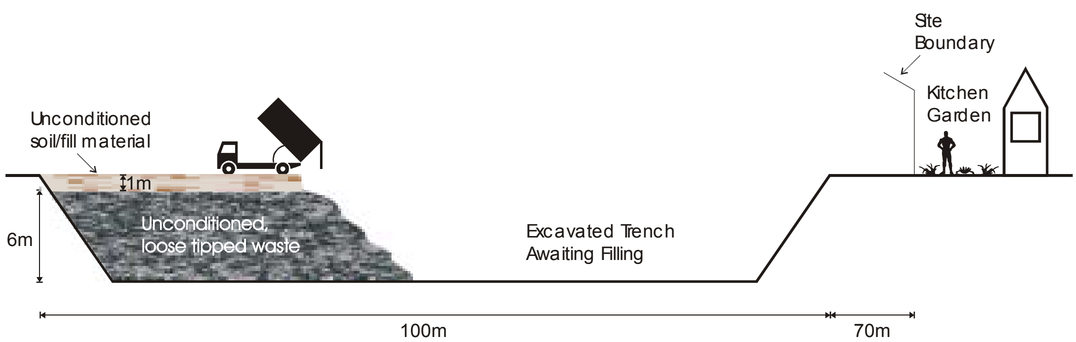

FIG. 4. Length-wise cross-section through an operational trench.

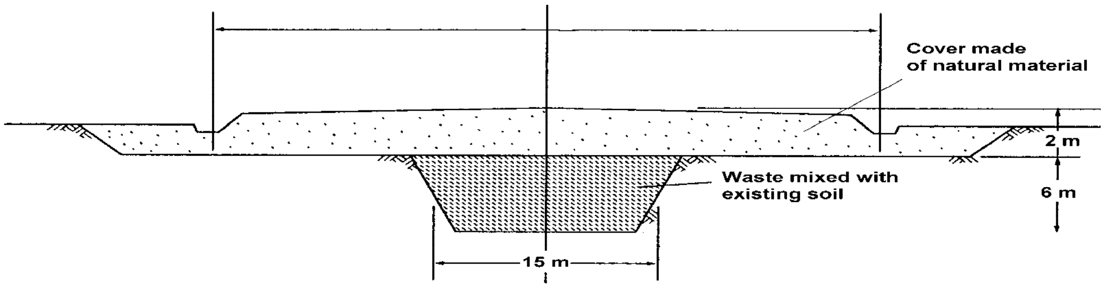

FIG. 5. Width-wise cross-section through a closed trench.

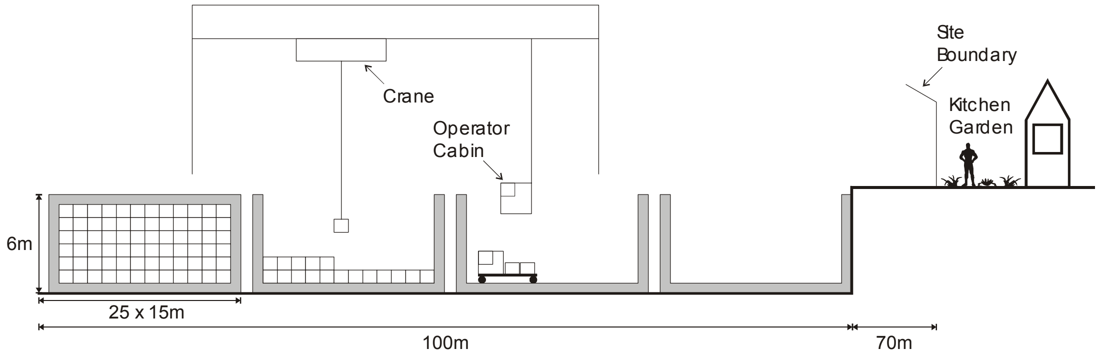

FIG. 6. Length-wise cross-section through an operational vault.

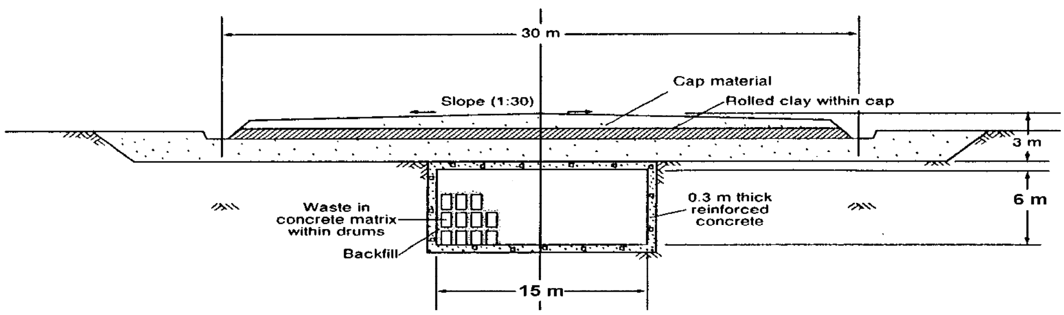

FIG. 7. Width-wise cross-section through a closed vault.

5.2.3. Geosphere and biosphere characteristics

Two geospheres are considered: a high permeability ("sandy") geosphere; and a low permeability ("clay") geosphere. The sandy geosphere is adapted from the NSARS Test Case 1 exercise [25], whilst the clay geosphere description has been developed specifically for this study. Parameter values for the geospheres are given in Appendix I. It is assumed that for the sandy geosphere, the geosphere-biosphere interface is a water abstraction well located at the site boundary, whilst for the clay geosphere, it is a river located 1500 m from the site boundary.

Two biosphere states are considered, temperate and arid. Details are provided in Appendix I. Both biospheres can support agricultural activity. It is assumed that the amount of precipitation and the nature of the geosphere material will affect the level of the water table. For the sandy geosphere, the water table is assumed to be below the disposal facility for both biosphere states, the depth to the water table being greater for the arid biosphere. For the clay geosphere, the water table is assumed to be at the base of the disposal unit in the temperate biosphere and below the disposal facility in the arid biosphere. It is recognized that over the timeframes of interest (Section 5.1.5) both the biosphere and human habits can significantly evolve. However, for the purposes of this illustration it is assumed that constant conditions and habits are maintained.

Figure 8 shows the combinations of disposal facility, with different geosphere and biosphere characteristics. It can be seen that there are a total of eight disposal systems identified.

5.3. SCENARIO AND MODEL DEVELOPMENT – OPERATIONAL PERIOD

5.3.1. Development of scenarios

The scenario development method, used for the illustrative purposes of this study, consists of:

- defining the main components to be considered in the assessment, and their associated states;
- constructing the combinations of states; and
- checking the scenarios generated and grouping them into main categories.

The first component is the waste. This can be mixed unconditioned waste or conditioned with a matrix (e.g. grouted), put or not put in containers. Consistent with the disposal system description (Section 5.2), their possible states of the waste for this study are given below.

- Completely conditioned off site. They are conditioned in grouted concrete containers and metal drums. heir containment is designed to minimize the release of radionuclides.

- Unconditioned waste for direct disposal. They are raw solid waste that do not provide any containment barrier against leaching or release by air.

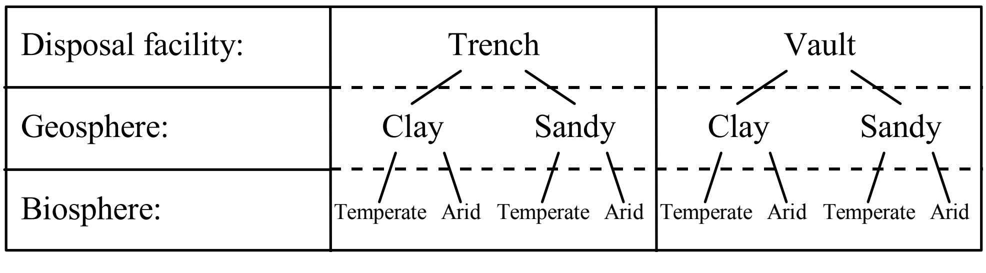

FIG. 8. Combinations of disposal system characteristics.

The second component is the design of the disposal facility. For this study two designs are considered:

— the trench with its minimally engineered features; and
— the vault with its heavily engineered features.

The third component is the operation of the disposal facility. Two possible states of the operational conditions can be envisaged:

— normal operation conditions: the operation of the facility complies with what is expected by adequate design, construction, procedures, controls, monitoring, that ensure adequate performance levels of equipment and personnel.
— abnormal operation conditions: failure to meet performance objectives could result from non respect of waste specifications, equipment failure, operating error, or also from events and processes generated outside the facility (natural and human).

5.3.2. Identification of scenarios

Having defined the main components of the disposal system and its operation with their different states in Section 5.3.1, it is possible to combine them to obtain Fig. 9. The associated scenarios are screened and discussed below in order to provide the set of scenarios given in Fig. 10 for which calculations are undertaken.

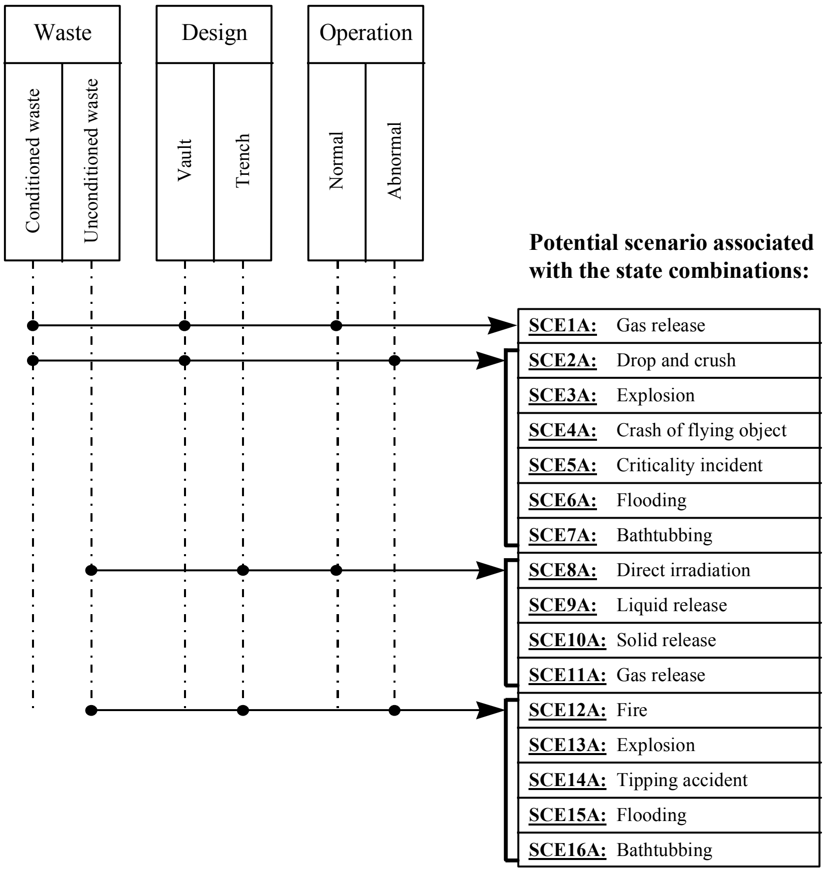

FIG. 9. Generation of a set of scenarios (SCE) for the operational period according to various disposal and operational components.

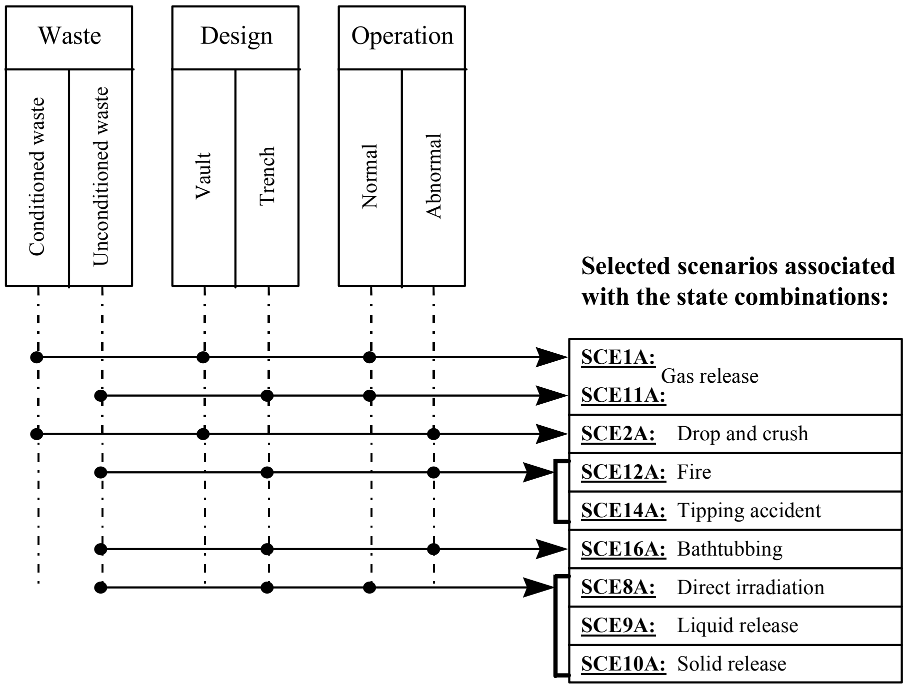

FIG. 10. The screened set of operational scenarios for further consideration.

Gas release corresponding to SCE1A and SCE11A in Fig. 9 should be considered under normal operational conditions both for conditioned packages in the vault and unconditioned wastes in the trench. Volatile radionuclides may originally be contained in the wastes (such as ^{3}H and ^{14}C) or generated in situ from the decay of parent radionuclides (for example ^{222}Rn). Their release will depend on a number of factors, such as the inventory, the leak tightness of the waste package, the design of the disposal facility, initial inventory of organic materials, microbial activity. The associated exposure of workers and members of the public will depend on the possibility of the accumulation of gases in relatively unventilated areas of the disposal facility, and on the release of gases into the wider environment. Workers could be exposed inside and outside the trench and vault. A critical population group downwind from the disposal facility or its ventilation outlet will be the members of the public liable to receive the highest dose from the gases.

The SCE2A (drop and crush of waste packages) (Fig. 9) is possible in operating a vault; it corresponds to an abnormal situation. Despite the fact that only one package is generally handled for disposal at a time, a drop onto other packages that have already been unloaded cannot be excluded. Exposure to workers will be principally due to direct irradiation and especially in case of damage of the waste package during repository operation. Indeed, vaults are generally operated in open air and the dilution volume is important, hence inhalation will be low but might need to be assessed. This scenario could also involve members of the public living near to the facility.

Spontaneous internal explosion involving one waste package to be disposed in vault or unconditioned wastes to the trench, or explosion of accumulated gases should be completely avoided by prevention measures. Indeed adequate requirements for waste acceptance and disposal facility design are assumed to be specified and respected. On the other hand, both internal and external explosion involving wastes would be more likely during waste conditioning operations. However, conditioning of waste at the site is excluded for the current illustrative exercise. Hence, for this example, both SCE3A and SCE13A in Fig. 9 are excluded.

The results of the impacts of flying objects on a facility cannot be completely dismissed. However the derivation of activity limits from these events for either the wastes or waste packages is not considered in this study on the basis of the low likelihood of such events and the wider (non-radiological) consequences that would occur. Rather the approach to be recommended is to reduce the probability of occurring of such events like crashes by restricting aircraft movements in the air space in the vicinity of the facility. In addition, the potential consequences can be minimized by repository design and operational procedures such as timely sealing of the disposal cells with cover of adequate thickness. In conclusion SCE4A in Fig. 9 is not taken into account, for this example, neither for the vault nor for the trench.

Criticality is not expected in wastes suitable for near surface disposal. However, low and intermediate active wastes containing considerable amounts of fissile material and/or moderator materials do exist in some countries and have to be disposed of. Consequently, acceptance criteria for wastes suitable for near surface disposal should include activity limits and other controls (such as controls on the spatial distribution of such wastes in the facility) aimed at avoiding criticality. Nevertheless, it is not in the scope of the present study to derive those limits for fissile material. Thus SCE5A in Fig. 9 is excluded.

In the case of SCE6A, SCE15A, SCE7A and SCE16A in Fig. 9 it is assumed that flooding (SCE6A and SCE15A) and/or bathtubbing (SCE7A and SCE16A) can occur. Both scenarios are associated with abnormal conditions (failure of the drainage systems and absence of mitigating actions by the site operator) and the impact of both is expected to be broadly similar. . The impact of bathtubbing might be slightly greater since the disposal unit is assumed to be full of waste and so is considered further. The physical and hydraulic characteristics of the vault mean that bathtubbing can only occur for the vault after 60 years. Thus, since an operational period of 50 years is assumed (Section 5.1.5), bathtubbing in the vaults does not have to be considered for the operational period, but for the post-closure period (Section 5.4.3). In contrast, calculations are required for bathtubbing in the trenches (SCE16A) since the physical and hydraulic characteristics of the trench can result in bathtubbing after five years in the trench in clay under temperate conditions. It is recognized that the bathtubbing scenario is a highly conservative scenario for the trench that can be screened out through suitable facility design and/or location. Nevertheless, it is considered to be a bounding scenario for initial consideration in the illustrative assessment of disposal to the trench facility as the contaminated water resulting from accumulation could for example contaminate a garden adjacent to the disposal facility. Thus SCE16A is considered further, whilst SCE6A, SCE15A and SCE7A are all screened out.

Exposure due to direct irradiation of workers and the public is kept to acceptable levels by restricting the dose rate from conditioned wastes in the vaults. Generally the disposal of unconditioned wastes in a trench provides a higher risk of exposure by direct irradiation even in normal operational conditions. Therefore, it is necessary to check if SCE8A in Fig. 9 could be relevant to derive activity limits for unconditioned wastes in the trench.

In the case of unconditioned wastes to be disposed of in the trench, releases in normal operation situations exist and need to be considered. In contrast, conditioned wastes and the operation of the vault are designed to prevent releases under normal conditions. For liquid releases from the trench, it is assumed that rain water will infiltrate through the waste and result in the generation and migration of leachate to the trench drainage system. As noted in Section 3.6, the drainage system is assumed to flow to a local watercourse, which is used by members of the public. Thus SCE9A in Fig. 9 must be studied to determine if it is a limiting scenario. For unconditioned wastes tipped into the trench, release in the form of dust is assumed to occur in the associated scenario SCE10A (Fig. 9). External irradiation and inhalation need to be considered for the workers, whilst exposure pathways associated with the inhalation, external irradiation and deposition of suspended dust need to be assessed for the public.

Fire can be excluded for conditioned waste during unloading operations into vaults with regard to operational conditions and to appropriate intervention procedures. Nevertheless, for unconditioned wastes tipped onto trench, fire (SCE12A) (Fig. 9) must be considered. Workers and members of the public are potential exposure groups.

In the case of accidental operations during the tipping of the waste into the trench (SCE14A) (Fig. 9), direct contact has to be taken into account for workers for the situation where the uncontaminated cover is not yet spread over the waste.

5.3.3. Selection of scenarios

In light of the above discussion, it is possible to propose a limited and justified set of operational scenarios to be taken into account as a basis for deriving the illustrative activity limits. For this study, the scenarios considered are presented in Fig. 10 and listed below:

- the gas release for conditioned wastes in a vault in normal condition (the gas release scenario SCE1A);
- the dropping and crushing of a waste package to be disposed into a vault during unloading (the drop and crush scenario SCE2A);
- the direct irradiation for unconditioned waste in a trench in normal conditions (the direct irradiation scenario SCE8A);
- the liquid release from unconditioned waste in a trench in normal conditions (the liquid release SCE9A);
- the solid release from unconditioned waste in a trench in normal conditions before the covering of the waste (the solid release scenario SCE10A);
- the gas release for unconditioned wastes in a trench in normal conditions (the gas release scenario SCE11A);
- the fire in the unconditioned waste tipped into a trench before covering (the fire scenario SCE12A);
- the accidental spreading of waste in a trench in abnormal conditions (the trench tipping accident scenario SCE14A);
- the bathtubbing scenario for the trench in abnormal conditions (the bathtubbing scenario SCE16A).

Human intrusion scenario has not been considered as it is assumed that adequate control over the disposal facility site is ensured during operation, closure and post-closure period. It is recognized that many of the above scenarios, such as bathtubing are avoidable through the implementation of appropriate operational and/or engineering and optimization of the disposal system design. For example the impact of the liquid release could be reduced by ensuring that the leachate is collected, monitored and, if necessary, treated before discharge to the river. In an iterative study in which such measures can be introduced following the first pass through the assessment process, the scenarios can be re-assessed and screened out in light of modifications to operation practices and engineering design. However, as noted in Section 2, only one pass through the assessment process is being undertaken for the illustrative cases. The above set of scenarios should be seen as a conservative starting point for this first iteration through the assessment process.

5.3.4. Model formulation and implementation

Using the approach outlined in Section 4.5, the conceptual models summarized in Figs 11–19 can be generated for the various scenarios identified in Section 5.3.3. In each case, the conceptual model identifies (Table IV):

— the contaminant release mechanisms and media (i.e the mechanism causing the release of radionuclides from the waste, and the media in which the radionuclides are released);

— the contaminant transport media and mechanisms (i.e. the media in which and through which the radionuclides move before reaching humans, and the associated transport processes);

— the human exposure mechanisms (i.e. the pathways through which humans are exposed to the radionuclides).

Ways by which the radiological impact of the various scenarios can be quantitatively assessed are indicated in Appendix III. A suitable mathematical model has been developed for each scenario, based on the associated conceptual model. Each model describes the source term for the scenario and the resulting dose assessment. More detailed mathematical models might be required for certain assessments, but the current models are considered to be appropriate for the purposes of this illustration.

The data required for the solution of the mathematical models relate to the disposal system (disposal facility, geosphere and biosphere) (Appendix I), human exposure (Appendix III) and radionuclides and elements of concern (Appendix IV). Sources of the data for the disposal system are mostly adapted from the NSARS study [25]. Data relating to human exposure for the leaching scenario are mostly adapted from the Complementary Studies exercise of BIOMOVS II [26], those for the other scenarios have been adapted from NSARS Test Case 1 exercise [25] and other appropriate sources specified in Appendix III. Radionuclide and element data are taken from a number of sources; relevant references are indicated at the end of each table in Appendix IV. Data are provided in Appendix IV for the full range of radionuclides identified in Table II and their daughters.

Calculations of the radiological impact associated with each combination of operational scenario and disposal system (i.e. calculation case) are summarized in Table V. Parameter values differ between the different calculation cases only for the liquid release scenario for the trench (SCE9A). For this scenario values differ for both disposal facility parameters (sorption coefficient and infiltration rate) and biosphere parameters (sorption coefficient, infiltration rate, soil erosion rate, irrigation rate and the flow rate in the river) to reflect the different site characteristics (Section 5.2.3).

The computational tools used to implement and solve the mathematical models are described in Appendix V.1.

TABLE IV. CONTAMINANT RELEASE MECHANISMS AND MEDIA, TRANSPORT MEDIA AND MECHANISMS, AND HUMAN EXPOSURE MECHANISMS FOR THE OPERATIONAL SCENARIOS CONSIDERED

	Contaminant Release Mechanisms	**Contaminant Release Media**	**Contaminant Transport Media**	**Contaminant Transport Mechanisms**	**Human Exposure Mechanisms**
***SCE1A:* Gas release (vault)**	Volatilization Degradation Radioactive Decay	Gas	Atmosphere (gas)	Diffusion Dispersion	Inhalation of gas and vapour
***SCE2A:* Drop and Crush (vault)**	–	–	–	–	External irradiation from waste
***SCE8A:* Direct irradiation (trench)**	–	–	–	–	External irradiation from waste
***SCE9A:* Liquid release (trench)**	Leaching	Leachate	Drains River Soil Crops Animals Atmosphere (dust) Water (irrigation, drinking)	Discharge from drains River flow Water abstraction for irrigation and drinking water Foliar interception Root uptake Adsorption Ingestion of water, pasture and soil by cows Leaching Erosion	Ingestion of water, crops, and animal products Inhalation of dust External irradiation from soil
***SCE10A:* Solid release (trench)**	Suspension	Dust	Atmosphere (dust) Soil Crops	Deposition Foliar interception Root uptake	Inhalation of dust External irradiation from dust and soil Ingestion of crops Inadvertent ingestion of soil
***SCE11A:* Gas release (trench)**	Volatilization Degradation Radioactive Decay	Gas	Atmosphere (gas)	Dispersion	Inhalation of gas
***SCE12A:* Fire (trench)**	Fire	Particles Gases Vapours	Atmosphere (particles, gases and vapours) Soil Crops	Deposition Foliar interception Root uptake	Inhalation of particles, gases and vapours External irradiation from particles and soil Ingestion of crops Inadvertent ingestion of soil
***SCE14A:* Tipping accident (trench)**	Suspension Contamination	Dust particles (deposed)	Atmosphere	Dispersion Deposition	External irradiation from soil External irradiation from inadvertent contamination Ingestion Inhalation
***SCE16A:* Bathtubbing (trench)**	Leaching	Leachate	Overflow leachate Soil Atmosphere (dust) Crops	Overflow of leachate Suspension Root uptake Adsorption	Ingestion of crops Inadvertent ingestion of soil Inhalation of dust External irradiation from soil

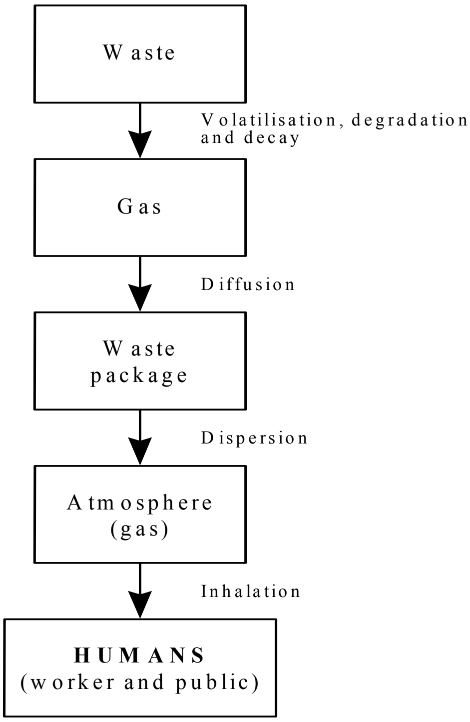

FIG. 11. Simplified representation of the conceptual model for the Operational Vault Gas Release Scenario (SCE1A).

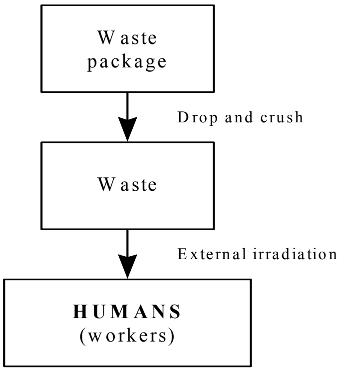

FIG. 12. Simplified representation of the conceptual model for the Operational Vault Drop and Crush Scenario (SCE2A).

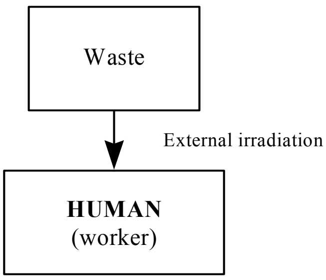

FIG. 13. Simplified representation of the conceptual model for Operational Trench Direct Irradiation Scenario (SCE8A).

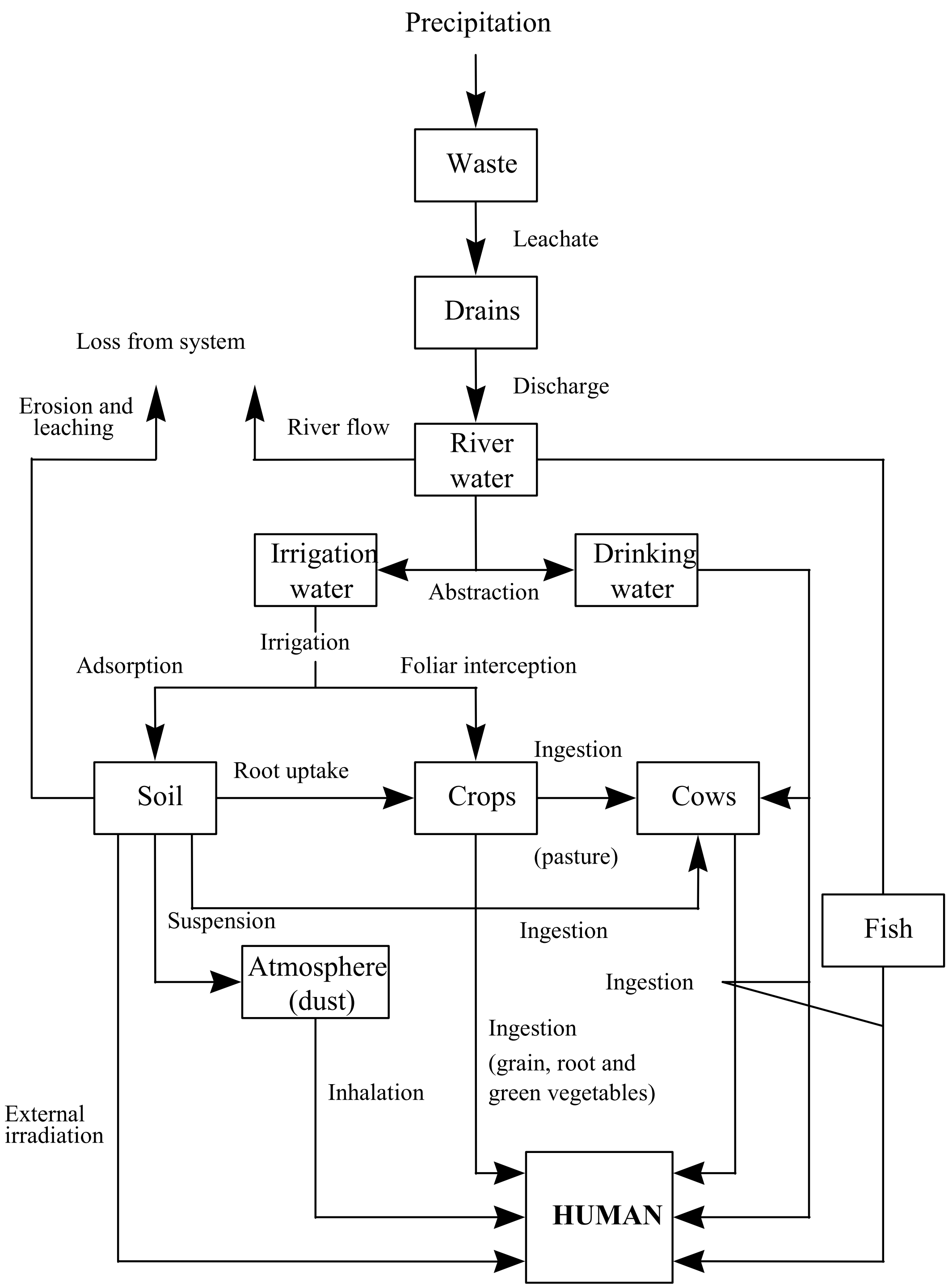

FIG. 14. Simplified representation of the conceptual model for the Operational Trench Liquid Scenario (SCE9A).

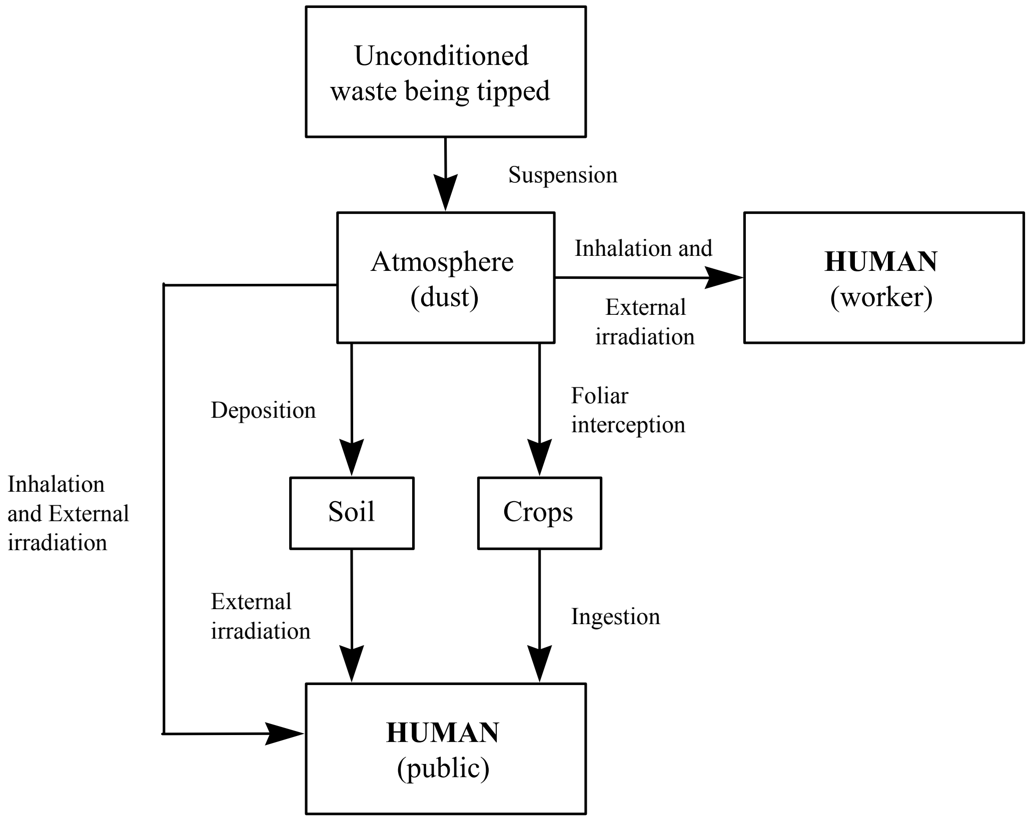

FIG. 15. Simplified representation of the conceptual model for the Operational Trench Solid Release Scenario (SCE10A).

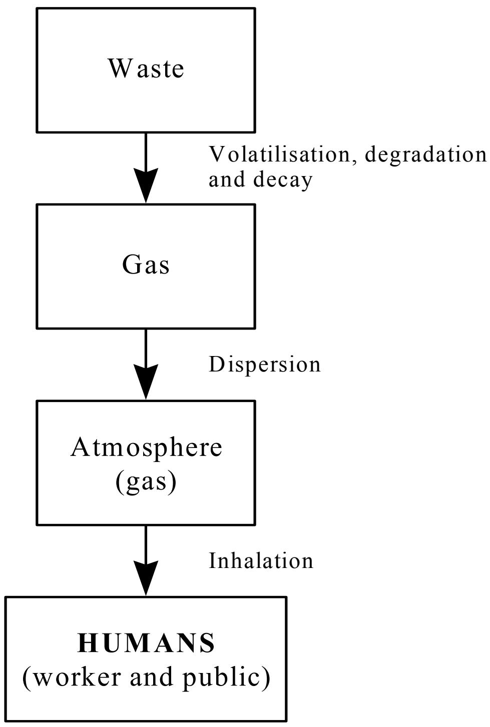

FIG. 16. Simplified representation of the conceptual model for the Operational Trench Gas Release Scenario (SCE11A).

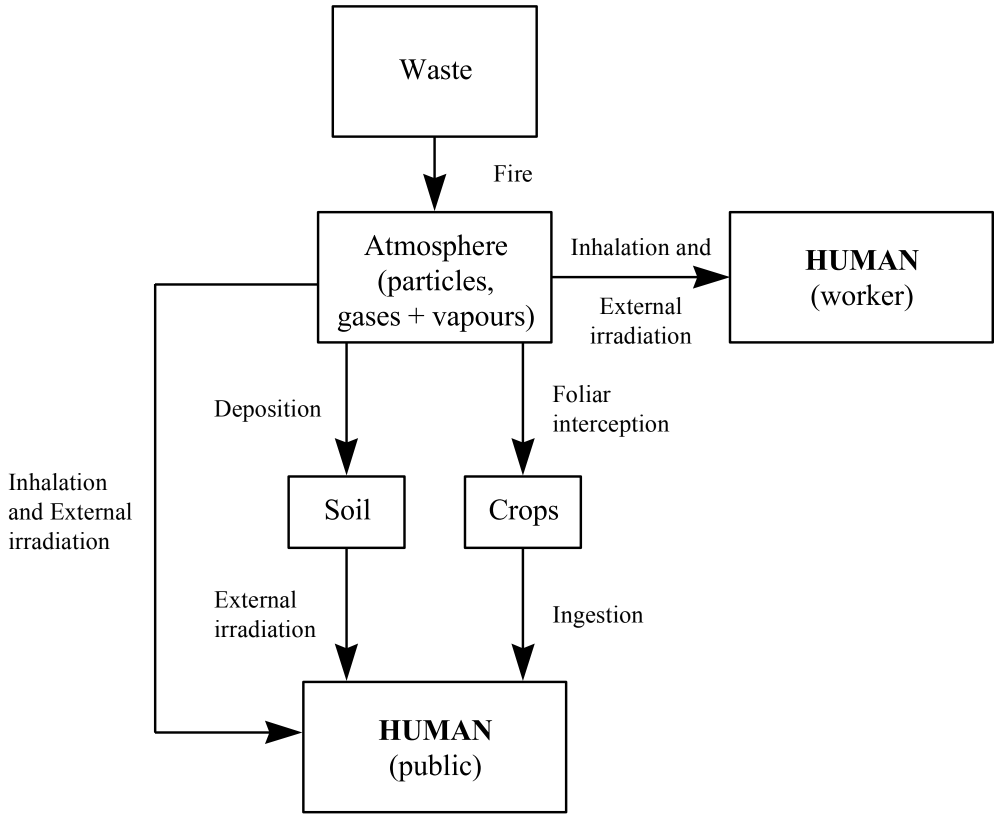

FIG. 17. Simplified representation of the conceptual model for the Operational Trench Fire Scenario (SCE12A).

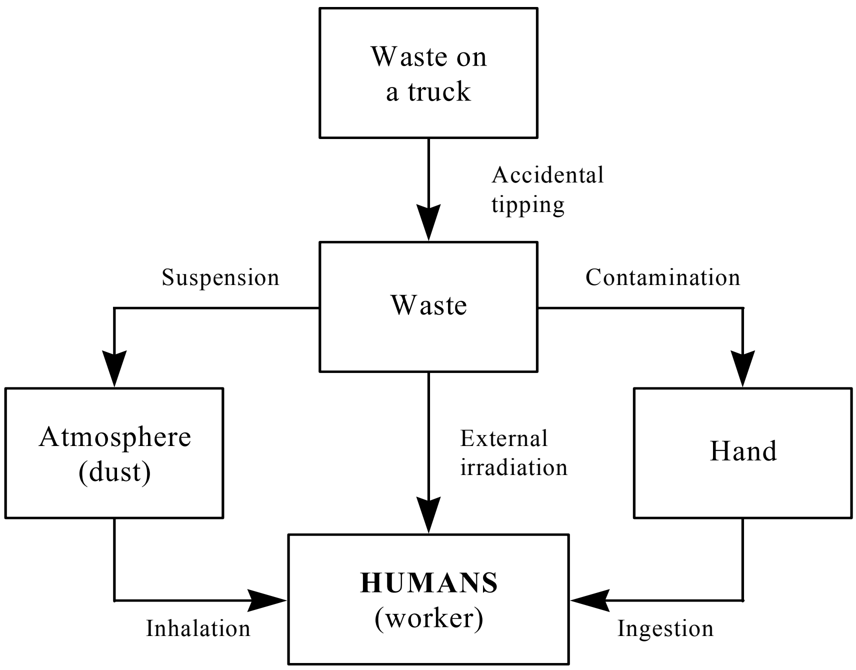

FIG. 18. Simplified representation of the conceptual model for the Operational Trench Tipping Accident Scenario (SCE14A).

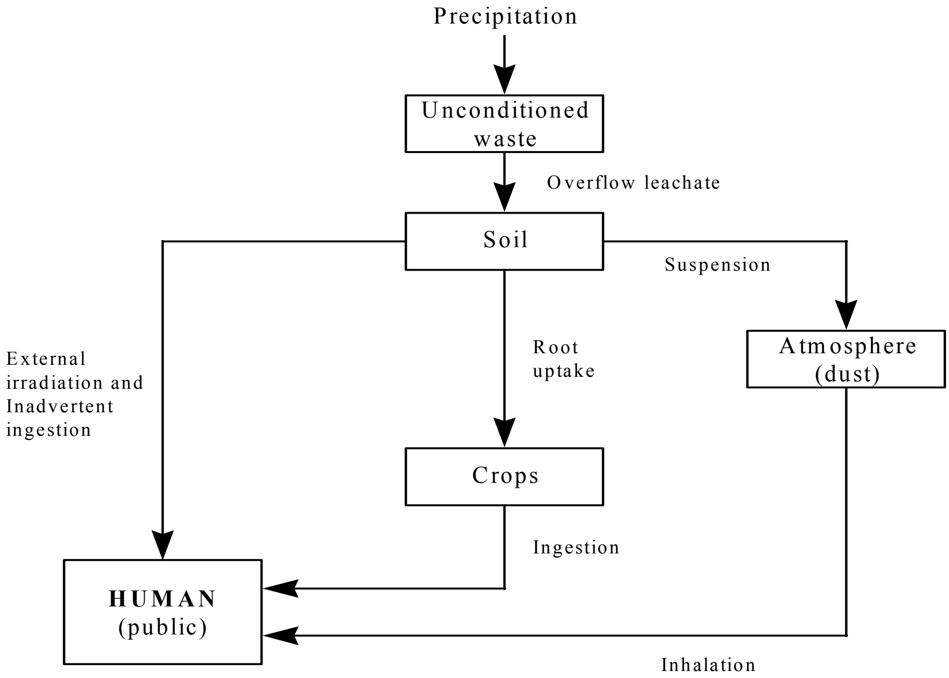

FIG. 19. Simplified representation of the conceptual model for the Operational Trench Bathtubbing Scenario (SCE16A).

TABLE V SUMMARY OF CALCULATION CASES FOR THE OPERATIONAL PERIOD

Scenario	**Temperate**				**Arid**			
	Sand		**Clay**		**Sand**		**Clay**	
	Trench	**Vault**	**Trench**	**Vault**	**Trench**	**Vault**	**Trench**	**Vault**
SCE1A: Gas release (vault)	×	✓	×	(1)	×	(1)	×	(1)
SCE2A: Drop and Crush (vault)	×	✓	×	(1)	×	(1)	×	(1)
SCE8A: Direct irradiation (trench)	✓	×	(2)	×	(2)	×	(2)	×
SCE9A: Liquid release (trench)	✓	×	✓	×	✓	×	✓	×
SCE10A: Solid release (trench)	✓	×	(2)	×	(2)	×	(2)	×
SCE11A: Gas release (trench)	✓	×	(2)	×	(2)	×	(2)	×
SCE12A: Fire (trench)	✓	×	(2)	×	(2)	×	(2)	×
SCE14A: Tipping accident (trench)	✓	×	(2)	×	(2)	×	(2)	×
SCE16A: Bathtubbing (trench)	×(3)	×(3)	✓	×(4)	×(3)	×(3)	×(3)	×(3)

Notes:
- ✓ Calculation required.
- × Calculation not required since scenario and disposal system combination is not possible based on scenario description (Section 5.3) and data given in Appendix I, II and III.
- (1) The same calculation as for the temperate, sand, vault case.
- (2) The same calculation as for the temperate, sand, trench case.
- (3) Calculation not required since infiltration rate and hydraulic characteristics of disposal system do not result in bathtubbing (see data given in Appendix I, II and III).
- (4) Infiltration rate and hydraulic characteristics of disposal system result in bathtubbing only after 60 years of closure of vaults (see data given in Appendix I, II and III). Therefore, this case is considered under the post-closure period calculations (see Section 5.4.3).

5.4. SCENARIO AND MODEL DEVELOPMENT – POST-CLOSURE PERIOD

5.4.1. Development of scenarios

The same scenario development method as used for the operational scenarios can be used for the post-closure scenarios. It:

- defines the main components to be considered in the assessment, and their states;
- constructs the state combinations; and
- checks the scenarios generated and groups them into main categories.

The first component is the waste. This can be unconditioned or conditioned waste with a matrix (e.g. grouted), put or not put in containers. Their states are chosen as:

- put in a waste container and unaltered: only a given minimal amount of water can flow through the waste container and leach the waste (relevant only to the vault facility);
- partly degraded: due to weathering, ageing or defects, an increasing substantial amount of water can flow through the waste container and leach the waste (relevant only to the vault facility); and
- totally degraded or no waste container: the waste form is not a limiting factor for water flow and for the leaching of activity (relevant to the trench and vault facilities).

The second component is the engineered features of the disposal facility (the cover in particular). Their states affect the water flow rate and the potential for intrusion:

- the unaltered state ensures a low flow rate;
- the partly-degraded state tends to increase the flow rate, usually with time; and
- the non-existing/disappeared state means the absence of such barriers.

The geosphere (saturated and unsaturated zones) and biosphere are considered as broadly pre-determined and time invariant (i.e. their state remains constant). Consequently, they are not considered to be components for which time varying states need to be attributed.

The third component is human behaviour relating to the institutional control of the site. Its three main states are:

- the existence of an institutional control period preventing any intrusion on the site and ensuring the disposal maintenance;
- a limited possibility of access on but without intrusion in the system due to partial control (e.g. limited surveillance and environmental monitoring) preventing residence and heavy constructions but not casual intrusions; and
- the access without restriction if the site is released into the public domain after the institutional control period.

5.4.2. Identification of scenarios

Having defined the main assessment components with their different states, it is possible to combine them, so that to obtain Fig. 20.

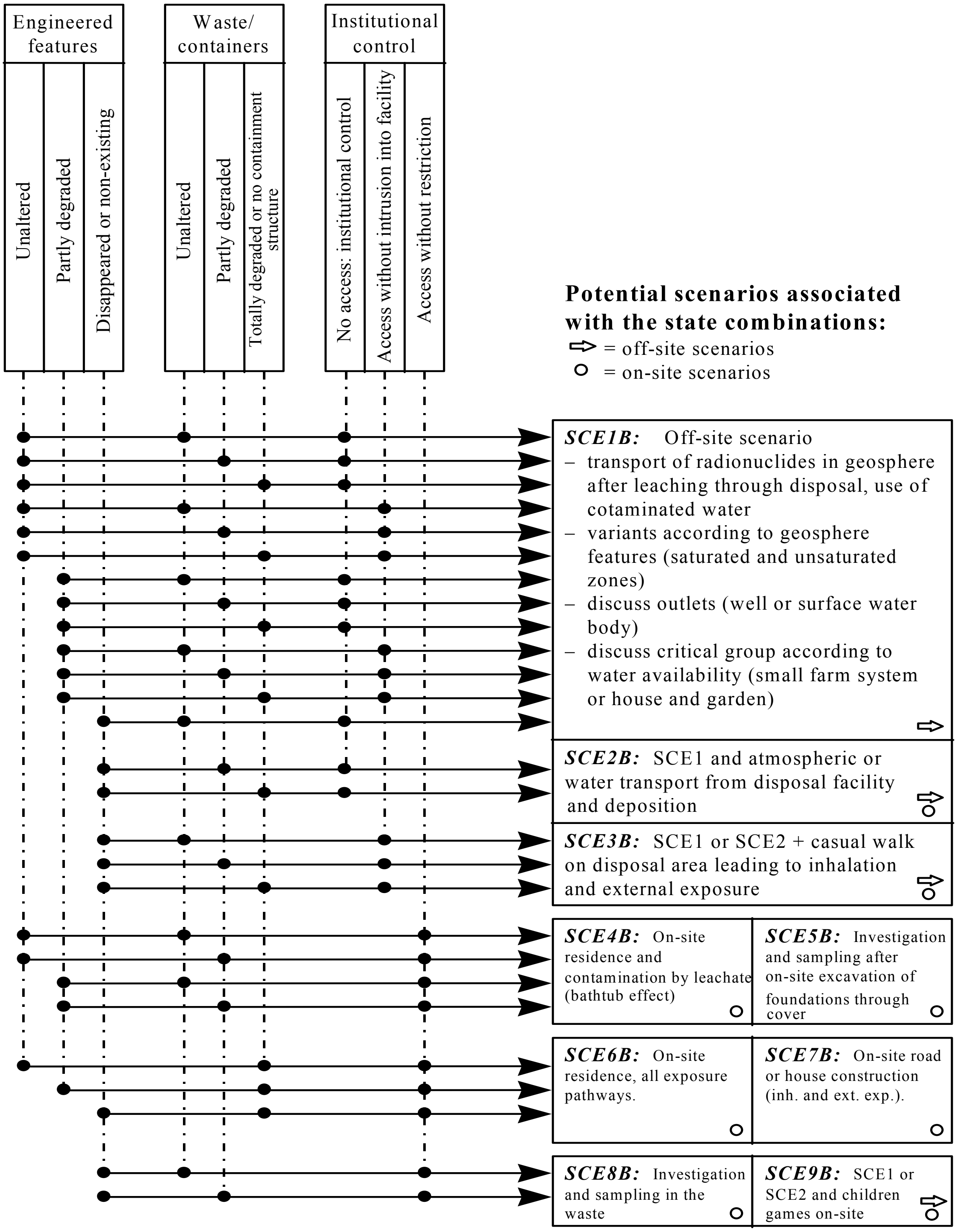

FIG. 20. Generation of a set of scenarios (SCE) for the post-closure period according to various states of the disposal and human behaviour components.

Scenario SCE1B corresponds to the use of contaminated water in the biosphere compartment at the interface with the geosphere, after migration of the radionuclides through the geosphere. The radionuclide concentration in water at the interface does not only depend on the waste and cover performances but also on the geosphere characteristics. For example, the existence or not of an unsaturated zone below the disposal and the hydrogeological properties of the geosphere are important features to be taken into account during the modelling phase. The interface between the geosphere and the biosphere can be either a well intercepting the radioactive plume in the geosphere downstream of the disposal facility, or a surface water body. Whereas the surface water body is generally considered on a site specific basis, the well is usually arbitrarily located in an off-site location where the concentration is the highest (e.g. at the downstream site boundary). Nonetheless, it should not be forgotten that there is a need to ensure consistency between the water availability and the nature of the biosphere assumed. Accordingly, the biosphere can be composed of a small farm system when water is not limiting, or of a kitchen garden, when water is limiting.

Scenario SCE2B differs from SCE1B because of the fact that the cover has disappeared — or was not at all present — and that the waste structures are at least partly degraded, enabling the wind erosion of the disposal and the subsequent atmospheric transfer and deposition of radioactive particles in the critical group location. Depending on the site features (terrain morphology), water erosion and the transport of radionuclides by the water flow can also be processes leading to the contamination of an off-site biosphere system.

In scenario SCE3B, it is considered at the same time that there is no cover (any more) and casual access onto the site is possible. Under such conditions, casual internal exposure or external exposure can occur during these short time intrusions on the disposal facility.

In scenarios SCE4B and SCE5B, the existence of a cover and the unaltered/partly-degraded nature of the waste structures limit the site exploitation and thus reduce the transfer pathways. It is only considered that boreholes can be drilled into the disposal facility. In the particular case of SCE4B, it is envisaged that the water resulting from an accumulation of leachate (bathtub effect) could contaminate a residence system by overflow. However, once again, it is necessary to emphasize the need for a proper justification of such a scenario (e.g. water availability and time necessary for filling the structure with water before the overflow).

In scenarios SCE6B and SCE7B, the wastes are considered totally degraded and so are in a physical state that could result in multiple exposure pathways if they were to be excavated. Nevertheless, consideration should be given to the status and thickness of the cover, which provides some protection against intrusion, due to its thickness. Moreover, in the case of fully engineered facilities (e.g. waste packages grouted in vaults), it is necessary to consider assumptions like the fact that most of the structures should have collapsed and that people will not use their technology for analysing the system. Thus, the suggestion is to assume that such scenarios could not occur before a period of time in the order of several centuries, for example 500 years, consistent with the time-scale for concrete degradation.

Finally, for scenarios SCE8B and SCE9B, even if the cover is absent, the fact that the waste structures remain unaltered, or only partly degraded, limit the potential exposure to the radioactive materials because the number of relevant pathway is reduced for such a case.

5.4.3. Selection of scenarios

Having identified a set of possible scenarios, it is then necessary to sort them according to their likelihood. Some events are almost certain to occur and should therefore be used to define a normal evolution scenario (sometimes also called the reference or base case). The assumptions used in developing this normal evolution scenario are based on extrapolation of existing conditions into the future and incorporation of changes expected to occur with the passage of time, and do not usually consider major perturbations of the system. Typically, a scenario like SCE1B in Fig. 20, where a small farm system is located downstream the disposal facility is a relevant and high probable type of normal evolution scenario. This use of a farm system is also considered and is a mean to ensure that a comprehensive range of exposure pathways is assessed.

Events that are less likely to occur may introduce significant perturbations to the system and require the development of alternative evolution scenarios. Even if not certain, some of them are usually considered on a deterministic basis as relevant for the safety assessment in general, and the derivation of activity limits in particular [14]. Typically, such scenarios include on-site situations like SCE6B, related to the residence on the disposal facility, and SCE7B related to a road construction across the disposal facility. Moreover, some situations are considered as very unlikely to occur, but leading potentially to important radiological impacts. For example, contact with and sampling of a relatively high concentration “hot-spot” (SCE5B and SCE8B) can produce a non-negligible impact but with a low probability. In such cases, the probability of occurrence could be assessed at the same time as the associated dose received by the exposed member of a critical group.

Generally, leaching scenarios like SCE1B and on-site residence/road construction scenarios like SCE6B or SCE7B, are assumed independent from each other. One difficulty that arises is the apparent discrepancy between the assumptions of initial maximized waste leaching, and the assumptions on minimized source term by considering loss by radioactive decay only. In fact, scenarios such as residence and road construction are often envisaged at the very end of the institutional control period during which the disposal system is supposed to be maintained. If a cover has been properly designed, than the infiltration rate can be assumed reduced and constant during the control period, leading also to limited waste leaching. Moreover, the selection of leaching scenarios is justified on the basis that the radionuclides will migrate through the geosphere. Such migration usually takes longer than the control period duration, except perhaps for a very mobile radionuclide such as ^{3}H.

However, one should be aware of the existence of such discrepancies, all the more since some scenarios can account for mixed situations, partly off-site and partly on-site (e.g. see SCE3B and SCE9B).

In light of the above discussion, it is possible to propose a limited and justified set of scenarios to be taken as a basis for deriving the illustrative activity limits. For this study, the scenarios to be considered are:

- the small farm system using water extracted from a well or a surface water body (the leaching scenario – SCE1B);
- the road construction scenario (SCE7B);
- the on-site residence scenario on totally degraded waste (the on-site residence scenario — SCE6B); and
- the residence scenario incurring the contamination by leachate accumulated in the vault disposal facility in clay under temperate conditions (the bathtubbing scenario – SCE4B).

5.4.4. Model formulation and implementation

Using the approach outlined in Section 4.5, the conceptual models summarized in Figs 21–25 can be generated for the various scenarios identified in Section 5.4.3. In each case, the conceptual model identifies (Table V):

— the contaminant release mechanisms and media (i.e. the mechanism causing the release of radionuclides from the waste, and the media in which the radionuclides are released);

— the contaminant transport media and mechanisms (i.e. the media in which and through which the radionuclides move before reaching humans, and the associated transport processes); and

— the human exposure mechanisms (i.e. the pathways through which humans are exposed to the radionuclides).

The timescales over which contaminants might be released from the disposal facilities into the biosphere can be in excess of 1000 years (for example [19]). Human and environmental changes can be considerable over such timescales. Therefore the conceptual models developed, especially for the biosphere, should only be considered to be illustrative.

Ways by which the radiological impact of the scenarios can be quantitatively assessed at a scoping level are indicated in Appendix II.9 to II.13. A suitable mathematical model has been developed for each scenario, based on the associated conceptual model. Each model describes the source term for the scenario and the resulting dose assessment. More detailed mathematical models might be required for certain assessments, but the current models are considered to be appropriate for the purposes of this illustration.

The data required for the solution of the mathematical models relate to the disposal system (disposal facility, geosphere and biosphere) (Appendix I), human exposure (Appendix III) and radionuclides and elements of concern (Appendix IV). Sources of the data for the disposal system are mostly adapted from [31 and 32]. Data relating to human exposure for the leaching scenario are mostly adapted from [26], those for the other scenarios have been adapted from [26] and other appropriate sources specified in Appendix III. Radionuclide and element data are taken from a number of sources; relevant references are indicated at the end of each table in Appendix IV. Data are provided in Appendix IV for the full range of radionuclides identified in Table III and their daughters.

Calculations of the radiological impact associated with each combination of post-closure scenario and disposal system (i.e. calculation case) are summarized in Table VI. A summary of parameters whose values differ between the different calculation cases for each scenario is given in Table VII.

The computational tools used to implement and solve the mathematical models are described in Appendix V.2.

TABLE V. CONTAMINANT RELEASE MECHANISMS AND MEDIA, TRANSPORT MEDIA AND MECHANISMS, AND HUMAN EXPOSURE MECHANISMS FOR THE POST-CLOSURE SCENARIOS CONSIDERED

	Contaminant Release Mechanisms	Contaminant Release Media	Contaminant Transport Media	Contaminant Transport Mechanisms	Human Exposure Mechanisms
***SCE1B:* Leaching (sandy geosphere)**	Leaching	Leachate	Solute in groundwater Well (irrigation, drinking) Soil Crops Cows Atmosphere (dust)	Advection Dispersion Water abstraction for irrigation and drinking water Foliar interception Root uptake Adsorption Ingestion of water, pasture and soil by cows Leaching Erosion	Ingestion of water, crops, and animal produce Inhalation of dust External irradiation from soil
***SCE1B:* Leaching (clay geosphere)**	Leaching	Leachate	Solute in groundwater River (irrigation, drinking) Soil Crops Animals (cows and fish) Atmosphere (dust)	Advection Dispersion Water abstraction for irrigation and drinking water Foliar interception Root uptake Adsorption Ingestion of water, pasture and soil by cows Leaching Erosion River flow	Ingestion of water, crops, and animal produce Inhalation of dust External irradiation from soil
***SCE4B:* Bathtubbing**	Leaching	Leachate	Overflow leachate Soil Atmosphere (dust) Crops	Overflow of leachate Suspension Root uptake Adsorption	Ingestion of crops Inadvertent ingestion of soil Inhalation of dust External irradiation from soil
***SCE6B:* On-site residence**	Excavation Gas generation	Excavated waste Gas	House Gas Soil Atmosphere (dust) Crops	Gas advection Root uptake Adsorption Suspension	Ingestion of crops Inadvertent ingestion of soil Inhalation of dust and gas External irradiation from soil
***SCE7B:* Road construction**	Excavation	Dust	Atmosphere (dust)	Suspension	Inadvertent ingestion of contaminated material and waste Inhalation of dust External irradiation from contaminated material and waste

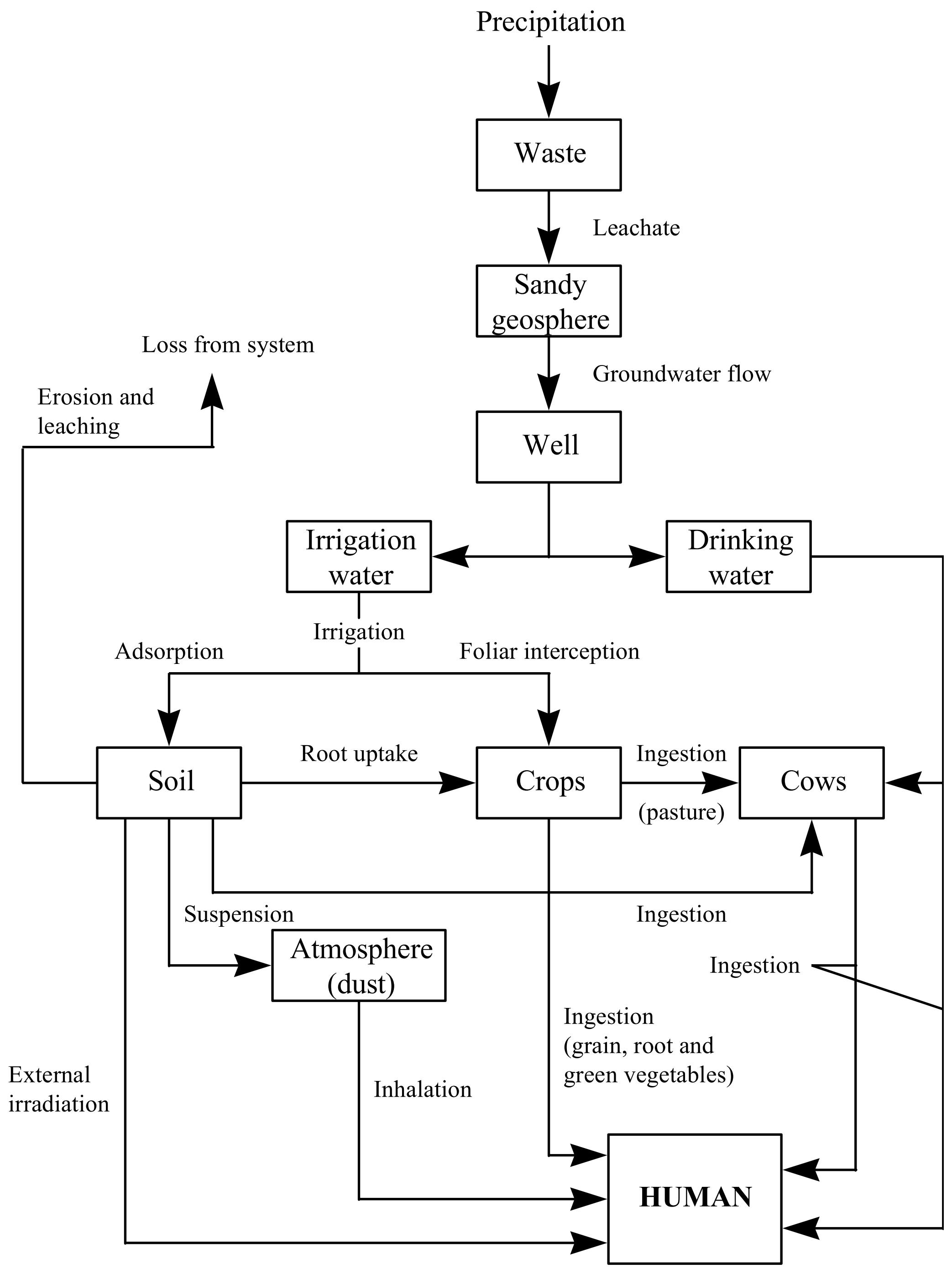

FIG. 21. Simplified representation of the conceptual model for the Post-closure Leaching Scenario (SCE1B) for the sandy geosphere disposal system.

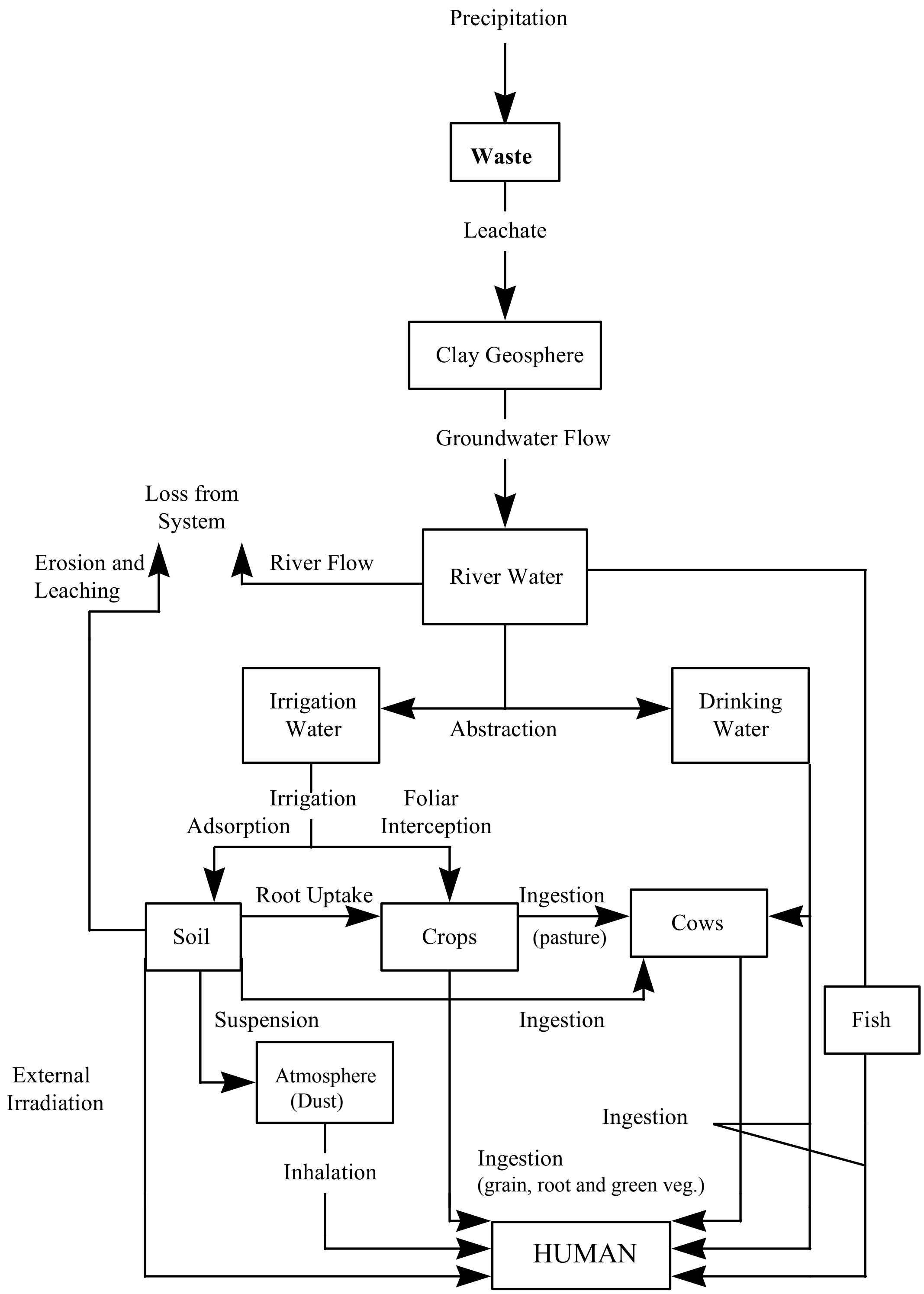

FIG. 22. Simplified representation of the conceptual model for the Post-closure Leaching Scenario (SCE1B) for the clay geosphere disposal system.

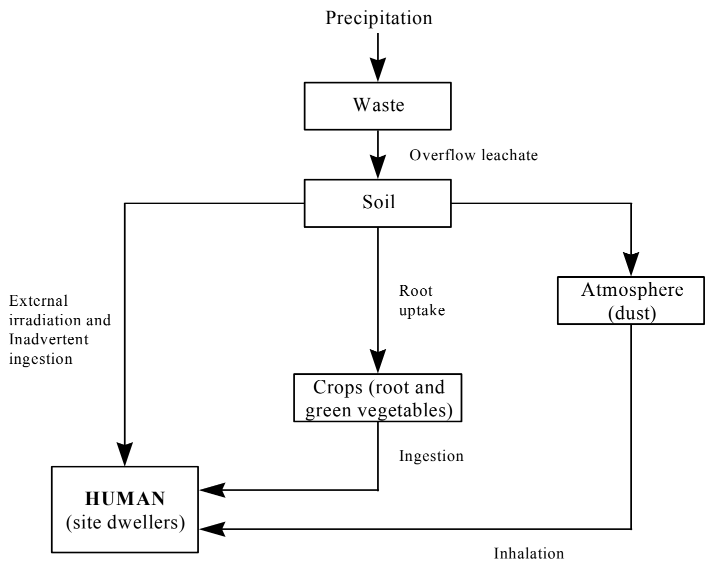

FIG. 23. Simplified representation of the conceptual model for the Post-closure Bathtubbing Scenario (SCE4B).

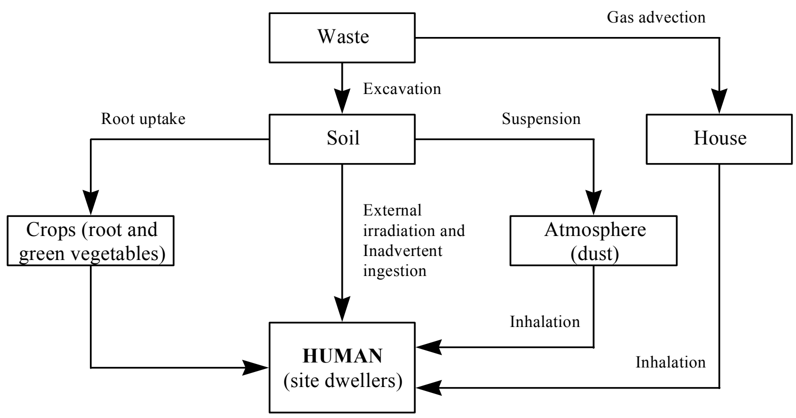

FIG. 24. Simplified representation of the conceptual model for the Post-closure On-site Residence Scenario (SCE6B).

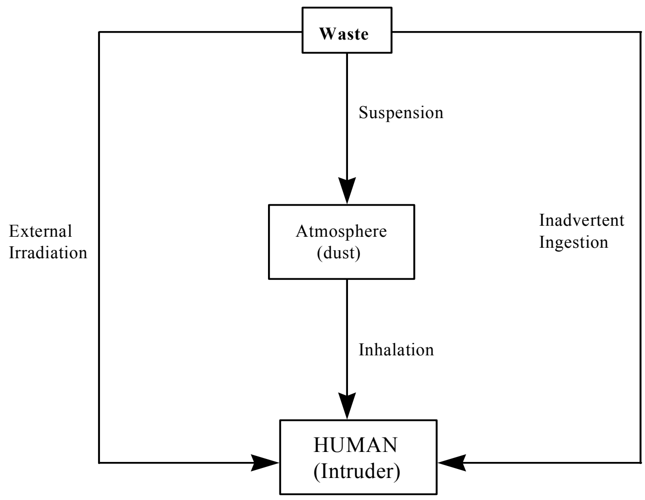

FIG. 25. Simplified representation of the conceptual model for the Post-closure Road Construction Scenario (SCE7B).

TABLE VI. SUMMARY OF CALCULATION CASES FOR THE POST-CLOSURE PERIOD

Scenario	Temperate				Arid			
	Sand		Clay		Sand		Clay	
	Trench	Vault	Trench	Vault	Trench	Vault	Trench	Vault
Leaching (SCE1B)	✓	✓	✓	✓	✓	✓	✓	✓
Bathtubbing (SCE4B)	× (1)	× (1)	× (2)	✓	× (1)	× (1)	× (1)	× (1)
On-site Residence (SCE6B) • Exposure to contaminated soil	✓	× (3)	(4)	× (3)	(4)	× (3)	(4)	× (3)
• Exposure to radon	✓	✓	(4)	(5)	(4)	(5)	(4)	(5)
Road Construction (SCE7B)	✓	✓	(4)	(5)	(4)	(5)	(4)	(5)

Notes:

✓ Calculation required.

× Calculation not required since scenario and disposal system combination is not possible based on scenario description (Section 5.3) and data given in Appendix I, II and III.

(1) Calculation not required since infiltration rate and hydraulic characteristics of disposal system do not result in bathtubbing (see data given in Appendix I, II and III).

(2) Infiltration rate and hydraulic characteristics of disposal system result in bathtubbing within 5 years of closure of a trench (see data given in Appendix I, II and III). Therefore, this case is considered under the operational period calculations (see Section 5.3.3).

(3) Calculation not required since it is assumed that associated intrusion does not penetrate into the vault (see Appendix III.2.2).

(4) The same calculation as for the temperate, sand, trench case.

(5) The same calculation as for the temperate, sand, vault case.

TABLE VII. SUMMARY OF PARAMETERS, WHICH DIFFER BETWEEN CALCULATION CASES FOR THE POST-CLOSURE SCENARIOS

Scenario	Disposal Facility Parameters	Geosphere Parameters	Biosphere Parameters
Leaching (SCE1B)	• Sorption coefficients • Bulk density • Kinetic porosity • Infiltration rate • Waste to backfill ratio • Chemical properties of the vault (e.g. pH)	• Sorption coefficients • Bulk density • Unsaturated zone thickness • Unsaturated zone moisture content • Saturated zone hydraulic gradient • Saturated zone hydraulic conductivity • Saturated zone dispersivities • Distance to geosphere-biosphere interface • Nature of geosphere-biosphere interface	• Sorption coefficients • Infiltration rate • Unsaturated zone thickness • Erosion rate • Irrigation rate • Flow rate in river
Bathtubbing (SCE4B)	• Sorption coefficients • Bulk density • Kinetic porosity • Time of occurrence	–	–
On-site Residence (SCE6B) • Exposure to contaminated soil	• Thickness of cover	–	–
• Exposure to radon	• Thickness of cover • Emanation factor	–	–
Road Construction (SCE7B)	• Time of occurrence • Dilution factor	–	–

5.5. DERIVATION OF ILLUSTRATIVE ACTIVITY LIMITS

In order to derive activity limits in this study two specific disposal systems (a trench and a vault) are considered to illustrate the approach for derivation of activity limits:

(a) The peak dose for each operational and post-closure scenario resulting from a unit activity (concentration or amount) of each radionuclide in the waste disposed of in two disposal systems is calculated using appropriate computational tools (see Appendix V).

(b) The doses are then compared to derive the limiting scenario (i.e. the scenario resulting in the highest dose) from operational and post-closure periods.

(c) The associated activity for each radionuclide is calculated for that limiting scenario using the equations given in Section 4.6.

(d) For each specific system, activity limits are presented in terms of activity concentrations for all scenarios and the most restrictive concentration is identified.

Given the above discussion, the activity limits in Sections 5.5.1 (operational) and 5.5.2 (post-closure) are presented in terms of activity concentration (Bq kg^{-1} of waste) for all scenarios involving a fraction of the total inventory of the disposal system. The activity concentration limits have been calculated using Equation 4.1. In the case of scenarios involving the entire inventory of the disposal system (only the post-closure leaching scenario SCE1B), total activity limits calculated using Equation 4.2 are presented in Section 5.5.3 (comparison of operational and post-closure activity limits). As noted in Section 5.1.2, dose limits of 20 $mSv.y^{-1}$ for workers and 1 $mSv.y^{-1}$ for members of the public are used for calculating the illustrative activity limits given below.

In deriving activity concentrations, the mass of waste that contributes to the radiological impact of each scenario needs to be considered (see Equation 4.1). Different masses contribute to the illustrative operational and post-closure scenarios that have been developed in Sections 5.3 and 5.4. At one extreme there is an individual waste package (for example in the case of the operational drop and crush scenario SCE2A) or the trench tipping face (for example in the case of the operational solid release scenario SCE10A), at the other extreme there is the entire inventory of the disposal facility (for example in the case of the post-closure leaching scenario SCE1B). Table VIII summarized the location and mass of waste in the disposal facility that contributes to the radiological impact of each of the operational and post-closure scenarios assessed for the illustrative cases. It can be seen that a range of different masses are affected. It can be seen from Table VIII that a range of activity limits could be specified for different masses when deriving activity based waste acceptance criteria. However, from a practical viewpoint, it is more appropriate to use only one or two activity limits for indicative masses. One cautious approach would be to apply all the illustrative activity concentration limits obtained to each waste package, even if the affected waste mass is the whole disposal facility. A more realistic approach would be to apply all the illustrative activity concentration limits obtained to each waste package for all scenarios except those that involve the entire inventory of the disposal system. For those that involve the entire inventory of the disposal system, total activity limits would be used instead. If required these total activity limits could be converted into “facility averaged” activity concentration limits by dividing by the planned/estimated total mass of waste to be disposed of in the disposal facility. However, when comparing such activity limits with the activity concentration limits for the scenarios involving only a fraction of the total inventory, it would be important to recognize that the mass of waste to which they apply is much larger. Therefore, individual waste packages with radionuclide concentrations higher than the "facility averaged" activity concentration limits could be acceptable so long as:

(a) the waste package concentrations did not exceed the activity concentration limits for individual waste packages; and

(b) the concentrations averaged out over all waste packages in the disposal facility did not exceed the "facility averaged" activity concentration limits.

5.5.1. Operational period

When considering the illustrative operational activity limits given in Tables 5.9–5.14, it is important to recognize that operational safety can be managed in a more active and direct manner then post-closure safety, by modifying or adopting additional operational procedures in an iterative manner.

The operational period calculations in this publication very much represent a first iteration, prior to the introduction of additional procedures, and so the associated limits must be seen as cautious. This is especially the case for the trench disposal facility where minimal engineering and waste management procedures are in place. Indeed, as noted in Section 5.2.2, the trench disposal facility could be seen as a starting point for an iterative assessment in which additional engineered barriers and operational controls can be progressively introduced until the desired degree of safety is achieved.

TABLE VIII. LOCATION AND MASS OF WASTE IN THE DISPOSAL FACILITY THAT CONTRIBUTES TO THE RADIOLOGICAL IMPACT OF EACH SCENARIO

Scenario	**Location of waste in disposal facility contributing to radiological impact of scenario**	**Mass of waste (kg) contributing to radiological impact of scenario**
Operational		
Gas release		
Vault (SCE1A)	Single disposal unit (1)	9.0E6 kg conditioned waste
Trench (SCE11A)	Single disposal unit (1)	4.5E6 kg unconditioned waste
Drop and crush (SCE2A) (vault only)	Single waste package uncovered	5.0E2 kg conditioned waste
Direct irradiation (SCE8A) (trench only)	Single disposal unit (1)	4.5E6 kg unconditioned waste
Liquid release (SCE9A) (trench only)	Single disposal unit (1)	4.5E6 kg unconditioned waste
Solid release (SCE10A) (trench only)	Tipping face of a single disposal unit	4.5E1 kg unconditioned waste
Fire (SCE12A) (trench only)	Tipping face of a single disposal unit	4.5E4 kg unconditioned waste
Tipping accident (SCE14A) (trench only)	Lorry load of waste	4.0E3 kg unconditioned waste
Bathtubbing (SCE16A) (trench only)	Single disposal unit (1)	4.5E6 kg unconditioned waste
Post-closure		
Leaching (SCE1B)	Entire disposal facility	Vault: 9.0E7 kg conditioned waste Trench: 4.5E7 kg unconditioned waste
Bathtubbing (SCE4B) (vault only)	Single disposal unit (1)	9.0E6 kg conditioned waste
On-site residence (SCE6B)		
Exposure to contaminated soil (trench only)	Part of single disposal unit	1.5E4 kg unconditioned waste
Exposure to radon	Part of single disposal unit	Vault: 6E5 kg conditioned waste Trench: 3E5 kg unconditioned waste
Road construction (SCE7B)	Two disposal units	Vault: 8.2E5–1.1E7 kg conditioned waste Trench: 4.1E5–6.3E6 kg unconditioned waste

Note:

(1) It is recognized that impacts could arise from more than one disposal unit for these scenarios. However, doses are assumed to scale linearly (resulting in the same activity limit) and for the purposes of illustration a single disposal unit is considered.

From Tables IX-XII, it can be seen that, for the trench disposal system under all geosphere and biosphere conditions, it is the liquid release scenario that is the most commonly restricting scenario, sometimes by several orders of magnitude. For this scenario, it is cautiously assumed in the first iteration that the leachate is discharged directly via a drainage system into a river. In light of the significance of this scenario, a collection, monitoring and treatment procedure could be introduced into the design to reduce the impact of liquid releases. Alternatively, or in addition, a more heavily engineered cover could be placed over the trench during and after filling to limit the volume of water infiltrating into the waste.

For the Pu isotopes and ^{241}Am, Tables IX–XII show that the trench fire scenario can be the most restrictive scenario for the trench. For the trench in clay under temperate conditions, the bathtubbing scenario occurs and can, for certain radionuclides (10 in total), become more restrictive than the liquid release scenario (Table X). The direct irradiation scenario can also be limiting for certain radionuclides.

For the vault, only two scenarios are considered; the gas scenario, and the drop and crush scenario. Activity limits that are applicable to all geosphere and biosphere combinations for the vault disposal facility are given in Table XIII. It can seen that for ^{3}H, ^{14}C and ^{226}Ra, it is the gas scenario is the more limiting, whilst for all other radionuclides it is the drop and crush scenario.

Table XIV compares the limiting concentrations for the operational scenarios for the trench and vault in a clay geosphere under temperate conditions. The trench concentrations are more limiting than those for the vault (generally by more than four orders of magnitude) for all radionuclides. This reflects the vault's additional engineering and high standard of waste management and emphasizes that the operational limits for the trench should be seen as first iteration limits that can be increased if appropriate engineering and waste management controls are put in place for the trench.

5.5.2. Post-closure period

The illustrative activity concentration limits ($Bq \cdot kg^{-1}$ of waste) for the post-closure period are presented in Tables XXV–XXVII for the road construction scenario, on-site residence scenario and, if appropriate, the bathtubbing scenario. These are presented separately for the trench and vault disposal facilities. In the case of the trench (Table XV, the limits are applicable to all geosphere and biosphere combinations (see Fig. 8). For the vault, there is a need to distinguish between the limits for the clay geosphere under temperate conditions (Table XVI) and the limits for all other geosphere and biosphere combinations (Table XVII). This is because the bathtubbing scenario only needs to be assessed for the clay geosphere under temperate conditions due to the hydraulic conditions resulting in the potential for bathtubbing.

From Table XV it can be seen that, the limiting scenario for all radionuclides is the on-site residence, soil. As expected, the period of institutional control affects the concentration limits for the shorter lived radionuclides such as ^{3}H, ^{90}Sr and ^{137}Cs, but has no impact on the concentration of longer lived radionuclides such as ^{129}I and ^{238}U.

For the vault system with the clay geosphere and temperate conditions (Table XVI), the limiting on-site scenario is the bathtubbing scenario for the shorter lived radionuclides (for example ^{3}H, ^{90}Sr and ^{137}Cs) and mobile longer lived radionuclide (^{129}I).

TABLE IX. OPERATIONAL SCENARIO CONCENTRATIONS (Bq·kg^{-1} OF WASTE) FOR THE TRENCH IN SAND UNDER TEMPERATE CONDITIONS

Radionuclide	Direct Irradiation (SCE8A)	Liquid Release (SCE9A)	Solid Release (SCE10A)		Gas Release (SCE11A)		Fire Release (SCE12A)		Tipping Accident (SCE14)
	Worker	Public	Worker	Public	Worker	Public	Worker	Public	Worker
^{3}H	1.E+20	**2.E+07**	6.E+12	3.E+12	1.E+14	3.E+12	6.E+10	3.E+08	1.E+14
^{10}Be	1.E+20	**5.E+07**	5.E+10	2.E+10			5.E+08	2.E+09	8.E+11
^{14}C	1.E+20	**1.E+07**	3.E+11	1.E+11	6.E+11	2.E+10	3.E+09	**1.E+07**	4.E+12
^{22}Na	**5.E+06**	**5.E+06**	9.E+11	3.E+09			9.E+09	9.E+08	8.E+10
^{41}Ca	1.E+20	**1.E+07**	9.E+12	1.E+12			9.E+10	1.E+11	3.E+13
^{54}Mn	**1.E+07**	**4.E+07**	1.E+12	8.E+09			1.E+10	3.E+10	2.E+11
^{55}Fe	1.E+20	**2.E+08**	2.E+12	5.E+11			2.E+10	5.E+10	1.E+13
^{59}Ni	1.E+20	**2.E+09**	4.E+12	1.E+12			4.E+10	1.E+11	5.E+13
^{63}Ni	1.E+20	**6.E+08**	1.E+12	5.E+11			1.E+10	5.E+10	2.E+13
^{60}Co	**4.E+06**	**5.E+06**	5.E+10	2.E+09			5.E+08	2.E+09	7.E+10
^{65}Zn	**2.E+07**	**3.E+07**	7.E+11	1.E+10			7.E+09	7.E+07	3.E+11
^{90}Sr	1.E+20	**1.E+05**	1.E+10	4.E+09			1.E+08	4.E+08	1.E+11
^{93}Zr	5.E+11	**1.E+08**	7.E+10	3.E+10			7.E+08	3.E+09	1.E+12
^{94}Nb	**8.E+06**	**9.E+06**	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{99}Tc	7.E+13	**4.E+05**	1.E+11	6.E+10			1.E+09	6.E+09	2.E+12
^{106}Ru	8.E+07	**4.E+06**	3.E+10	8.E+09			3.E+08	1.E+07	2.E+11
^{110m}Ag	**2.E+06**	**2.E+07**	1.E+11	3.E+09			1.E+09	5.E+08	6.E+10
^{121m}Sn	3.E+10	**5.E+07**	4.E+11	1.E+11			4.E+09	1.E+10	4.E+12
^{125}Sb	**1.E+07**	**1.E+07**	1.E+11	1.E+10			1.E+09	6.E+07	4.E+11
^{126}Sn	**2.E+06**	**4.E+06**	6.E+10	2.E+09			6.E+08	2.E+09	9.E+10
^{129}I	5.E+09	**4.E+03**	5.E+10	3.E+09			5.E+08	3.E+05	6.E+10
^{134}Cs	**4.E+06**	**4.E+06**	8.E+10	3.E+09			8.E+08	1.E+07	9.E+10
^{137}Cs	2.E+07	**5.E+06**	4.E+10	5.E+09			4.E+08	1.E+07	2.E+11
^{144}Ce	1.E+08	**5.E+07**	3.E+10	1.E+10			3.E+08	1.E+09	4.E+11
^{147}Pm	9.E+12	**3.E+08**	3.E+11	2.E+11			3.E+09	2.E+10	5.E+12
^{151}Sm	7.E+13	**6.E+08**	4.E+11	2.E+11			4.E+09	2.E+10	7.E+12
^{152}Eu	**4.E+06**	4.E+07	4.E+10	4.E+09			4.E+08	2.E+09	8.E+06
^{154}Eu	**3.E+06**	**3.E+07**	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{204}Tl	4.E+10	**6.E+07**	4.E+12	2.E+11			4.E+10	2.E+10	6.E+12
^{210}Pb	3.E+10	3.E+04	3.E+08	1.E+08			3.E+06	**2.E+04**	4.E+09
^{226}Ra	9.E+05	**2.E+04**	2.E+08	8.E+07	2.E+09	4.E+07	2.E+06	9.E+06	3.E+09
^{227}Ac	6.E+06	9.E+04	3.E+06	2.E+06			**3.E+04**	2.E+05	6.E+07
^{228}Ra	1.E+06	**2.E+05**	1.E+08	5.E+07			1.E+06	5.E+06	2.E+09
^{232}Th	8.E+05	**2.E+05**	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08
^{234}U	2.E+11	**2.E+05**	2.E+08	9.E+07			2.E+06	9.E+06	3.E+09
^{235}U	4.E+07	**2.E+05**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{238}U	8.E+07	**2.E+05**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{237}Np	4.E+08	**1.E+04**	3.E+07	2.E+07			3.E+05	2.E+06	7.E+08
^{238}Pu	5.E+11	6.E+05	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08
^{239}Pu	2.E+11	5.E+05	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08
^{240}Pu	6.E+11	5.E+05	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08
^{241}Pu	3.E+10	3.E+07	7.E+08	4.E+08			**7.E+06**	4.E+07	1.E+10
^{241}Am	1.E+09	2.E+06	2.E+07	9.E+06			**2.E+05**	9.E+05	3.E+08

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for the public and 20 mSv.y^{-1} for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE X. OPERATIONAL SCENARIO CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE TRENCH IN SAND UNDER ARID CONDITIONS

Radionuclide	**Direct Irradiation (SCE8A)**	**Liquid Release (SCE9A)**	**Solid Release (SCE10A)**		**Gas Release (SCE11A)**		**Fire Release (SCE12A)**		**Tipping Accident (SCE14)**
	Worker	**Public**	**Worker**	**Public**	**Worker**	**Public**	**Worker**	**Public**	**Worker**
^{3}H	1.E+20	**3.E+06**	6.E+12	3.E+12	1.E+14	3.E+12	6.E+10	3.E+08	1.E+14
^{10}Be	1.E+20	**1.E+07**	5.E+10	2.E+10			5.E+08	2.E+09	8.E+11
^{14}C	1.E+20	**4.E+06**	3.E+11	1.E+11	6.E+11	2.E+10	3.E+09	1.E+07	4.E+12
^{22}Na	5.E+06	**1.E+06**	9.E+11	3.E+09			9.E+09	9.E+08	8.E+10
^{41}Ca	1.E+20	**3.E+06**	9.E+12	1.E+12			9.E+10	1.E+11	3.E+13
^{54}Mn	1.E+07	**8.E+06**	1.E+12	8.E+09			1.E+10	3.E+10	2.E+11
^{55}Fe	1.E+20	**4.E+07**	2.E+12	5.E+11			2.E+10	5.E+10	1.E+13
^{59}Ni	1.E+20	**3.E+08**	4.E+12	1.E+12			4.E+10	1.E+11	5.E+13
^{63}Ni	1.E+20	**1.E+08**	1.E+12	5.E+11			1.E+10	5.E+10	2.E+13
^{60}Co	4.E+06	**1.E+06**	5.E+10	2.E+09			5.E+08	2.E+09	7.E+10
^{65}Zn	2.E+07	**7.E+06**	7.E+11	1.E+10			7.E+09	7.E+07	3.E+11
^{90}Sr	1.E+20	**2.E+04**	1.E+10	4.E+09			1.E+08	4.E+08	1.E+11
^{93}Zr	5.E+11	**2.E+07**	7.E+10	3.E+10			7.E+08	3.E+09	1.E+12
^{94}Nb	8.E+06	**2.E+06**	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{99}Tc	7.E+13	**8.E+04**	1.E+11	6.E+10			1.E+09	6.E+09	2.E+12
^{106}Ru	8.E+07	**8.E+05**	3.E+10	8.E+09			3.E+08	1.E+07	2.E+11
^{110m}Ag	**2.E+06**	4.E+06	1.E+11	3.E+09			1.E+09	5.E+08	6.E+10
^{121m}Sn	3.E+10	**1.E+07**	4.E+11	1.E+11			4.E+09	1.E+10	4.E+12
^{125}Sb	1.E+07	**3.E+06**	1.E+11	1.E+10			1.E+09	6.E+07	4.E+11
^{126}Sn	2.E+06	**7.E+05**	6.E+10	2.E+09			6.E+08	2.E+09	9.E+10
^{129}I	5.E+09	**9.E+02**	5.E+10	3.E+09			5.E+08	3.E+05	6.E+10
^{134}Cs	4.E+06	**1.E+06**	8.E+10	3.E+09			8.E+08	1.E+07	9.E+10
^{137}Cs	2.E+07	**1.E+06**	4.E+10	5.E+09			4.E+08	1.E+07	2.E+11
^{144}Ce	1.E+08	**1.E+07**	3.E+10	1.E+10			3.E+08	1.E+09	4.E+11
^{147}Pm	9.E+12	**6.E+07**	3.E+11	2.E+11			3.E+09	2.E+10	5.E+12
^{151}Sm	7.E+13	**1.E+08**	4.E+11	2.E+11			4.E+09	2.E+10	7.E+12
^{152}Eu	**4.E+06**	9.E+06	4.E+10	4.E+09			4.E+08	2.E+09	8.E+06
^{154}Eu	**3.E+06**	7.E+06	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{204}Tl	4.E+10	**1.E+07**	4.E+12	2.E+11			4.E+10	2.E+10	6.E+12
^{210}Pb	3.E+10	**5.E+03**	3.E+08	1.E+08			3.E+06	2.E+04	4.E+09
^{226}Ra	9.E+05	**5.E+03**	2.E+08	8.E+07	2.E+09	4.E+07	2.E+06	9.E+06	3.E+09
^{227}Ac	6.E+06	**2.E+04**	3.E+06	2.E+06			3.E+04	2.E+05	6.E+07
^{228}Ra	1.E+06	**4.E+04**	1.E+08	5.E+07			1.E+06	5.E+06	2.E+09
^{232}Th	8.E+05	**3.E+04**	2.E+07	8.E+06			2.E+05	8.E+05	3.E+08
^{234}U	2.E+11	**4.E+04**	2.E+08	9.E+07			2.E+06	9.E+06	3.E+09
^{235}U	4.E+07	**4.E+04**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{238}U	8.E+07	**4.E+04**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{237}Np	4.E+08	**2.E+03**	3.E+07	2.E+07			3.E+05	2.E+06	7.E+08
^{238}Pu	5.E+11	**1.E+05**	2.E+07	8.E+06			2.E+05	8.E+05	3.E+08
^{239}Pu	2.E+11	**1.E+05**	1.E+07	7.E+06			1.E+05	7.E+05	3.E+08
^{240}Pu	6.E+11	**1.E+05**	1.E+07	7.E+06			1.E+05	7.E+05	3.E+08
^{241}Pu	3.E+10	**6.E+06**	7.E+08	4.E+08			7.E+06	4.E+07	1.E+10
^{241}Am	1.E+09	5.E+05	2.E+07	9.E+06			**2.E+05**	9.E+05	3.E+08

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XI. OPERATIONAL SCENARIO CONCENTRATIONS (Bq·kg^{-1} OF WASTE FOR THE TRENCH IN CLAY UNDER TEMPERATE CONDITIONS

Radionuclide	Direct Irradiation (SCE8A)	Liquid Release (SCE9A)	Solid Release (SCE10)		Gas Release (SCE11)		Fire Release (SCE12)		Tipping Accident (SCE14)	Bathtubbing (SCE16)
	Worker	Public	Worker	Public	Worker	Public	Worker	Public	Worker	Public
^{3}H	1.E+20	2.E+07	6.E+12	3.E+12	1.E+14	3.E+12	6.E+10	3.E+08	1.E+14	**4.E+05**
^{10}Be	1.E+20	**3.E+08**	5.E+10	2.E+10			5.E+08	2.E+09	8.E+11	2.E+10
^{14}C	1.E+20	**1.E+07**	3.E+11	1.E+11	6.E+11	2.E+10	3.E+09	**1.E+07**	4.E+12	5.E+07
^{22}Na	5.E+06	3.E+07	9.E+11	3.E+09			9.E+09	9.E+08	8.E+10	**4.E+06**
^{41}Ca	1.E+20	5.E+08	9.E+12	1.E+12			9.E+10	1.E+11	3.E+13	**2.E+08**
^{54}Mn	**1.E+07**	1.E+08	1.E+12	8.E+09			1.E+10	3.E+10	2.E+11	2.E+08
^{55}Fe	1.E+20	**2.E+08**	2.E+12	5.E+11			2.E+10	5.E+10	1.E+13	1.E+11
^{59}Ni	1.E+20	**3.E+09**	4.E+12	1.E+12			4.E+10	1.E+11	5.E+13	1.E+10
^{63}Ni	1.E+20	**1.E+09**	1.E+12	5.E+11			1.E+10	5.E+10	2.E+13	5.E+09
^{60}Co	4.E+06	4.E+07	5.E+10	2.E+09			5.E+08	2.E+09	7.E+10	**3.E+06**
^{65}Zn	**2.E+07**	4.E+08	7.E+11	1.E+10			7.E+09	7.E+07	3.E+11	9.E+09
^{90}Sr	1.E+20	**9.E+05**	1.E+10	4.E+09			1.E+08	4.E+08	1.E+11	1.E+06
^{93}Zr	5.E+11	**5.E+08**	7.E+10	3.E+10			7.E+08	3.E+09	1.E+12	1.E+10
^{94}Nb	8.E+06	4.E+07	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11	**4.E+06**
^{99}Tc	7.E+13	8.E+05	1.E+11	6.E+10			1.E+09	6.E+09	2.E+12	**9.E+03**
^{106}Ru	8.E+07	3.E+07	3.E+10	8.E+09			3.E+08	**1.E+07**	2.E+11	7.E+08
^{110m}Ag	**2.E+06**	4.E+07	1.E+11	3.E+09			1.E+09	5.E+08	6.E+10	2.E+08
^{121m}Sn	3.E+10	**3.E+08**	4.E+11	1.E+11			4.E+09	1.E+10	4.E+12	**3.E+08**
^{125}Sb	**1.E+07**	6.E+07	1.E+11	1.E+10			1.E+09	6.E+07	4.E+11	2.E+07
^{126}Sn	**2.E+06**	2.E+07	6.E+10	2.E+09			6.E+08	2.E+09	9.E+10	**2.E+06**
^{129}I	5.E+09	4.E+05	5.E+10	3.E+09			5.E+08	**3.E+05**	6.E+10	5.E+05
^{134}Cs	**4.E+06**	3.E+07	8.E+10	3.E+09			8.E+08	1.E+07	9.E+10	6.E+07
^{137}Cs	2.E+07	3.E+07	4.E+10	5.E+09			4.E+08	1.E+07	2.E+11	2.E+07
^{144}Ce	**1.E+08**	2.E+09	3.E+10	1.E+10			3.E+08	1.E+09	4.E+11	5.E+11
^{147}Pm	9.E+12	**2.E+09**	3.E+11	2.E+11			3.E+09	2.E+10	5.E+12	2.E+11
^{151}Sm	7.E+13	**3.E+09**	4.E+11	2.E+11			4.E+09	2.E+10	7.E+12	2.E+11
^{152}Eu	**4.E+06**	2.E+08	4.E+10	4.E+09			4.E+08	2.E+09	8.E+06	9.E+06
^{154}Eu	**3.E+06**	2.E+08	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11	1.E+07
^{204}Tl	4.E+10	**1.E+08**	4.E+12	2.E+11			4.E+10	2.E+10	6.E+12	4.E+09
^{210}Pb	3.E+10	1.E+05	3.E+08	1.E+08			3.E+06	**2.E+04**	4.E+09	3.E+06
^{226}Ra	9.E+05	**1.E+05**	2.E+08	8.E+07	2.E+09	4.E+07	2.E+06	9.E+06	3.E+09	7.E+06
^{227}Ac	6.E+06	5.E+05	3.E+06	2.E+06			**3.E+04**	2.E+05	6.E+07	3.E+07
^{228}Ra	**1.E+06**	3.E+06	1.E+08	5.E+07			**1.E+06**	5.E+06	2.E+09	1.E+07
^{232}Th	8.E+05	2.E+06	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08	1.E+07
^{234}U	2.E+11	7.E+06	2.E+08	9.E+07			**2.E+06**	9.E+06	3.E+09	7.E+08
^{235}U	4.E+07	7.E+06	2.E+08	1.E+08			**2.E+06**	1.E+07	4.E+09	1.E+08
^{238}U	8.E+07	8.E+06	2.E+08	1.E+08			**2.E+06**	1.E+07	4.E+09	2.E+08
^{237}Np	4.E+08	**1.E+05**	3.E+07	2.E+07			3.E+05	2.E+06	7.E+08	2.E+06
^{238}Pu	5.E+11	5.E+06	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08	4.E+08
^{239}Pu	2.E+11	5.E+06	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08	4.E+08
^{240}Pu	6.E+11	5.E+06	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08	4.E+08
^{241}Pu	3.E+10	3.E+08	7.E+08	4.E+08			**7.E+06**	4.E+07	1.E+10	2.E+10
^{241}Am	1.E+09	1.E+07	2.E+07	9.E+06			**2.E+05**	9.E+05	3.E+08	8.E+08

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for the public and 20 mSv.y^{-1} for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XII. OPERATIONAL SCENARIO CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE TRENCH IN CLAY UNDER ARID CONDITIONS

Radionuclide	Direct Irradiation (SCE8A)	Liquid Release (SCE9A)	Solid Release (SCE10A)		Gas Release (SCE11A)		Fire Release (SCE12A)		Tipping Accident (SCE14)
	Worker	**Public**	**Worker**	**Public**	**Worker**	**Public**	**Worker**	**Public**	**Worker**
^{3}H	1.E+20	**3.E+06**	6.E+12	3.E+12	1.E+14	3.E+12	6.E+10	3.E+08	1.E+14
^{10}Be	1.E+20	**6.E+07**	5.E+10	2.E+10			5.E+08	2.E+09	8.E+11
^{14}C	1.E+20	**4.E+06**	3.E+11	1.E+11	6.E+11	2.E+10	3.E+09	1.E+07	4.E+12
^{22}Na	**5.E+06**	**5.E+06**	9.E+11	3.E+09			9.E+09	9.E+08	8.E+10
^{41}Ca	1.E+20	**1.E+08**	9.E+12	1.E+12			9.E+10	1.E+11	3.E+13
^{54}Mn	**1.E+07**	3.E+07	1.E+12	8.E+09			1.E+10	3.E+10	2.E+11
^{55}Fe	1.E+20	**3.E+07**	2.E+12	5.E+11			2.E+10	5.E+10	1.E+13
^{59}Ni	1.E+20	**5.E+08**	4.E+12	1.E+12			4.E+10	1.E+11	5.E+13
^{63}Ni	1.E+20	**2.E+08**	1.E+12	5.E+11			1.E+10	5.E+10	2.E+13
^{60}Co	**4.E+06**	9.E+06	5.E+10	2.E+09			5.E+08	2.E+09	7.E+10
^{65}Zn	**2.E+07**	9.E+07	7.E+11	1.E+10			7.E+09	7.E+07	3.E+11
^{90}Sr	1.E+20	**2.E+05**	1.E+10	4.E+09			1.E+08	4.E+08	1.E+11
^{93}Zr	5.E+11	**1.E+08**	7.E+10	3.E+10			7.E+08	3.E+09	1.E+12
^{94}Nb	**8.E+06**	**8.E+06**	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{99}Tc	7.E+13	**2.E+05**	1.E+11	6.E+10			1.E+09	6.E+09	2.E+12
^{106}Ru	8.E+07	**6.E+06**	3.E+10	8.E+09			3.E+08	1.E+07	2.E+11
^{110m}Ag	**2.E+06**	9.E+06	1.E+11	3.E+09			1.E+09	5.E+08	6.E+10
^{121m}Sn	3.E+10	**6.E+07**	4.E+11	1.E+11			4.E+09	1.E+10	4.E+12
^{125}Sb	**1.E+07**	**1.E+07**	1.E+11	1.E+10			1.E+09	6.E+07	4.E+11
^{126}Sn	**2.E+06**	3.E+06	6.E+10	2.E+09			6.E+08	2.E+09	9.E+10
^{129}I	5.E+09	**8.E+04**	5.E+10	3.E+09			5.E+08	3.E+05	6.E+10
^{134}Cs	**4.E+06**	6.E+06	8.E+10	3.E+09			8.E+08	1.E+07	9.E+10
^{137}Cs	**2.E+07**	7.E+06	4.E+10	5.E+09			4.E+08	1.E+07	2.E+11
^{144}Ce	**1.E+08**	5.E+08	3.E+10	1.E+10			3.E+08	1.E+09	4.E+11
^{147}Pm	9.E+12	**3.E+08**	3.E+11	2.E+11			3.E+09	2.E+10	5.E+12
^{151}Sm	7.E+13	**7.E+08**	4.E+11	2.E+11			4.E+09	2.E+10	7.E+12
^{152}Eu	**4.E+06**	5.E+07	4.E+10	4.E+09			4.E+08	2.E+09	8.E+06
^{154}Eu	**3.E+06**	4.E+07	3.E+10	3.E+09			3.E+08	2.E+09	1.E+11
^{204}Tl	4.E+10	**3.E+07**	4.E+12	2.E+11			4.E+10	2.E+10	6.E+12
^{210}Pb	3.E+10	3.E+04	3.E+08	1.E+08			3.E+06	**2.E+04**	4.E+09
^{226}Ra	9.E+05	**3.E+04**	2.E+08	8.E+07	2.E+09	4.E+07	2.E+06	9.E+06	3.E+09
^{227}Ac	6.E+06	1.E+05	3.E+06	2.E+06			**3.E+04**	2.E+05	6.E+07
^{228}Ra	1.E+06	**7.E+05**	1.E+08	5.E+07			1.E+06	5.E+06	2.E+09
^{232}Th	8.E+05	3.E+05	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08
^{234}U	2.E+11	**2.E+06**	2.E+08	9.E+07			2.E+06	9.E+06	3.E+09
^{235}U	4.E+07	**2.E+06**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{238}U	8.E+07	**2.E+06**	2.E+08	1.E+08			2.E+06	1.E+07	4.E+09
^{237}Np	4.E+08	**3.E+04**	3.E+07	2.E+07			3.E+05	2.E+06	7.E+08
^{238}Pu	5.E+11	1.E+06	2.E+07	8.E+06			**2.E+05**	8.E+05	3.E+08
^{239}Pu	2.E+11	1.E+06	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08
^{240}Pu	6.E+11	1.E+06	1.E+07	7.E+06			**1.E+05**	7.E+05	3.E+08
^{241}Pu	3.E+10	5.E+07	7.E+08	4.E+08			**7.E+06**	4.E+07	1.E+10
^{241}Am	1.E+09	2.E+06	2.E+07	9.E+06			**2.E+05**	9.E+05	3.E+08

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XIII. OPERATIONAL SCENARIO CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE VAULT IN CLAY UNDER TEMPERATE CONDITIONS

Radionuclide	Gas Release (SCE1A)		Drop and Crush (SCE2A)	
	Worker	**Public**	**Crane operator**	**Worker**
^{3}H	6.E+13	**1.E+12**	1.E+20	1.E+20
^{10}Be			**1.E+20**	**1.E+20**
^{14}C	3.E+11	**8.E+09**	1.E+20	1.E+20
^{22}Na			4.E+11	**4.E+11**
^{41}Ca			2.E+15	**3.E+14**
^{54}Mn			1.E+12	**1.E+12**
^{55}Fe			8.E+14	**1.E+12**
^{59}Ni			7.E+14	**9.E+13**
^{63}Ni			**1.E+20**	**1.E+20**
^{60}Co			4.E+11	**4.E+11**
^{65}Zn			2.E+12	**2.E+12**
^{90}Sr			**1.E+20**	**1.E+20**
^{93}Zr			**1.E+20**	**1.E+20**
^{94}Nb			6.E+11	**5.E+11**
^{99}Tc			**1.E+20**	**1.E+20**
^{106}Ru			5.E+12	**4.E+12**
^{110m}Ag			4.E+11	**3.E+11**
^{121m}Sn			4.E+14	**1.E+14**
^{125}Sb			2.E+12	**2.E+12**
^{126}Sn			5.E+11	**5.E+11**
^{129}I			2.E+13	**2.E+13**
^{134}Cs			6.E+11	**6.E+11**
^{137}Cs			2.E+12	**2.E+12**
^{144}Ce			**8.E+10**	2.E+13
^{147}Pm			2.E+17	**2.E+17**
^{151}Sm			9.E+16	**2.E+16**
^{152}Eu			8.E+11	**8.E+11**
^{154}Eu			7.E+11	**7.E+11**
^{204}Tl			6.E+14	**6.E+14**
^{210}Pb			2.E+14	**9.E+13**
^{226}Ra	1.E+07	**4.E+05**	6.E+11	5.E+11
^{227}Ac			2.E+12	**2.E+12**
^{228}Ra			**1.E+20**	**1.E+20**
^{232}Th			2.E+15	**3.E+14**
^{234}U			2.E+15	**3.E+14**
^{235}U			5.E+12	**4.E+12**
^{238}U			3.E+13	**3.E+13**
^{237}Np			4.E+12	**3.E+12**
^{238}Pu			2.E+15	**3.E+14**
^{239}Pu			4.E+15	**7.E+14**
^{240}Pu			2.E+15	**3.E+14**
^{241}Pu			**1.E+20**	**1.E+20**
^{241}Am			3.E+13	**2.E+13**

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XIV. COMPARISON OF LIMITING CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR OPERATIONAL SCENARIOS FOR THE TRENCH AND VAULT IN CLAY UNDER TEMPERATE CONDITIONS

Radionuclide	Trench Limit	Trench Limiting Scenario	Vault Limit	Vault Limiting Scenario
^{3}H	**4.E+05**	Bathtubbing (public)	1.E+12	Gas release (public)
^{10}Be	**3.E+08**	Liquid release (public)	1.E+20	Drop and crush
^{14}C	**1.E+07**	Fire release (public)	8.E+09	Gas release (public)
^{22}Na	**4.E+06**	Bathtubbing (public)	4.E+11	Drop and crush (worker)
^{41}Ca	**2.E+08**	Bathtubbing (public)	3.E+14	Drop and crush (worker)
^{54}Mn	**1.E+07**	Liquid release (public)	1.E+12	Drop and crush (worker)
^{55}Fe	**2.E+08**	Liquid release (public)	1.E+12	Drop and crush (worker)
^{59}Ni	**3.E+09**	Liquid release (public)	9.E+13	Drop and crush (worker)
^{63}Ni	**1.E+09**	Liquid release (public)	1.E+20	Drop and crush
^{60}Co	**3.E+06**	Bathtubbing (public)	4.E+11	Drop and crush (worker)
^{65}Zn	**2.E+07**	Direct irradiation (worker)	2.E+12	Drop and crush (worker)
^{90}Sr	**9.E+05**	Liquid release (public)	1.E+20	Drop and crush
^{93}Zr	**5.E+08**	Liquid release (public)	1.E+20	Drop and crush
^{94}Nb	**4.E+06**	Bathtubbing (public)	5.E+11	Drop and crush (worker)
^{99}Tc	**9.E+03**	Bathtubbing (public)	1.E+20	Drop and crush
^{106}Ru	**1.E+07**	Fire release (public)	4.E+12	Drop and crush (worker)
^{110m}Ag	**2.E+06**	Direct irradiation (worker)	3.E+11	Drop and crush (worker)
^{121m}Sn	**3.E+08**	Bathtubbing (public)	1.E+14	Drop and crush (worker)
^{125}Sb	**2.E+07**	Bathtubbing (public)	2.E+12	Drop and crush (worker)
^{126}Sn	**2.E+06**	Bathtubbing (public)	5.E+11	Drop and crush (worker)
^{129}I	**3.E+05**	Fire release (public)	2.E+13	Drop and crush (worker)
^{134}Cs	**4.E+06**	Direct irradiation (worker)	6.E+11	Drop and crush (worker)
^{137}Cs	**1.E+07**	Fire release (public)	2.E+12	Drop and crush (worker)
^{144}Ce	**1.E+08**	Direct irradiation (worker)	8.E+10	Drop and crush (crane operator)
^{147}Pm	**2.E+09**	Liquid release (public)	2.E+17	Drop and crush (worker)
^{151}Sm	**3.E+09**	Liquid release (public)	2.E+16	Drop and crush (worker)
^{152}Eu	**4.E+06**	Direct irradiation (worker)	8.E+11	Drop and crush (worker)
^{154}Eu	**3.E+06**	Direct irradiation (worker)	7.E+11	Drop and crush (worker)
^{204}Tl	**1.E+08**	Liquid release (public)	6.E+14	Drop and crush (worker)
^{210}Pb	**2.E+04**	Fire release (public)	9.E+13	Drop and crush (worker)
^{226}Ra	**1.E+05**	Liquid release (public)	4.E+05	Gas release (public)
^{227}Ac	**3.E+04**	Direct irradiation and fire release (worker)	2.E+12	Drop and crush (worker)
^{228}Ra	**1.E+06**	Fire release (worker)	1.E+20	Drop and crush
^{232}Th	**2.E+05**	Fire release (worker)	3.E+14	Drop and crush (worker)
^{234}U	**2.E+06**	Fire release (worker)	3.E+14	Drop and crush (worker)
^{235}U	**2.E+06**	Fire release (worker)	4.E+12	Drop and crush (worker)
^{238}U	**2.E+06**	Fire release (worker)	3.E+13	Drop and crush (worker)
^{237}Np	**1.E+05**	Liquid release (public)	3.E+12	Drop and crush (worker)
^{238}Pu	**2.E+05**	Fire release (worker)	3.E+14	Drop and crush (worker)
^{239}Pu	**1.E+05**	Fire release (worker)	7.E+14	Drop and crush (worker)
^{240}Pu	**1.E+05**	Fire release (worker)	3.E+14	Drop and crush (worker)
^{241}Pu	**7.E+06**	Fire release (worker)	1.E+20	Drop and crush
^{241}Am	**2.E+05**	Fire release (worker)	2.E+13	Drop and crush (worker)

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

For the longer lived radionuclides (for example ^{14}C, ^{226}Ra and ^{238}U) and radionuclides with long lived daughters (for example ^{238}Pu and ^{241}Pu), it is the road construction scenario. For the vault system for all other geosphere and biosphere combinations (Table XVII), the road construction scenario is the most limiting scenario for all radionuclides. Table XVIII compares the limiting concentrations for the trench and vault systems with a sand geosphere under temperate conditions. An institutional control period of 100 years is assumed. Table XVIII shows that the vault concentration is at least a factor of 20 higher than the trench concentration for any given radionuclide. In general, the vault concentrations are more than four orders of higher for the shorter lived radionuclides, whilst they are about two to three order of magnitude higher for the longer lived radionuclides. The higher vault limits are a consequence of the greater containment offered by the vault affecting the nature and timing of the exposure scenarios.

The illustrative activity limits (Bq) for the post-closure period are presented in Tables XIX and XX for the leaching scenario (SCE1B). The following general comments can be made relating to the results:

— The activity which can be disposed to a disposal facility in the clay geosphere is many orders of magnitude higher than the sandy geosphere due to the longer transport time to the geosphere-biosphere interface and the greater dilution in the geosphere and biosphere.

— For the sandy geosphere (see for example Table XXI), the vault system generally allows higher total activities by up to two orders of magnitude than the trench system for the short lived radionuclides such as ^{3}H, ^{90}Sr and ^{137}Cs since the transport time of radionuclides to the well is greater for the vault (due to reduced leaching) and so there is greater decay. For the long lived radionuclides, such as ^{129}I and ^{238}U, the difference is often less than an order of magnitude since the delay offered by the vault system is insignificant in comparison with radionuclides' half-lives.

— For the sandy geosphere, the arid climate allows up to two orders of magnitude more radionuclides to be disposed due to the reduced infiltration and leaching, and an increased depth of unsaturated zone.

— For the clay geosphere, it is only the long lived radionuclides (^{99}Tc, ^{129}I, ^{234}U, ^{235}U and ^{238}U) which have any significant impact since the long transport time to the geosphere-biosphere interface results in the shorter lived radionuclides decaying to insignificant levels.

— The long transport time to the interface in the clay system results in there being only minor differences in the activity limits for the trench and vault systems for the radionuclides of concern.

— For the clay system under the arid climate, there is less dilution of activity in the river due to the lower river flow rate. This, together with the higher irrigation rate, results in one to two orders of magnitude lower activity limits for the clay system under the arid climate than under the temperate climate.

In order to check and build confidence in the illustrative post-closure activity limits derived in this report, a comparison was undertaken between them and the limits derived by the NEA in 1987 [14] for a scenario considered in both studies: the human intrusion scenario. When comparing with the NEA fully engineered system under temperate conditions for a road construction scenario, there is a good agreement (within one order of magnitude) for all radionuclides, except for ^{14}C, ^{99}Tc and ^{137}Cs (Table XXII). The reasons for the discrepancies for these three radionuclides can be explained and are provided in the footnotes to Table XXII. A good agreement is also found when comparing the NEA residence on the minimum engineered system with the on-site residence scenario for the trench (Table XXIII). Finally, a comparison was performed between the NEA house construction scenario and the road construction "hot-spot" scenario in this publication, which presents similarities with the building of foundation for a house. Once again, a good agreement is found for all radionuclides, except for ^{14}C and ^{99}Tc (Table XXIV). Reasons for the discrepancies for these two radionuclides are given in the footnotes to Table XXIV.

TABLE XV. CONCENTRATIONS (Bq·kg^{-1} OF WASTE) FOR POST-CLOSURE ROAD CONSTRUCTION AND ON-SITE RESIDENCE SCENARIOS FOR THE TRENCH DISPOSAL SYSTEMS

Radionuclide	**Road Construction (SCE7B) Institutional control period and depth**						**Hot Spot (SCE7B) Institutional control**			**On-site Residence: Soil (SCE6B) Institutional control**			**On-site Residence: Radon (SCE6B) Institutional control**		
	30y 3m	**30y 6&9m**	**100y 3m**	**100y 6&9m**	**300y 3m**	**300y 6&9m**	**30y**	**100y**	**300y**	**30y**	**100y**	**300y**	**30y**	**100y**	**300y**
^{3}H	2.E+11	9.E+10	1.E+13	5.E+12	8.E+17	3.E+17	1.E+12	7.E+13	5.E+18	**1.E+06**	7.E+07	5.E+12			
^{14}C	1.E+09	6.E+08	1.E+09	6.E+08	1.E+09	6.E+08	9.E+09	9.E+09	1.E+10	**4.E+05**	4.E+05	4.E+05			
^{41}Ca	5.E+09	2.E+09	5.E+09	2.E+09	5.E+09	2.E+09	4.E+10	4.E+10	4.E+10	**2.E+05**	2.E+05	2.E+05			
^{55}Fe	7.E+12	3.E+12	1.E+20	1.E+20	1.E+20	1.E+20	5.E+13	1.E+20	1.E+20	**3.E+11**	2.E+19	1.E+20			
^{60}Co	4.E+06	2.E+06	4.E+10	2.E+10	9.E+19	4.E+19	2.E+07	2.E+11	1.E+20	**1.E+05**	1.E+09	2.E+18			
^{59}Ni	1.E+10	6.E+09	1.E+10	6.E+09	1.E+10	6.E+09	9.E+10	9.E+10	9.E+10	**1.E+07**	1.E+07	1.E+07			
^{63}Ni	7.E+09	3.E+09	1.E+10	5.E+09	5.E+10	2.E+10	5.E+10	8.E+10	3.E+11	**6.E+06**	1.E+07	4.E+07			
^{90}Sr	2.E+07	1.E+07	1.E+08	5.E+07	1.E+10	6.E+09	2.E+08	8.E+08	9.E+10	**2.E+03**	1.E+04	1.E+06			
^{93}Zr	5.E+08	2.E+08	5.E+08	2.E+08	5.E+08	2.E+08	3.E+09	3.E+09	3.E+09	**3.E+06**	3.E+06	3.E+06			
^{94}Nb	1.E+05	5.E+04	1.E+05	5.E+04	1.E+05	5.E+04	7.E+05	7.E+05	7.E+05	**3.E+03**	3.E+03	3.E+03			
^{99}Tc	1.E+09	4.E+08	1.E+09	4.E+08	1.E+09	4.E+08	7.E+09	7.E+09	7.E+09	**3.E+03**	3.E+03	3.E+03			
^{129}I	1.E+07	4.E+06	1.E+07	4.E+06	1.E+07	4.E+06	6.E+07	6.E+07	6.E+07	**2.E+03**	2.E+03	2.E+03			
^{134}Cs	3.E+09	1.E+09	5.E+19	2.E+19	1.E+20	1.E+20	2.E+10	1.E+20	1.E+20	**8.E+07**	1.E+18	2.E+19			
^{137}Cs	6.E+05	3.E+05	3.E+06	1.E+06	3.E+08	1.E+08	4.E+06	2.E+07	2.E+09	**2.E+04**	8.E+04	9.E+06			
^{151}Sm	6.E+09	3.E+09	1.E+10	4.E+09	5.E+10	2.E+10	4.E+10	7.E+10	3.E+11	**1.E+08**	2.E+08	1.E+09			
^{226}Ra	6.E+04	3.E+04	6.E+04	3.E+04	6.E+04	3.E+04	4.E+05	4.E+05	4.E+05	**8.E+02**	8.E+02	9.E+02	3.E+07	3.E+07	4.E+07
^{228}Ra	2.E+06	8.E+05	9.E+09	4.E+09	4.E+19	2.E+19	1.E+07	6.E+10	1.E+20	**2.E+04**	1.E+08	6.E+17			
^{232}Th	5.E+04	2.E+04	5.E+04	2.E+04	5.E+04	2.E+04	4.E+05	3.E+05	3.E+05	**6.E+02**	6.E+02	6.E+02			
^{234}U	3.E+06	1.E+06	3.E+06	1.E+06	3.E+06	1.E+06	2.E+07	2.E+07	2.E+07	**3.E+05**	3.E+05	3.E+05			
^{235}U	1.E+06	5.E+05	1.E+06	5.E+05	1.E+06	4.E+05	8.E+06	8.E+06	7.E+06	**5.E+04**	5.E+04	4.E+04			
^{238}U	2.E+06	9.E+05	2.E+06	9.E+05	2.E+06	9.E+05	1.E+07	1.E+07	1.E+07	**1.E+05**	1.E+05	1.E+05			
^{237}Np	4.E+05	2.E+05	4.E+05	2.E+05	4.E+05	2.E+05	3.E+06	3.E+06	3.E+06	**2.E+04**	2.E+04	2.E+04			
^{238}Pu	3.E+05	1.E+05	6.E+05	3.E+05	3.E+06	1.E+06	2.E+06	4.E+06	2.E+07	**9.E+04**	2.E+05	8.E+05			
^{239}Pu	2.E+05	1.E+05	2.E+05	1.E+05	3.E+05	1.E+05	2.E+06	2.E+06	2.E+06	**7.E+04**	7.E+04	7.E+04			
^{240}Pu	2.E+05	1.E+05	3.E+05	1.E+05	3.E+05	1.E+05	2.E+06	2.E+06	2.E+06	**7.E+04**	7.E+04	7.E+04			
^{241}Pu	1.E+07	4.E+06	1.E+07	5.E+06	1.E+07	6.E+06	7.E+07	7.E+07	1.E+08	**2.E+06**	2.E+06	3.E+06			
^{241}Am	3.E+05	1.E+05	4.E+05	2.E+05	5.E+05	2.E+05	2.E+06	2.E+06	3.E+06	**8.E+04**	8.E+04	1.E+05			

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) Activity limits for chains containing radiologically significant long lived daughters can become more restrictive with time due to ingrowth of daughters. However, for the trench disposal system at 10,000 years, no activity limits are more restrictive than those given in the table, except for ^{234}U (1.E+04) and ^{235}U (3.E+03).

(4) The most restrictive limit for each radionuclide is emboldened.

TABLE XVI. CONCENTRATIONS (Bq·kg^{-1} OF WASTE) FOR THE POST-CLOSURE ROAD CONSTRUCTION, ON-SITE RESIDENCE AND BATHTUBBING SCENARIOS FOR THE VAULT DISPOSAL SYSTEM WITH A CLAY GEOSPHERE UNDER TEMPERATE CONDITIONS

Radionuclide	**Road Construction (SCE7B)**		**Road Construction Hot Spot (SCE7B)**	**On-site Residence: Radon (SCE6B)**			**Bathtubbing (SCE4B)**
	500y 6m	**500y 9m**	**500y**	**30y**	**100y**	**300y**	**60y**
^{3}H	1.E+20	1.E+20	1.E+20				**3.E+06**
^{14}C	**1.E+09**	**1.E+09**	1.E+10				2.E+09
^{41}Ca	5.E+09	4.E+09	4.E+10				**1.E+09**
^{55}Fe	1.E+20	1.E+20	1.E+20				**2.E+17**
^{60}Co	1.E+20	1.E+20	1.E+20				**1.E+09**
^{59}Ni	1.E+10	1.E+10	1.E+11				**2.E+09**
^{63}Ni	2.E+11	2.E+11	1.E+12				**2.E+09**
^{90}Sr	2.E+12	1.E+12	1.E+13				**5.E+04**
^{93}Zr	**4.E+08**	**4.E+08**	3.E+09				1.E+10
^{94}Nb	1.E+05	**9.E+04**	8.E+05				7.E+06
^{99}Tc	9.E+08	8.E+08	7.E+09				**1.E+07**
^{129}I	9.E+06	7.E+06	6.E+07				**5.E+03**
^{134}Cs	1.E+20	1.E+20	1.E+20				**6.E+12**
^{137}Cs	3.E+10	3.E+10	2.E+11				**8.E+04**
^{151}Sm	**2.E+11**	**2.E+11**	1.E+12				7.E+11
^{226}Ra	6.E+04	**5.E+04**	5.E+05	8.E+09	9.E+09	9.E+09	4.E+05
^{228}Ra	1.E+20	1.E+20	1.E+20				**5.E+08**
^{232}Th	5.E+04	**4.E+04**	4.E+05				3.E+05
^{234}U	3.E+06	**2.E+06**	2.E+07				1.E+09
^{235}U	9.E+05	**7.E+05**	6.E+06				2.E+08
^{238}U	**2.E+06**	**2.E+06**	1.E+07				5.E+08
^{237}Np	**3.E+05**	**3.E+05**	3.E+06				2.E+08
^{238}Pu	**1.E+07**	**1.E+07**	9.E+07				5.E+08
^{239}Pu	**2.E+05**	**2.E+05**	2.E+06				3.E+08
^{240}Pu	**2.E+05**	**2.E+05**	2.E+06				3.E+08
^{241}Pu	**2.E+07**	**2.E+07**	1.E+08				1.E+10
^{241}Am	6.E+05	**5.E+05**	5.E+06				3.E+08

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) Activity limits for chains containing radiologically significant long lived daughters can become more restrictive with time due to ingrowth of daughters. For the vault disposal system at 10 000 years, no activity limits are more restrictive than those given in the table, except for ^{234}U (4.E+05) and ^{235}U (1.E+05).

(4) Since the building foundations for the residence scenario are assumed to be set at 3 m, only the gas pathway is relevant for the on-site residence scenario for the vault disposal facility with its cap of 3 m.

(5) Bathtubbing scenario is only applicable for the vault disposal system with a clay geosphere and temperate climate. The combination of infiltration rate and hydraulic parameters results in bathtubbing after 60 years. It is conservatively assumed that land contaminated by bathtubbing is used for growing crops.

(6) The most restrictive limit for each radionuclide is emboldened.

TABLE XVII. CONCENTRATIONS (Bq·kg^{-1} OF WASTE) FOR THE POST-CLOSURE ROAD CONSTRUCTION AND ON-SITE RESIDENCE SCENARIOS FOR ALL OTHER VAULT DISPOSAL SYSTEMS

Radionuclide	**Road Construction (SEC7B)**		**Road Construction Hot Spot (SCE7B)**	**On-site Residence: Radon (SCE6B)**		
	500y 6m	**500y 9m**	**500y**	**30y**	**100y**	**300y**
^{3}H	**1.E+20**	**1.E+20**	**1.E+20**			
^{14}C	**1.E+09**	**1.E+09**	1.E+10			
^{41}Ca	5.E+09	**4.E+09**	4.E+10			
^{55}Fe	**1.E+20**	**1.E+20**	**1.E+20**			
^{60}Co	**1.E+20**	**1.E+20**	**1.E+20**			
^{59}Ni	**1.E+10**	**1.E+10**	1.E+11			
^{63}Ni	**2.E+11**	**2.E+11**	1.E+12			
^{90}Sr	2.E+12	**1.E+12**	1.E+13			
^{93}Zr	**4.E+08**	**4.E+08**	3.E+09			
^{94}Nb	1.E+05	**9.E+04**	8.E+05			
^{99}Tc	9.E+08	**8.E+08**	7.E+09			
^{129}I	9.E+06	**7.E+06**	6.E+07			
^{134}Cs	**1.E+20**	**1.E+20**	**1.E+20**			
^{137}Cs	**3.E+10**	**3.E+10**	2.E+11			
^{151}Sm	**2.E+11**	**2.E+11**	1.E+12			
^{226}Ra	6.E+04	**5.E+04**	5.E+05	8.E+09	9.E+09	9.E+09
^{228}Ra	**1.E+20**	**1.E+20**	**1.E+20**			
^{232}Th	5.E+04	**4.E+04**	4.E+05			
^{234}U	3.E+06	**2.E+06**	2.E+07			
^{235}U	9.E+05	**7.E+05**	6.E+06			
^{238}U	**2.E+06**	**2.E+06**	1.E+07			
^{237}Np	**3.E+05**	**3.E+05**	3.E+06			
^{238}Pu	**1.E+07**	**1.E+07**	9.E+07			
^{239}Pu	**2.E+05**	**2.E+05**	2.E+06			
^{240}Pu	**2.E+05**	**2.E+05**	2.E+06			
^{241}Pu	**2.E+07**	**2.E+07**	1.E+08			
^{241}Am	6.E+05	**5.E+05**	5.E+06			

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) Activity limits for chains containing radiologically significant long lived daughters can become more restrictive with time due to ingrowth of daughters. For the vault disposal system at 10 000 years, no activity limits are more restrictive than those given in the table, except for ^{234}U (4.E+05) and ^{235}U (1.E+05).

(4) Since the building foundations for the residence scenario are assumed to be set at 3 m, only the gas pathway is relevant for the on-site residence scenario for the vault disposal facility with its cap of 3 m.

(5) Bathtubbing scenario is only applicable for the vault disposal system with a clay geosphere and temperate climate. The combination of infiltration rate and hydraulic parameters results in bathtubbing after 60 years. It is conservatively assumed that land contaminated by bathtubbing is used for growing crops.

(6) The most restrictive limit for each radionuclide is emboldened.

TABLE XVIII. LIMITING CONCENTRATIONS (Bq·kg^{-1}) AND ASSOCIATED CALCULATION CASES FOR THE POST-CLOSURE ROAD CONSTRUCTION AND ON-SITE RESIDENCE SCENARIOS FOR THE TRENCH AND VAULT DISPOSAL SYSTEMS WITH A SAND GEOSPHERE UNDER TEMPERATE CONDITONS WITH AN INSTITUTIONAL CONTROL PERIOD OF 100 YEARS

Radionuclide	**Trench**		**Vault**	
	Limiting Concentration (Bq·kg^{-1} of waste)	**Associated Scenario**	**Limiting Concentration (Bq·kg^{-1} of waste)**	**Associated Scenario**
^{3}H	**7.00E+07**	Onsite residence (soil)	1.E+20	Road construction
^{14}C	**4.00E+05**	Onsite residence (soil)	1.00E+10	Road construction
^{41}Ca	**2.00E+05**	Onsite residence (soil)	4.00E+10	Road construction
^{55}Fe	**2.00E+19**	Onsite residence (soil)	1.E+20	Road construction
^{60}Co	**1.00E+09**	Onsite residence (soil)	1.E+20	Road construction
^{59}Ni	**1.00E+07**	Onsite residence (soil)	1.00E+11	Road construction
^{63}Ni	**1.00E+07**	Onsite residence (soil)	1.00E+12	Road construction
^{90}Sr	**1.00E+04**	Onsite residence (soil)	1.00E+13	Road construction
^{93}Zr	**3.00E+06**	Onsite residence (soil)	3.00E+09	Road construction
^{94}Nb	**3.00E+03**	Onsite residence (soil)	8.00E+05	Road construction
^{99}Tc	**3.00E+03**	Onsite residence (soil)	7.00E+09	Road construction
^{129}I	**2.00E+03**	Onsite residence (soil)	6.00E+07	Road construction
^{134}Cs	**1.00E+18**	Onsite residence (soil)	1.E+20	Road construction
^{137}Cs	**8.00E+04**	Onsite residence (soil)	2.00E+11	Road construction
^{151}Sm	**2.00E+08**	Onsite residence (soil)	1.00E+12	Road construction
^{226}Ra	**8.00E+02**	Onsite residence (soil)	5.00E+05	Road construction
^{228}Ra	**1.00E+08**	Onsite residence (soil)	1.E+20	Road construction
^{232}Th	**6.00E+02**	Onsite residence (soil)	4.00E+05	Road construction
^{234}U	**3.00E+05**	Onsite residence (soil)	2.00E+07	Road construction
^{235}U	**5.00E+04**	Onsite residence (soil)	6.00E+06	Road construction
^{238}U	**1.00E+05**	Onsite residence (soil)	1.00E+07	Road construction
^{237}Np	**2.00E+04**	Onsite residence (soil)	3.00E+06	Road construction
^{238}Pu	**2.00E+05**	Onsite residence (soil)	9.00E+07	Road construction
^{239}Pu	**7.00E+04**	Onsite residence (soil)	2.00E+06	Road construction
^{240}Pu	**7.00E+04**	Onsite residence (soil)	2.00E+06	Road construction
^{241}Pu	**2.00E+06**	Onsite residence (soil)	1.00E+08	Road construction
^{241}Am	**8.00E+04**	Onsite residence (soil)	5.00E+06	Road construction

Notes:

(1) Activity limits calculated using a dose limit of 1 mSv.y^{-1} for each radionuclide.
(2) Activity limits calculated assuming a probability of unity for each scenario.
(3) Activity limits for chains containing radiologically significant long lived daughters can become more restrictive with time due to ingrowth of daughters. However, for the trench disposal system at 10,000 years, no activity limits are more restrictive than those given in the table, except for ^{234}U (1.E+04) and ^{235}U (3.E+03). For the vault disposal system at 10,000 years, no activity limits are more restrictive than those given in the table, except for ^{234}U (4.E+05) and ^{235}U (1.E+05).
(4) The most restrictive limit for each radionuclide is emboldened.

TABLE XIX. TOTAL ACTIVITIES (Bq) FOR THE POST-CLOSURE LEACHING SCENARIO FOR THE TRENCH DISPOSAL SYSTEM

Radionuclide	**Sand Temperate**	**Sand Arid**	**Clay Temperate**	**Clay Arid**
^{3}H	**7.E+11**	3.E+12	1.E+20	1.E+20
^{14}C	**9.E+11**	3.E+12	1.E+20	9.E+18
^{41}Ca	**2.E+12**	6.E+12	1.E+20	1.E+20
^{55}Fe	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{60}Co	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{59}Ni	**3.E+14**	1.E+15	1.E+20	1.E+20
^{63}Ni	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{90}Sr	**8.E+13**	4.E+15	1.E+20	1.E+20
^{93}Zr	**2.E+12**	5.E+12	1.E+20	1.E+20
^{94}Nb	**4.E+12**	1.E+13	1.E+20	1.E+20
^{99}Tc	**2.E+10**	5.E+10	2.E+17	1.E+16
^{129}I	**4.E+08**	1.E+09	6.E+14	3.E+13
^{134}Cs	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{137}Cs	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{151}Sm	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{226}Ra	**1.E+12**	5.E+13	1.E+20	1.E+20
^{228}Ra	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{232}Th	**3.E+10**	9.E+10	1.E+20	1.E+20
^{234}U	**2.E+10**	3.E+10	4.E+19	2.E+18
^{235}U	**2.E+10**	3.E+10	5.E+16	2.E+15
^{238}U	2.E+11	**8.E+10**	2.E+16	8.E+14
^{237}Np	**1.E+11**	4.E+11	1.E+20	1.E+19
^{238}Pu	**4.E+13**	7.E+13	1.E+20	1.E+20
^{239}Pu	**1.E+11**	5.E+11	1.E+20	7.E+19
^{240}Pu	**3.E+11**	3.E+12	1.E+20	1.E+19
^{241}Pu	**2.E+16**	5.E+16	1.E+20	1.E+20
^{241}Am	**7.E+14**	2.E+15	1.E+20	1.E+20

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for each radionuclide.

(2) The most restrictive limit for each radionuclide is emboldened.

TABLE XX. TOTAL ACTIVITIES (Bq) FOR THE POST-CLOSURE LEACHING SCENARIO FOR THE VAULT DISPOSAL SYSTEM

Radionuclide	**Sand Temperate**	**Sand Arid**	**Clay Temperate**	**Clay Arid**
^{3}H	**1.E+13**	3.E+13	1.E+20	1.E+20
^{14}C	**7.E+13**	3.E+14	1.E+20	1.E+20
^{41}Ca	**1.E+14**	3.E+14	1.E+20	1.E+20
^{55}Fe	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{60}Co	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{59}Ni	**3.E+14**	1.E+15	1.E+20	1.E+20
^{63}Ni	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{90}Sr	**5.E+16**	2.E+17	1.E+20	1.E+20
^{93}Zr	**2.E+13**	5.E+13	1.E+20	1.E+20
^{94}Nb	**8.E+12**	5.E+13	1.E+20	1.E+20
^{99}Tc	**4.E+13**	9.E+13	2.E+17	2.E+16
^{129}I	3.E+09	**2.E+09**	6.E+14	3.E+13
^{134}Cs	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{137}Cs	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{151}Sm	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{226}Ra	**2.E+12**	1.E+14	1.E+20	1.E+20
^{228}Ra	**1.E+20**	**1.E+20**	**1.E+20**	**1.E+20**
^{232}Th	**3.E+10**	9.E+10	1.E+20	1.E+20
^{234}U	**2.E+10**	4.E+10	4.E+19	3.E+18
^{235}U	**3.E+10**	5.E+10	5.E+16	2.E+15
^{238}U	1.E+11	**6.E+10**	2.E+16	8.E+14
^{237}Np	**6.E+11**	1.E+12	1.E+20	1.E+19
^{238}Pu	**5.E+13**	1.E+14	1.E+20	1.E+20
^{239}Pu	**3.E+11**	4.E+12	1.E+20	7.E+19
^{240}Pu	**1.E+12**	5.E+13	1.E+20	1.E+19
^{241}Pu	**8.E+16**	1.E+17	1.E+20	1.E+20
^{241}Am	**3.E+15**	5.E+15	1.E+20	1.E+20

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XXI. LIMITING TOTAL ACTIVITY (Bq) FOR THE POST-CLOSURE LEACHING SCENARIO FOR THE TRENCH AND VAULT DISPOSAL SYSTEMS WITH A SAND GEOSPHERE UNDER TEMPERATE CONDITIONS

Radionuclide	Trench	Vault
^{3}H	**7.E+11**	1.E+13
^{14}C	**9.E+11**	7.E+13
^{41}Ca	**2.E+12**	1.E+14
^{55}Fe	**1.E+20**	**1.E+20**
^{60}Co	**1.E+20**	**1.E+20**
^{59}Ni	**3.E+14**	**3.E+14**
^{63}Ni	**1.E+20**	1.E+20
^{90}Sr	**8.E+13**	5.E+16
^{93}Zr	**2.E+12**	2.E+13
^{94}Nb	**4.E+12**	8.E+12
^{99}Tc	**2.E+10**	4.E+13
^{129}I	**4.E+08**	3.E+09
^{134}Cs	**1.E+20**	**1.E+20**
^{137}Cs	**1.E+20**	**1.E+20**
^{151}Sm	**1.E+20**	**1.E+20**
^{226}Ra	**1.E+12**	2.E+12
^{228}Ra	**1.E+20**	**1.E+20**
^{232}Th	**3.E+10**	**3.E+10**
^{234}U	**2.E+10**	**2.E+10**
^{235}U	**2.E+10**	3.E+10
^{238}U	2.E+11	**1.E+11**
^{237}Np	**1.E+11**	6.E+11
^{238}Pu	**4.E+13**	5.E+13
^{239}Pu	**1.E+11**	3.E+11
^{240}Pu	**3.E+11**	1.E+12
^{241}Pu	**2.E+16**	8.E+16
^{241}Am	**7.E+14**	3.E+15

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XXII. COMPARISON OF ACTIVITY CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE POST-CLOSURE ROAD CONSTRUCTION SCENARIO FOR A VAULT DISPOSAL SYSTEM FROM THIS PUBLICATION AND THE NEA PUBLICATION

Radionuclide	This Publication	NEA Publication [14]
^{14}C	1.E+09	4.E+11
^{90}Sr	1.E+12	6.E+11
^{94}Nb	9.E+04	4.E+05
^{99}Tc	8.E+08	8.E+10
^{129}I	7.E+06	2.E+07
^{137}Cs	3.E+10	8.E+08
^{226}Ra	5.E+04	1.E+05
^{238}U	2.E+06	4.E+07
^{237}Np	3.E+05	1.E+06
^{239}Pu	2.E+05	1.E+06
^{240}Pu	2.E+05	1.E+06
^{241}Pu	2.E+07	6.E+07
^{241}Am	5.E+05	2.E+06

Notes:

(1) The activity limits in this report are for intrusion into the vault facility to a depth of 9m and an institution control period of 500 years and a dose limit of 1 $mSv.y^{-1}$ (see Table XVI). No differentiation is made between temperate and arid climate systems for this scenario.

(2) The NEA activity limits are for intrusion into the fully engineered facility in a temperate area to a depth of 10 m with an institution control period of 300 years. In the NEA publication [14] a dose limit of 5 $mSv.y^{-1}$ was applied to this scenario. However, for the purposes of comparison with the activity limits derived in this report, the NEA activity limits have been recalculated using a dose limit of $1mSv.y^{-1}$.

(3) Activity limits for ^{14}C and ^{99}Tc differ between the two studies by more than an order of magnitude due to an order of magnitude difference in the dose coefficients used in the studies.

(4) The activity limit for ^{137}Cs differs by more than an order of magnitude between the two studies due to the difference in the institutional control period assumed.

TABLE XXIII. COMPARISON OF ACTIVITY CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE POST-CLOSURE ON-SITE RESIDENCE SCENARIO FOR A TRENCH DISPOSAL SYSTEM FROM THIS PUBLICATION AND THE NEA PUBLICATION

Radionuclide	This Report	NEA Publication [14]
^{14}C	4.E+05	2.E+06
^{90}Sr	2.E+03 to 1.E+06	1.E+04 to 5.E+06
^{94}Nb	3.E+03	9.E+03
^{99}Tc	3.E+03	1.E+04
^{129}I	2.E+03	2.E+04
^{137}Cs	2.E+04 to 9.E+06	6.E+04 to 2.E+07
^{226}Ra	9.E+02	4.E+02
^{238}U	1.E+05	2.E+05
^{237}Np	2.E+04	1.E+04
^{239}Pu	7.E+04	5.E+04
^{240}Pu	7.E+04	5.E+04
^{241}Pu	2.E+06	8.E+05
^{241}Am	8.E+04	3.E+04

Notes:

(1) The activity limits from this report are for the trench facility with an institution control period of 300 years and a dose limit of 1 $mSv.y^{-1}$. For ^{90}Sr and ^{137}Cs institutional control periods from 30 to 300 years are considered. No differentiation is made between temperate and arid climate systems for this scenario.

(2) The NEA activity limits are for the minimum engineered facility in a temperate area with an institution control period of 300 years and a dose limit of 1 $mSv.y^{-1}$. For ^{90}Sr and ^{137}Cs, institutional control periods from 50 to 300 years are considered.

TABLE XXIV. COMPARISON OF ACTIVITY CONCENTRATIONS ($Bq \cdot kg^{-1}$ OF WASTE) FOR THE HOT SPOT CALCULATION FOR THE POST-CLOSURE ROAD CONSTRUCTION SCENARIO FOR THE TRENCH DISPOSAL SYSTEM FROM THIS REPORT AND THE HOUSE CONCENTRATION SCENARIO FROM THE NEA REPORT

Radionuclide	This Report	NEA Report [14]
^{14}C	1.E+10	2.E+11
^{90}Sr	2.E+08 to 9.E+10	1.E+09 to 6.E+11
^{94}Nb	7.E+05	8.E+04
^{99}Tc	7.E+09	8.E+10
^{129}I	6.E+07	6.E+06
^{137}Cs	4.E+06 to 2.E+09	6.E+05 to 2.E+08
^{226}Ra	4.E+05	8.E+04
^{238}U	1.E+07	4.E+06
^{237}Np	3.E+06	8.E+05
^{239}Pu	2.E+06	1.E+06
^{240}Pu	2.E+06	1.E+06
^{241}Pu	1.E+08	4.E+07
^{241}Am	3.E+06	1.E+06

Notes:

(1) The activity limits from this report are for the trench facility with an institution control period of 300 years and a dose limit of 1 $mSv.y^{-1}$. For ^{90}Sr and ^{137}Cs institutional control periods from 30 to 300 years are considered. No differentiation is made between temperate and arid climate systems for the scenario.

(2) The NEA activity limits are for the minimum engineered facility in a temperate area with an institution control period of 300 years. For ^{90}Sr and ^{137}Cs institutional control periods from 50 to 300 years are considered. In the NEA report [15] a dose limit of 5 $mSv.y^{-1}$ was applied to the scenario. However, for the purposes of comparison with the activity limits derived in this report, the NEA activity limits have been recalculated using a dose limit of $1mSv.y^{-1}$.

(3) Activity limits for ^{14}C and ^{99}Tc differ between the two studies by more than an order of magnitude due to an order of magnitude difference in the dose coefficients used in the studies.

5.5.3. Combined operational and post-closure periods

In Sections 5.5.1 and 5.5.2 activity limits have been provided for the operational and post-closure periods, respectively. It is possible to compare these operational and post-closure limits for the same disposal system (i.e. the same disposal facility, geosphere and biosphere). As way of illustration, Table XXV summarizeds the most restrictive operational and post-closure limits from the trench disposal facility in a sandy geosphere under arid conditions, assuming an institutional control period of 100 years. Two post-closure limits are provided: one relating to the most limiting scenario that affects only part of the disposal facility (for example the on-site residence scenario or the road construction scenario); and the other relating to the leaching scenario that affects the entire disposal facility. For the leaching scenario, the total activity provided in Table XIX has been divided by the total mass of waste in the disposal facility to derive a facility-averaged concentration ($Bq\ kg^{-1}$ of waste). Table XXVI provides a similar summary for the vault disposal facility in a clay geosphere under temperate conditions, also assuming an institutional control period of 100 years.

Table XXV shows that for 16 of the 43 radionuclides, calculations were undertaken for only operational period primarily due to the short half-lives of the radionuclides (see Section 5.2.1). For the remaining 27 radionuclides, the operational limit is more restrictive than the post-closure limit for only seven radionuclides. These are mostly relatively short lived radionuclides, such as ^{55}Fe and ^{60}Co, whose activity will have decayed significantly during the 100 year institutional control period resulting in comparatively small post-closure impacts. Furthermore, as stated in Section 5.5.1, the operational limits for the trench should be seen as first iteration limits that can be increased if appropriate engineering and waste management controls are put in place. For the 20 radionuclides for which the post-closure limit is more restrictive, there is an approximately even divide between the residence and leaching scenarios being more restrictive, depending on factors such as the half-life and mobility of the radionuclide in question.

Table XXVI shows that for the 27 radionuclides for which operational and post-closure calculations were undertaken, the operational limit is more restrictive than the post-closure limit for only two radionuclides. These are ^{55}Fe and ^{134}Cs, whose activity will have decayed significantly during the 100 year institutional control period, due to their short half-lives, resulting in comparatively small post-closure impacts. Furthermore, as stated in Section 5.5.1, the vault disposal facility's additional engineering and high standard of waste management ensures higher operational limits than for the trench disposal facility. For the 25 radionuclides for which the post-closure limit is more restrictive, the leaching scenario is never the most restrictive scenario due to the vault's engineered barriers and the clay geosphere causing greater delay, and hence greater decay, of the radionuclides prior to their release into the biosphere. There is an approximately even divide between the bathtubbing and road construction scenarios being the most restrictive scenario, depending on factors such as the half-life and mobility of the radionuclides.

The illustrative activity limits in Tables XIX–XXVI have been calculated assuming reference dose limits of 20 $mSv.y^{-1}$ for workers and 1 $mSv.y^{-1}$ for members of the public. As noted in Section 3.3. the radiation dose from a single source of radiation such as a waste repository should be restricted according to a dose constraint and a value of 0.3 mSv per year is recommended. Section 3.3. also gives guidance on reference values for judging the significance of human intrusion and recommends that if exposures of members of the public approaching 100 mSv per year are predicted then efforts should be made to reduce the probability of human intrusion or its consequences. If the dose constraint (0.3 mSv per year) and the reference value of 100 mSv per year for situation of human intrusion (road construction, on-site residence) are used instead of the reference dose limit (1 mSv) the results presented in Table XXV and Table XXVI will be changed, as follows:

- activity limits for operational period scenarios resulting in the exposure of workers remain the same;
- activity limits for operational period scenarios resulting in the exposure of the public are reduced by a factor of 70%;
- activity limits for post-closure period scenarios resulting in the exposure of the public from natural processes are reduced by a factor of 70%; and
- activity limits for post-closure period scenarios resulting in the exposure of the public from human intrusion (e.g. on-site residence and road construction) are increased by up to two orders of magnitude.

TABLE XXV. LIMITING OPERATIONAL AND POST-CLOSURE PERIOD CONCENTRATIONS FOR THE TRENCH DISPOSAL FACILITY WITH A SAND GEOSPHERE UNDER ARID CONDITIONS

Radionuclide	Operational Period		Post-closure Period	
	Limiting Concentration ($Bq\cdot kg^{-1}$ of waste)	Associated Scenario	Limiting Concentration ($Bq\cdot kg^{-1}$ of waste)	Associated Scenario
^{3}H	3.E+06	Liquid release (public)	**7.E+04**	Leaching scenario
^{10}Be	**1.E+07**	Liquid release (public)	N/A	N/A
^{14}C	4.E+06	Liquid release (public)	**7.E+04**	Leaching scenario
^{22}Na	**1.E+06**	Liquid release (public)	N/A	N/A
^{41}Ca	3.E+06	Liquid release (public)	**1.E+05**	Leaching scenario
^{54}Mn	**8.E+06**	Liquid release (public)	N/A	N/A
^{55}Fe	**4.E+07**	Liquid release (public)	2.E+19	On-site residence (soil)
^{59}Ni	3.E+08	Liquid release (public)	**1.E+07**	On-site residence (soil)
^{63}Ni	1.E+08	Liquid release (public)	**1.E+07**	On-site residence (soil)
^{60}Co	**1.E+06**	Liquid release (public)	1.E+09	On-site residence (soil)
^{65}Zn	**7.E+06**	Liquid release (public)	N/A	N/A
^{90}Sr	2.E+04	Liquid release (public)	**1.E+04**	On-site residence (soil)
^{93}Zr	2.E+07	Liquid release (public)	**1.E+05**	Leaching scenario
^{94}Nb	2.E+06	Liquid release (public)	**3.E+03**	On-site residence (soil)
^{99}Tc	8.E+04	Liquid release (public)	**1.E+03**	Leaching scenario
^{106}Ru	**8.E+05**	Liquid release (public)	N/A	N/A
^{110m}Ag	**2.E+06**	Direct irradiation (worker)	N/A	N/A
^{121m}Sn	**1.E+07**	Liquid release (public)	N/A	N/A
^{125}Sb	**3.E+06**	Liquid release (public)	N/A	N/A
^{126}Sn	**7.E+05**	Liquid release (public)	N/A	N/A
^{129}I	9.E+02	Liquid release (public)	**2.E+01**	Leaching scenario
^{134}Cs	**1.E+06**	Liquid release (public)	1.E+18	On-site residence (soil)
^{137}Cs	1.E+06	Liquid release (public)	**8.E+04**	On-site residence (soil)
^{144}Ce	**1.E+07**	Liquid release (public)	N/A	N/A
^{147}Pm	**6.E+07**	Liquid release (public)	N/A	N/A
^{151}Sm	**1.E+08**	Liquid release (public)	2.E+08	On-site residence (soil)
^{152}Eu	**4.E+06**	Direct irradiation (worker)	N/A	N/A
^{154}Eu	**3.E+06**	Direct irradiation (worker)	N/A	N/A
^{204}Tl	**1.E+07**	Liquid release (public)	N/A	N/A
^{210}Pb	**5.E+03**	Liquid release (public)	N/A	N/A
^{226}Ra	5.E+03	Liquid release (public)	**8.E+02**	On-site residence (soil)
^{227}Ac	**2.E+04**	Liquid release (public)	N/A	N/A
^{228}Ra	**4.E+04**	Liquid release (public)	1.E+08	On-site residence (soil)
^{232}Th	3.E+04	Liquid release (public)	**6.E+02**	On-site residence (soil)
^{234}U	4.E+04	Liquid release (public)	**7.E+02**	Leaching scenario
^{235}U	4.E+04	Liquid release (public)	**7.E+02**	Leaching scenario
^{238}U	4.E+04	Liquid release (public)	**2.E+03**	Leaching scenario
^{237}Np	**2.E+03**	Liquid release (public)	2.E+04	On-site residence (soil)
^{238}Pu	**1.E+05**	Liquid release (public)	2.E+05	On-site residence (soil)
^{239}Pu	1.E+05	Liquid release (public)	**1.E+04**	Leaching scenario
^{240}Pu	1.E+05	Liquid release (public)	**7.E+04**	Leaching scenario
^{241}Pu	6.E+06	Liquid release (public)	**2.E+06**	On-site residence (soil)
^{241}Am	2.E+05	Fire release (worker)	**8.E+04**	On-site residence (soil)

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

TABLE XXVI. LIMITING OPERATIONAL AND POST-CLOSURE PERIOD CONCENTRATIONS FOR THE VAULT DISPOSAL FACILITY WITH A CLAY GEOSPHERE UNDER TEMPERATE CONDITIONS

Radionuclide	Operational Period		Post-closure Period	
	Limiting Concentration ($Bq \cdot kg^{-1}$ of waste)	Associated Scenario	Limiting Concentration ($Bq \cdot kg^{-1}$ of waste)	Associated Scenario
^{3}H	1.E+12	Gas release (public)	**3.E+06**	Bathtubbing
^{10}Be	**1.E+20**	Drop and crush	N/A	N/A
^{14}C	8.E+09	Gas release (public)	**1.E+09**	Road Construction
^{22}Na	**4.E+11**	Drop and crush (worker)	N/A	N/A
^{41}Ca	3.E+14	Drop and crush (worker)	**1.E+09**	Bathtubbing
^{54}Mn	**1.E+12**	Drop and crush (worker)	N/A	N/A
^{55}Fe	**1.E+12**	Drop and crush (worker)	2.E+17	Bathtubbing
^{59}Ni	9.E+13	Drop and crush (worker)	**2.E+09**	Bathtubbing
^{63}Ni	1.E+20	Drop and crush	**2.E+09**	Bathtubbing
^{60}Co	4.E+11	Drop and crush (worker)	**1.E+09**	Bathtubbing
^{65}Zn	**2.E+12**	Drop and crush (worker)	N/A	N/A
^{90}Sr	1.E+20	Drop and crush	**5.E+04**	Bathtubbing
^{93}Zr	1.E+20	Drop and crush	**4.E+08**	Road Construction
^{94}Nb	5.E+11	Drop and crush (worker)	**9.E+04**	Road Construction
^{99}Tc	1.E+20	Drop and crush	**1.E+07**	Bathtubbing
^{106}Ru	**4.E+12**	Drop and crush (worker)	N/A	N/A
^{110m}Ag	**3.E+11**	Drop and crush (worker)	N/A	N/A
^{121m}Sn	**1.E+14**	Drop and crush (worker)	N/A	N/A
^{125}Sb	**2.E+12**	Drop and crush (worker)	N/A	N/A
^{126}Sn	**5.E+11**	Drop and crush (worker)	N/A	N/A
^{129}I	2.E+13	Drop and crush (worker)	**5.E+03**	Bathtubbing
^{134}Cs	**6.E+11**	Drop and crush (worker)	6.E+12	Bathtubbing
^{137}Cs	2.E+12	Drop and crush (worker)	**8.E+04**	Bathtubbing
^{144}Ce	**8.E+10**	Drop and crush (crane operator)	N/A	N/A
^{147}Pm	**2.E+17**	Drop and crush (worker)	N/A	N/A
^{151}Sm	2.E+16	Drop and crush (worker)	**2.E+11**	Road Construction
^{152}Eu	**8.E+11**	Drop and crush (worker)	N/A	N/A
^{154}Eu	**7.E+11**	Drop and crush (worker)	N/A	N/A
^{204}Tl	**6.E+14**	Drop and crush (worker)	N/A	N/A
^{210}Pb	**9.E+13**	Drop and crush (worker)	N/A	N/A
^{226}Ra	4.E+05	Gas release (public)	**5.E+04**	Road Construction
^{227}Ac	**2.E+12**	Drop and crush (worker)	N/A	N/A
^{228}Ra	1.E+20	Drop and crush	**5.E+08**	Bathtubbing
^{232}Th	3.E+14	Drop and crush (worker)	**4.E+04**	Road Construction
^{234}U	3.E+14	Drop and crush (worker)	**2.E+06**	Road Construction
^{235}U	4.E+12	Drop and crush (worker)	**7.E+05**	Road Construction
^{238}U	3.E+13	Drop and crush (worker)	**2.E+06**	Road Construction
^{237}Np	3.E+12	Drop and crush (worker)	**3.E+05**	Road Construction
^{238}Pu	3.E+14	Drop and crush (worker)	**1.E+07**	Road Construction
^{239}Pu	7.E+14	Drop and crush (worker)	**2.E+05**	Road Construction
^{240}Pu	3.E+14	Drop and crush (worker)	**2.E+05**	Road Construction
^{241}Pu	1.E+20	Drop and crush	**2.E+07**	Road Construction
^{241}Am	2.E+13	Drop and crush (worker)	**5.E+05**	Road Construction

Notes:

(1) Activity limits calculated using a dose limit of 1 $mSv.y^{-1}$ for the public and 20 $mSv.y^{-1}$ for workers for each radionuclide.

(2) Activity limits calculated assuming a probability of unity for each scenario.

(3) The most restrictive limit for each radionuclide is emboldened.

5.6. TREATMENT OF UNCERTAINTIES ARISING FROM THE ASSESSMENT

When analysing the illustrative activity limits derived in this report, it must not be forgotten that several kinds of uncertainties are associated with the results of such a quantitative safety assessment. The sources of these uncertainties need to be appropriately managed when deriving activity limits, since such uncertainties could strongly affect the results obtained. It is important to remember that there are uncertainties associated with the activity limits derived and that the limits must not be seen as absolutely certain numbers.

Although the sources of uncertainty are identical for both the operational and post-closure scenarios identified within a safety assessment, the ways to address these uncertainties can strongly differ between the two periods. For the operational period, actual experience from operational radioactive waste facilities is available and growing. This experience can directly be used to reduce the operational safety-relevant uncertainties for a disposal facility (equipment failure, containment failure, shielding, probabilities of incidents, etc.). For post-closure scenarios, with radiological impacts far into the future, equivalent experience is unavailable. Here, the reduction of uncertainties should be sought by other means or other approaches, such as a comprehensive scenario development, the use of different conceptual models and different lines of reasoning, and a detailed characterization of the components of the disposal system that contribute to safety.

The different ways that the uncertainties can be dealt with for the operational and post-closure scenarios relate to the fact that the operational period is by definition an active period, during which safety is primarily based on active measures, such as monitoring, surveillance and maintenance. Whilst these measures will remain in place during the active institutional control period of the post-closure phase, such measures will not be in place for the remainder of the post-closure period.

This difference explains why for a real disposal system the different types of uncertainties remain an essential element of the safety assessments for the post-closure scenarios, while for the operational scenarios benefit can be taken from active safety measures and existing experience. Therefore, the discussion of the importance of the uncertainties will largely focus on uncertainties associated with the safety assessment of the post-closure period.

Uncertainties in the safety assessment approach used to derive the illustrative activity limits can be considered to arise from three sources [27]:

- uncertainty in the description of the operation of the facility and the evolution of the disposal system over the timescales of interest (scenario uncertainty);
- uncertainty in the conceptual, mathematical and computer models used to simulate the behaviour and evolution of the disposal system (e.g. owing to the practical limitations on representing the complete real world system in a manageable model, approximations used in solving the model equations, and incorrect implementation of the mathematical models in codes) (model uncertainty); and
- uncertainty/variability in the data and parameters used as inputs in the calculations.

Each of these sources of uncertainty is discussed in turn below, in relation to the illustrative activity limits obtained in Section 5.5. In particular, information is provided on their management and their relative importance for the illustrative cases considered in the current study. However, a detailed uncertainty analysis is beyond the scope of this publication, mainly because of the hypothetical nature of the illustrative disposal systems that are considered. It should also be recognized that there are additional sources of uncertainty that need to be considered when deriving and applying waste acceptance criteria (and not just deriving activity limits). Such sources of uncertainly include uncertainty associated with the measurement of activity in waste packages and uncertainty with the existing inventory already disposed in the site (if it is an operating site). Again, a more detailed consideration of these uncertainties is beyond the scope of this publication given its focus on the approach for derivation of activity limits.

Both operational and post-closure scenarios are stylized situations due to the complex nature of the disposal system, and due to the unpredictable evolution of the disposal system and human behaviour. Therefore, there is uncertainty associated with the *scenarios* considered in this report. However, it has been observed in several generic studies, for example [13, 14], that the appropriate selection of scenarios provides an adequate estimate of the range of likely impacts. Such an approach has been adopted in this report. Thus the scenarios selected are considered to cover a sufficiently wide range of possible events, pathways and parameters for the purpose of illustrative modelling and criteria development. For example, the associated range in results for the post-closure scenarios used to derive activity concentration limits (SCE 4B, SCE6B, SCE7B) can be seen in Tables XV–XVII for the trench and vault disposal systems. Differences in the results for different scenarios can be as little as an order of magnitude (for example for ^{14}C disposed to the vault) to several orders of magnitude (for example for ^{3}H disposed to the vault); three to five orders of magnitude are a common range.

With regard to the particular case of the evolution of the biosphere component of the disposal system, the following points should be noted.

— Any actual biosphere system can be highly variable within a timeframe of a few decades; this variability has been used to justify recourse to the use of ‘stylized’ biosphere systems called reference biospheres; such biospheres are often based on the structure of actual or past biosphere systems.
— The uncertainties attached to the biosphere modelling does not have the same relevance as the other components of the disposal system, because the biosphere does not contribute to the safe confinement of the disposed waste.
— However, some exercises performed previously (for example [26]) have shown that the eventual effective dose received by an average member of a small agricultural community is a robust estimator, i.e. its variability is lower than that of some of its internal parameters (e.g. radionuclide distribution and transfer coefficients).

Although not explicitly addressed in this report, the uncertainty associated with biosphere evolution can be considered to a certain extent by comparing the variations in results for the temperate and the arid states of the disposal system. For all the post-closure and operational scenarios used to derive activity concentration limits, with the exception of the bathtubbing scenario, the water budget and other environmental parameters are not relevant to the scenario, leading to an independence of the results with regards to this particular type of uncertainty. For the post-closure leaching scenario (SCE1B) and the operational liquid release scenario (SCE9A), the climate conditions influence various parameters (infiltration rate, unsaturated zone thickness, soil erosion rate, irrigation rate and river flow rate). Tables XIX and XX indicate nonetheless that the changes induced by such variations do not exceed two (and rarely one) orders of magnitude for the radionuclides considered. A similar conclusion is true for the operational liquid release scenario (see Tables IX–XII).

The effect of the uncertainties associated with the *conceptual, mathematical and computer models* used to simulate the post-closure behaviour and evolution of the considered disposal systems can be investigated in a preliminary manner by comparing the results from two independent sets of calculations performed, with different conceptual, mathematical and/or computer models (see Appendix V).

For the post-closure scenarios SCE4B, SCE6B, SCE7B (Section 5.4.4), the same conceptual and mathematical models, consistent with those described in Section 5.4.4 and Appendix II, were used in the two independent sets of calculations. However, the computer tools used were different. Nonetheless, an exact agreement was achieved between the two sets of results.

For the post-closure leaching scenario (SCE1B), two different models were developed for assessing the associated impact of a limited set of radionuclides (^{3}H, ^{14}C, ^{90}Sr, ^{129}I, ^{137}Cs, ^{226}Ra, ^{238}U and ^{241}Pu) (see Appendix V). Comparison of the results given in Table V.1. shows that for many of the radionuclides considered differences do not exceed an order of magnitude. The adoption of a compartment approach for the geosphere modelling and the reduction of the disposal facility lay-out to a single area source (as modelled in the Model A) has produced more restrictive activity limits than its discretization into ten point sources (as modelled in model B). Note that the former, more cautious approach is the approach that has been used to derive the activity limits for the post-closure leaching scenario for the full set of radionuclides given in Tables XIX and XX.

The effect of uncertainties in *parameter values* can be appreciated by comparing the different calculation cases for the same scenario. For the post-closure scenarios used to derive activity concentration limits, calculation cases have been undertaken for which different values for the institutional control period, dilution factors, depth of intrusion and cover thickness have been adopted. The effect of these variations can be seen in Tables XV and XVI Differences can be as little as an order of magnitude (for example, for ^{94}Nb for the road construction scenario) to more than ten orders of magnitude (for example, for ^{60}Co disposed for the on-site residence scenario) depending upon the half-life of the radionuclide considered. The greater the half-life, the smaller the range. One to two orders of magnitude is a common range for the radionuclides with a half-life in excess of 30 years. Primarily, the difference in ranges reflects the importance of the duration of the institutional control period for the shorter lived radionuclides.

For the post-closure leaching scenario (SCE1B), parameters such as water infiltration rate through the repository, depth to water table, hydraulic conductivity, water table gradient, distance between repository and geosphere-biosphere interface, and distribution coefficients have been varied for the different calculation cases (Tables XIX and XX). As noted above, changes caused by varying the climate related parameters of infiltration rate, unsaturated zone thickness, erosion rate, irrigation rate and river flow rate do not exceed two orders of magnitude for the radionuclides considered, indeed they are mostly less than an order of magnitude. Varying parameters dependent upon the level of engineering in the facility (bulk density, kinetic porosity, infiltration rate, facility distribution coefficients, and waste to backfill ratio) have similar relatively limited effects. However, varying parameters dependent upon geosphere parameters, such as hydraulic conductivity, water table gradient, distance between repository and geosphere-biosphere interface, and geosphere distribution coefficients, result at a minimum in a variation of four orders of magnitude. This supports the findings of other similar studies, such as [18], that have highlighted the geosphere specific nature of activity limits derived for post-closure leaching scenarios.

6. GUIDANCE ON THE USE OF THE APPROACH AND THE DERIVED ILLUSTRATIVE ACTIVITY LIMITS

The approach developed in this report can be used:

(1) to derive disposal system specific activity limits for the disposal of radioactive waste to near surface facilities in a clear and well-documented manner; and

(2) in training events concerned with the safety assessment of such disposal facilities.

The derived illustrative activity limits can be used:

(a) as a starting point for countries that do not yet have disposal system specific data and are at an early stage in a near surface disposal programme. As such the obtained illustrative activity limits:

  (i) allow testing of the feasibility of disposal concepts with regards to the wastes which are to be managed, and

  (ii) can be used at the ‘screening’ stage of the near surface disposal programme.

(b) as a benchmark against which to compare disposal system specific limits. However, such comparisons should be undertaken with care. Checks should be made to ensure that any differences between the illustrative systems considered in this report and a site specific disposal system are documented and their effects on the derived limits correctly understood. These differences can be at the level of the assessment context (e.g. different regulatory criteria or timescales) or of the system description (with the associated scenarios, conceptual and mathematical models and data). Since the illustrative waste activity limits in this publication have been derived in a clear and well documented manner, the reasons for any differences between the illustrative limits and system specific limits should be easily identifiable, provided that the derivation of the system specific limits has been clearly documented.

(c) in training events relating to the safety assessment of near surface disposal facilities.

In applying the approach presented in this publication it is vital to consider the following points.

— In a comprehensive development of a specific disposal system the derivation of activity limits is an iterative process, that is strongly influenced by many factors, such as operational practices, disposal facility design, and site and waste characteristics. This iterative process is a mean to systematically optimize the disposal system and its planned or real operation, and in turn the associated activity limits. The iterative development of a safety case is system specific, and has not be applied to the illustrative cases in this publication.

— The approach is applicable to the assessment of the operational and post-closure radiological impacts of the disposal (as opposed to the transport, conditioning and/or storage) of solid and solidified radioactive waste to operating or planned near surface facilities.

When using the illustrative activity limits derived in this report the following caveats should be carefully considered.

- Application of the illustrative activity limits should be restricted to assessment contexts and disposal systems similar to those in this report; and any extrapolation to different assessment contexts and disposal systems should be undertaken with care. This is especially the case for the values associated with the post-closure leaching scenario which can be highly sensitive to geosphere parameters, such as hydraulic conductivities, water table gradient, distance between repository and geosphere-biosphere interface, and geosphere distribution coefficients (see Section 5.6).

- Compliance with activity limits derived using the approach described in this report does not, on its own, ensure the overall safety of disposal, since such safety also relies upon other elements. Some of these elements are qualitative (e.g. safety culture, quality management), some relate to the construction period (e.g. quality of materials and construction practices) and some relate to the operational period (e.g. safety measures and procedures during operation, waste emplacement operations).

- The various illustrative operational and post-closure scenarios that have been developed in Sections 5.3 and 5.4 can be classified on the basis of the mass of waste that contributes to the radiological impact of the scenario. At one extreme of the considered scenarios there is an individual waste package, at the other extreme there is the entire inventory of the disposal facility. A cautious approach would be to apply all the illustrative activity concentration limits obtained to each waste package for all scenarios except those that involve the entire inventory of the disposal system. For those that involve the entire inventory of the disposal system, total activity limits could be used instead.

- When using the illustrative limits derived in this report, the various sources of uncertainty discussed in Section 5.6 should be taken into consideration. Due to these uncertainties, the illustrative limits should not be seen as recommended limits to be used to assess site specific disposal systems. It is more appropriate to regard them as order of magnitude estimates. Therefore slight (factor of two or three) exceedance of these values would not necessarily mean that the specific disposal systems would result in non-compliance with concrete safety requirements and criteria, established by national regulatory bodies.

- The derivation of the illustrative limits assumes that there is a linear relationship between dose and the radionuclide inventory disposed in the facility. This might not always be the case, for example when the release of radionuclides is solubility limited.

- Each scenario assessed has been assigned a probability of unity and has been treated with the same dose limit (1 $mSv \cdot y^{-1}$ for the public and 20 $mSv \cdot y^{-1}$ for workers for Tables IX to XXVI). It is clear that for a specific disposal system dose (or risk) constraints and the ALARA principle have to be applied. It should also be noted that probabilities of scenario occurence can be introduced, associated with the use of risk limits and constraints, although it might not be straightforward to justify the choice of probability values. To obtain the illustrative activity limits that correspond to a given dose constraint (e.g. 0.3 $mSv \cdot y^{-1}$), the activity limits given in Tables IX to XXVI have to be multiplied with the appropriate factor corresponding to the dose constraint (e.g. 0.3).

— For the calculation of the illustrative activity limits, the radionuclides have been considered independently of each other (except for the decay chains), i.e. as if they were alone in the radioactive waste. When managing a spectrum of several radionuclides, as it is most often the case for real wastes, it is necessary to combine their impacts (summation rule) in order to ensure that the dose limit is not exceeded for any given scenario. This has been discussed in Section 4.6. Because of this, it is not really possible to identify the limiting radionuclides from the illustrative activity limits in this report. This can only be done in a summation exercise for a real waste spectrum of radionuclides.

— When applying the summation rule, it has to be recognized that the limiting value for each radionuclide varies as a function of time. Thus, the derivation of activity limits for a spectrum of radionuclides can be treated as an optimization problem.

— The radioactive inventory has been considered to be uniformly distributed in the disposal facility for all scenarios, with the exception of the hot-spot calculation for the road construction scenario for the post-closure period. As was explained above, the different operational and post-closure scenarios affect different masses of disposed waste. By applying all the activity concentration limits obtained to individual waste packages a cautious approach is followed with respect to non-uniform activity distributions within the disposal facility.

— Although the scope of the illustrative cases in this report has been limited to disposal facility with a uniformly distributed activity, the approach discussed can also be applied to disposal installations with highly non-uniform activity distributions (e.g. disposal of disused sealed sources) to derive associated activity limits.

— When considering the illustrative operational activity limits, it is important to recognize that operational safety can be managed in a more active and direct manner then post-closure safety, by modifying or adopting additional operational procedures in an iterative manner. The illustrative operational period calculations in this report very much represent a first iteration, prior to the introduction of additional procedures, and so the associated values derived must be considered accordingly. This is especially the case for the trench disposal facility where minimal engineering and waste management procedures are in place. Indeed, the trench disposal facility can be seen as a starting point for an iterative assessment in which additional engineered barriers and operational controls can be progressively introduced until the desired degree of safety is achieved.

— Activity limits derived from certain post-closure scenarios for chains containing radiologically significant long lived daughters can become more restrictive with time due to the ingrowth of daughters, see for example the discussion in [28]. Thus it is advisable that activity limits for such chains should be derived for a range of appropriate times, not just for the time at which institutional control is assumed to cease.

Appendix I

DISPOSAL SYSTEM DATA

The following data have been used for the derivation of the illustrative activity limits for the particular cases considered in Section 5 of this publication and should not be further applied without appropriate care.

Most of the data have been derived from [A-1] and [A-2], the remainder have been derived specifically for this study.

I-1. DISPOSAL FACILITY

I.1.1. Facility layout

— The facility is composed of 10 vaults or trenches (two rows of five units) (see Fig. I.1).

— The site boundary of the facility is represented by a fence located at 70 m from the edge of the disposal units (see Fig. I.1).

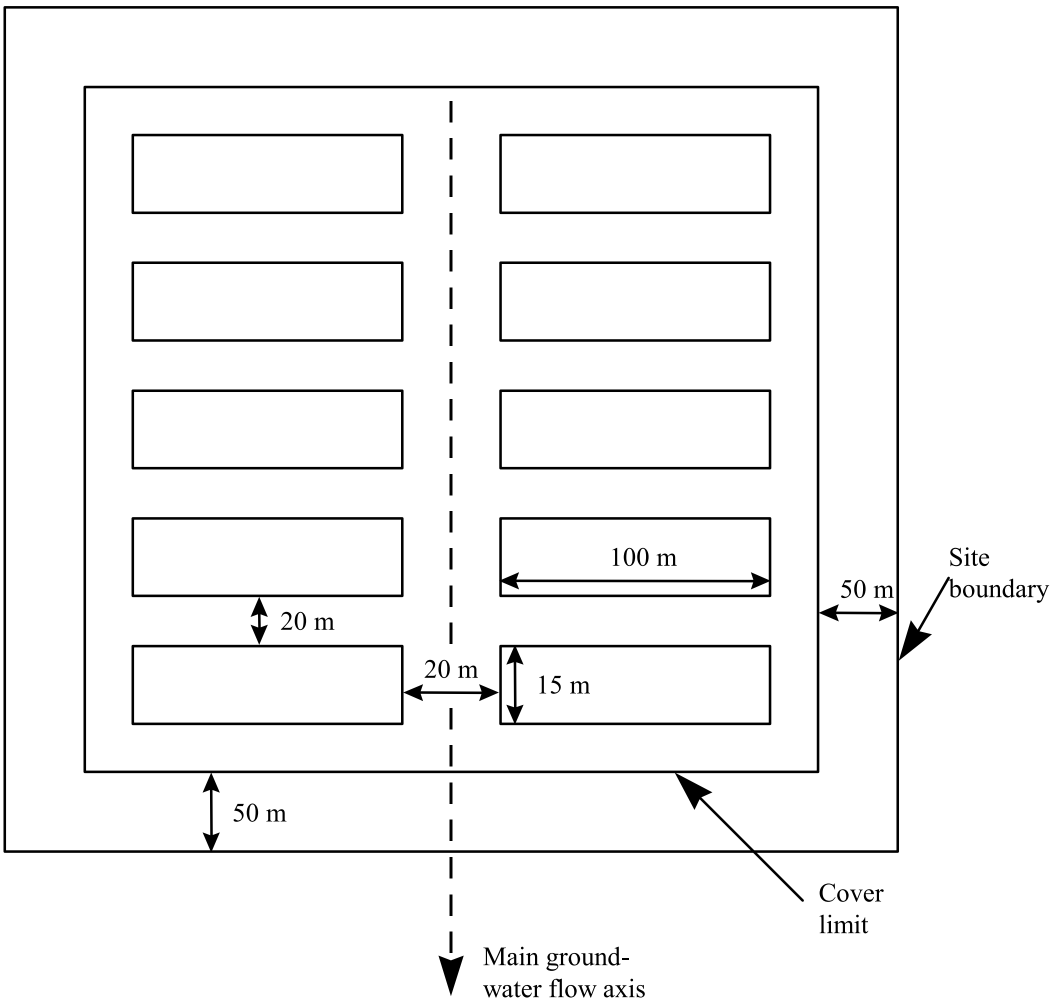

FIG. I.1. Facility layout.

I.1.2. Trench

— Dimensions:

internal length = 100 m

internal width = 15 m

internal depth = 6 m

distance between two trenches (edge to edge) = 20 m

— Homogeneous waste backfilled with native soil (consistent with [I.1], two cases considered: sand backfill; and clay backfill):

Kinematic and total porosity = 0.4

hydraulic conductivity = 10^{-5} m·s^{-1}

total bulk density = 500 kg·m^{-3}

— Cover made periodically of soil/fill material, a geotextile layer, stones and ashes/small aggregate):

thickness = 1 m

kinematic and total porosity = 0.3

hydraulic conductivity = 10^{-7} m·s^{-1}

bulk density = 1500 kg·m^{-3}

I.1.3. Vault

— Dimensions:

internal length = 100 m

internal width = 15 m

internal depth = 6 m

distance between two vaults (edge to edge) = 20 m

— Waste put in a concrete matrix within drums, which are grouted:

waste kinematic and total porosity = 0.3

waste bulk density = 1500 kg·m^{-3}

matrix hydraulic conductivity = 10^{-9} m·s^{-1}

backfill kinematic and total porosity = 0.3

backfill hydraulic conductivity = 10^{-9} m·s^{-1}

backfill bulk density = 1600 kg·m^{-3}

ratio waste and matrix volume : backfill volume = 2 : 1

— Vault base, walls and roof made of concrete:

thickness = 0.3 m

concrete kinematic and total porosity = 0.25

concrete bulk density = 1600 kg·m^{-3}

concrete hydraulic conductivity = 10^{-9} m·s^{-1}

— Multi-layered cap:

width of cap element above one vault = 30 m

length = 120 m

thickness = 3 m

thickness of rolled clay within cap = 1 m

slope of all layers = 1 : 30

runoff ratio due to slope = 0.3 (0.0 after institutional control period)

(this ratio gives the fraction of water that does not infiltrate through the cap)

kinematic and total porosity = 0.3

bulk density = 1600 kg.m^{-3}

clay hydraulic conductivity = 10^{-9} m.s^{-1}

degraded clay hydraulic conductivity = 10^{-7} m.s^{-1}

time of cap degradation : at the cessation of the institutional control period

cap material hydraulic conductivity = 10^{-7} m.s^{-1}

— Distance between the ground of the vault and the crane cabin = 10 m

I.2. GEOSPHERE

I.2.1. Sandy geosphere

— Properties:

unsaturated zone thickness (temperate) = 2 m below base of disposal unit

unsaturated zone thickness (arid) = 20 m below base of disposal unit

average moisture content of the unsaturated zone = 0.15

kinematic and total porosity = 0.3

bulk density = 2000 kg.m^{-3}

hydraulic conductivity = 10^{-5} m.s^{-1}

hydraulic gradient = 1 in 50

saturated thickness = 15 m

longitudinal dispersivity = distance to outlet / 10 (m)

transverse dispersivity = distance to outlet / 50 (m)

I.2.2. Clay geosphere

— Properties:

no unsaturated zone under temperate conditions below base of disposal unit

unsaturated zone thickness (arid) = 3 m below base of disposal unit

average moisture content of the unsaturated zone (arid) = 0.25

kinematic and total porosity = 0.3

bulk density = 1600 $kg.m^{-3}$

hydraulic conductivity = 10^{-7} $m.s^{-1}$

hydraulic gradient = 1 in 500

saturated thickness = 15 m

longitudinal dispersivity = distance to outlet / 3 (m)

transverse dispersivity = distance to outlet / 15 (m)

— Due to these properties, it is not possible to envisage that a well could be set to extract water from the saturated zone for the assumed water abstraction rate considered in Appendix III.1.

I.3. BIOSPHERE

— Climate:

temperate precipitation = 1000 $mm \cdot y^{-1}$ (yearly averaged)

temperate actual evapotranspiration = 400 $mm \cdot y^{-1}$

arid precipitation = 560 $mm \cdot y^{-1}$ over a two month period in a year

arid potential evapotranspiration = 2490 $mm \cdot y^{-1}$

arid monthly potential evapotranspiration = 208 mm

— Surface water body:

river located 1500 m from site boundary

river flow-rate (arid) = 10^{6} $m^{3} \cdot y^{-1}$

river flow-rate (temperate) = 10^{7} $m^{3} \cdot y^{-1}$

river width = 2 m

river depth = 0.5 m

river length over which leachate released from the disposal facility during the operational period is assumed to be mixed = 500 m

river length river over which radionuclides are assumed to be discharged from the clay geosphere during the post-closure period = 2000 m

suspended particles in water = 10^{-2} $kg \cdot m^{-3}$

— Top Soil:

thickness = 0.25 m

water infiltration rate through soil (temperate) = 0.6 $m \cdot y^{-1}$

water infiltration rate through soil (arid) = 0.144 $m \cdot y^{-1}$

kinematic and total porosity = 0.3

bulk density = 1800 $kg \cdot m^{-3}$

erosion rate (temperate) = $2\ 10^{-4}\ y^{-1}$

erosion rate (arid) = $2\ 10^{-3}\ y^{-1}$.

REFERENCES TO APPENDIX I

[I-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Co-ordinated Programme on the Safety Assessment of Near Surface Radioactive Waste Disposal Facilities (NSARS): Specification for Test Case 2C, IAEA, Vienna (1995).

[I-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Near Surface Radioactive Waste Disposal Facilities: Model Intercomparison using Simple Hypothetical Data (Test Case 1), IAEA-TECDOC-846, Vienna (1995).

Appendix II

MATHEMATICAL MODELS

The following models have been used for the derivation of the illustrative activity limits for the particular cases considered in Section 5 of this publication and should not be further applied without appropriate care.

II.1. GAS RELEASE (SCE1A AND SCE11A)

II.1.1. Source term modelling

For ^{3}H and ^{14}C, the release rate in gas, R_{gas} [$Bq \cdot y^{-1}$], is given by:

$$R_{gas} = \frac{A_r \cdot f_{gas}}{\tau_{gas}}$$

where

A_r is the residual activity (assuming loss by decay only) [Bq]

f_{gas} is the fraction of the activity associated with the gas [-]

τ_{gas} is the average timescale of generation of each gas [y]

For ^{222}Rn, the release rate in gas, R_{gas} [$Bq \cdot y^{-1}$], can be derived using the equations used in [II.1]:

$$R_{gas} = \lambda \cdot D_{area} A \cdot \rho_{bd} \cdot \tau \cdot H_1 \cdot e^{\frac{-h_2}{H_2}}$$

where

λ is the decay constant of ^{222}Rn [y^{-1}]

D_{area} is the surface area of the disposal unit [m^2]

A is the ^{226}Ra concentration in the waste [$Bq \cdot kg^{-1}$]

ρ_{bd} is the bulk density of the material in the disposal unit [$kg \cdot m^{-3}$]

τ is the emanation factor, defined as the fraction of the radon atoms produced which escape from the solid phase of the waste into the pore spaces [-]

H_1 is the effective diffusion relaxation length for the waste [m]

h_2 is the thickness of the cover [m]

H_2 is the effective relaxation length of the cover [m]

The associated air concentration of a radionuclide, $C_{air,gas}$ [$Bq.m^{-3}$], can be approximated by:

$$C_{air,gas} = R_{gas} / V_{air}$$

where

R_{gas} is the release rate of the radionuclide in gas [$Bq \cdot y^{-1}$]

V_{air} is the air volume into which the activity released per year is diluted [$m^3 \cdot y^{-1}$]

$$V_{air} = W \cdot u \cdot h \cdot 3.16\ 10^7$$

where

W is the width of the source perpendicular to the wind direction [m]

U is the mean wind speed [$m \cdot s^{-1}$]

H is the height for vertical mixing [m]

$3.16\ 10^7$ are the number of seconds in a year [$s \cdot y^{-1}$]

II.1.2. Dose assessment

The dose to the worker and public due to inhalation of the gas, $Dose_{inh}$ [$Sv \cdot y^{-1}$], is given by:

$$Dose_{inh} = C_{air,gas} \cdot t_{out} \cdot b_r \cdot DF_{inh}$$

where

$C_{air,gas}$ is the concentration of the gas in the air [$Bq \cdot m^{-3}$]

t_{out} is the time spent in the gas plume by the human [$h \cdot y^{-1}$]

$b_{r,}$ is the breathing rate of the human [$m^3 \cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

II.2. DROP AND CRUSH (SCE2A)

II.2.1. Source term modelling

Due to the long exposure distance, the exposed waste is assumed to be a point source.

The photon fluence rate at a reference point located at distance r (cm) from the source is given by:

$$\phi = \frac{S \cdot B \cdot e^{-b}}{4\pi r^2}$$

Where the spatial relationship between source and dose point is:

$$r = \left[x^2 + y^2 + z^2\right]^{1/2}$$

where

x	is the distance between source and dose point perpendicular to shield plane [cm]

y	is the distance between base of source and dose point parallel to shield plane [cm]

z	is the lateral displacement (offset) of the dose point normal to the x–y plane [cm]

And with: $$b = (r/x) \cdot \sum_i \mu_i d_i$$

Where

μ_I is the attenuation factor of the material i crossed

d_i is the thickness of the material i crossed [cm]

ϕ is the fluence rate [photon·s^{-1}·cm^{2}]

S is the activity source [photon·s^{-1}]

B is the buildup factor.

The buildup factor corresponds to the material of a single shield. However, all the relaxation lengths corresponding to the thickness of all the shields positioned between the source and the reference point are taken into account to determine the total buildup factor.

The analytical solution used by MICROSHIELD 5.02 to apply the buildup factor to the Taylor correlation:

$$B(b) = Ae^{-\alpha_1 b} + (1 - A)e^{-\alpha_2 b}$$

where A, α_1 and α_2 are empirical constants which depend on the photon energy.

The energy fluence rate Φ in (MeV.cm^{-2}.sec^{-1}) is obtained by multiplying the fluence rate for each photon group by the group energy. The total energy fluence rate is obtained by summing the contributions for each energy. That is:

$$\Phi\, total = \sum fluence\ i \cdot energy\ i$$

The absorbed dose to air in (mR.h^{-1}) from results of the photon fluence rate calculation is determined using Table 11 of ICRP Publication 51 [II.2].

II.2.2. Dose assessment

II.2.2.1. Operator on the cab

The dose to an operator in the cab due to the external exposure from the waste considered as a source point can be expressed as (in [Sv]):

$$Dose_{irr,op} = Dose_{irr,cab} + Dose_{irr,ladder} + Dose_{irr,walkway}$$

or by:

$$Dose_{irr,op} = DDose_{irr,cab} \cdot t_{cab} + DDose_{irr,ladder} \cdot t_{ladder} + DDose_{irr,walkway} \cdot t_{walkway}$$

where

$DDose_{irr,\ cab}$, $DDose_{irr,\ ladder}$, $DDose_{irr,\ walkway}$ are the effective dose equivalent rate in the cab, on the ladder, and on the walkway [$Sv.h^{-1}$]. They are calculated with Microshield code using Table 2 of ICRP Publication 51 [II.2], with the anterior/posterior geometry for the body on the cab, and posterior/anterior geometry for the body on the ladder and the walkway;

t_{cab} , t_{ladder}, $t_{walkway}$ are the exposure time on the cab, on the ladder, and on the walkway [h].

II.2.2.2. Worker on the ground level

The dose to a worker on the ground level due to the external exposure from the waste considered as a source point can be expressed as (in [Sv]):

$$Dose_{irr,work} = DDose_{irr,work} \cdot t_{work}$$

where

$DDose_{irr,\ work}$ is the effective dose equivalent rate in the ground level [$Sv.h^{-1}$]. It is calculated with Microshield code using Table 2 of ICRP Publication 51 [II.2], with the anterior/posterior geometry for the body;

t_{work} is the exposure time of the worker in front of the waste [h].

II.3. DIRECT IRRADIATION (SCE8A)

II.3.1. Source term modelling

II.3.1.1. Uncovered waste modelling into a trench

During the operation in the vicinity of the uncovered part of tipped waste in a trench, the worker is irradiated by these wastes. The distance between the ground and the worker is assumed to be 1 m.

Due to the short exposure distance (1 m) in comparison with the source term dimensions (tipped waste in a trench), the exposed waste is assumed to be an infinite slab source.

In reality, since the real distance between the source term and the exposed worker is always greater than 1 m, the beta radiation consequences are not evaluated.

The photon fluence rate equation is:

$$\phi = \int_{o}^{v} \frac{BSv\ e^{-b}}{4\pi r^2}$$

The infinite slab source is analytically integrated using the fundamental exponential integral functions and the Taylor formulation for build up factor.

In the above equation: b slabs = (r/X) · $\sum_i \mu_i d_i$

The fluence rate equation with Taylor Build up formulation is:

$$\phi = Sv/2\mu_s\ [\ C_1[E_2(b_{\ slabs}\ .(1+\alpha_1)) - E_2(b1\ .(1+\alpha_1))]]$$
$$+\ Sv/2\mu_s\ [\ C_2[E_2(b_{\ slabs}\ .(1+\alpha_2)) - E_2(b1\ .(1+\alpha_2))]]$$

where

C1 = A / (1+α_1)

C2 = (1 – A) / (1+α_2)

The energy fluence rate Φ [MeV/cm^2/s] is obtained by multiplying the fluence rate for each photon group by the group energy. The total energy fluence rate is obtained by summing the contributions for each energy. That is:

$$\Phi\, total = \sum fluence\ i \cdot energy\ i$$

The absorbed dose to air in (mR.h^{-1}) from results of the photon fluence rate calculation is determined using Table 11 of ICRP Publication 51 [II.2].

II.3.1.2.Covered waste modelling

The same principle of modelling is applied. The unconditioned waste is covered by a soil cover. The attenuation in this cover is taken into account.

II.3.2. Dose assessment

It is supposed for the dose assessment that the worker spends their time partly directly on the unconditioned waste, and most of the time on the cover.

The effective dose due to direct irradiation for worker can be expressed as (in [Sv y^{-1}]):

$$Dose_{irr} = Dose_{irr,cover} + Dose_{irr,waste}$$

where

$Dose_{\ irr,\ cover}$ is the dose to the worker due to the external exposure from the tipped waste covered by uncontaminated soil [Sv y^{-1}];

$Dose_{\ irr,\ waste}$ is the dose to the worker due to the external exposure from the tipped uncovered waste considered as an infinite thick slab [Sv y^{-1}].

The dose on the covered waste is expressed by:

$$Dose_{irr,cover} = DDose_{irr,cover} \cdot t_{cover}$$

where

DDose $_{irr,\ cover}$ is the effective dose equivalent rate on the cover [$Sv.h^{-1}$]. The waste is considered as an infinite thick slab and the cover as a 1m shield. It is calculated with Microshield code using Table 2 of ICRP Publication 51 [II.2], with the isotropic geometry for the body;

t $_{cover}$ is the exposure time of the worker in front of the waste [hy^{-1}].

The dose on the uncovered waste is expressed by:

$$Dose_{irr,waste} = DDose_{irr,waste} \cdot t_{waste}$$

where:

DDose $_{irr,\ waste}$ is the effective dose equivalent rate on the cover [$Sv.h^{-1}$]. The waste is considered as an infinite thick slab. It is calculated with Microshield code using Table 2 of ICRP Publication 51 [II.2], with the isotropic geometry for the body;

t $_{waste}$ is the exposure time of the worker in front of the waste [hy^{-1}].

II.4. LIQUID RELEASE (SCE9A)

II.4.1. Source term modelling

The equation governing the evolution of the residual activity A_r in a trench is:

$$\frac{dA_r}{dt} = (\lambda + ALF)\,A_r$$

where

A_r is the residual activity of the radionuclide as a function of time [Bq]

λ is the radionuclide decay constant [y^{-1}]

ALF is the annual leaching rate of the radionuclide [y^{-1}]

The annual leaching rate (ALF) is the ratio of the activity lost by leaching during the year t, A(t), over the total activity remaining that year, $A_r(t)$. It is expressed as:

$$ALF = \frac{Inf}{H_d\,(\omega_{cd} + \rho_{bd} Kd_d)}$$

where

Inf is the annual infiltration rate [$m \cdot y^{-1}$] accounting for the water budget and the hydraulic conductivity of the trench;

H_d is the trench depth [m];

ω_{cd} is the moisture content of the waste in the trench [-]

ρ_{bd} is the dry bulk density of the waste in the trench [$kg \cdot m^{-3}$]

Kd_d is the radionuclide distribution coefficient in the trench [$m^3 \cdot kg^{-1}$]

By multiplying both parts of the ratio expressing ALF by the area of the disposal unit, the following expression can be derived:

$$ALF = \frac{Q}{V_{dispunit}(\omega_{cd} + \rho_{bd} Kd_d)}$$

where

Q is the annual water flow through the trench [$m^3 \cdot y^{-1}$]

$V_{dispunit}$ is the volume of the trench [m^3]

The radioactivity which leaves the trench is:

$$A(t) = A_r(t) \times ALF \qquad [Bq \cdot y^{-1}].$$

An analytical solution can be presented for this particular case, as:

$$A_r(t) = A_{t=0} e^{-(\lambda + ALF).t}$$

For the purposes of the liquid release scenario, it is cautiously assumed that all this activity enters the drainage system, and is subsequently discharged to the river.

II.4.2. Dose assessment

The dose to a member of the critical group for the liquid release scenario can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{inh} + Dose_{ext} + Dose_{ing}$$

where

$Dose_{inh}$, $Dose_{ext}$ and $Dose_{ing}$ are the doses due to the inhalation, external exposure and the ingestion pathways [$Sv \cdot y^{-1}$].

The dose due to inhalation is expressed as:

$$D_{inh} = A_{soil} \cdot b_r \cdot 8766 \cdot \left[dust_{act} \%_{occup} + dust_{norm} \left(1 - \%_{occup} \right) \right] Df_{inh}$$

where

D_{inh} is the dose due to inhalation [$Sv \cdot y^{-1}$]

A_{soil} is the concentration of the radionuclide in the soil [$Bq \cdot kg^{-1}$ of soil]

b_r is the breathing rate [$m^3 \cdot h^{-1}$]

8766 are the hours in a year [$h \cdot y^{-1}$]

$dust_{act}$ and $dust_{norm}$ are the dust concentrations during ploughing and non-ploughing activities [$kg \cdot m^{-3}$]

$\%_{occup}$ is the occupancy factor for ploughing activities [-]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

The dose due to external exposure is expressed as:

$$D_{ext} = A_{soil} \cdot 8766 \cdot DF_{ext}$$

where

D_{ext} is the dose due to external exposure [$Sv \cdot y^{-1}$]

A_{soil} is the concentration of the radionuclide in the soil [$Bq \cdot kg^{-1}$ of soil]

8766 is the exposure duration [$h \cdot y^{-1}$]

DF_{ext} is the external exposure dose factor [$Sv \cdot h^{-1}.Bq^{-1} \cdot kg$]

The dose due to ingestion is expressed as:

$$D_{ing} = D_{ing\_water} + D_{ing\_fish} + D_{ing\_crop} + D_{ing\_animal}$$

where

D_{ing} is the dose due to ingestion [$Sv \cdot y^{-1}$]

D_{ing_water} is the dose due to filtered water ingestion [$Sv \cdot y^{-1}$]

$$D_{ing\_water} = Q_{water} \cdot C_{water} \cdot \frac{1}{1 + Kd_w \cdot part} \cdot DF_{ing}$$

where

Q_{water} is the annual intake of water [$m^3 \cdot y^{-1}$]

C_{water} is the concentration of radionuclides in water [$Bq \cdot m^{-3}$]

DF_{ing} is the dose factor for ingestion [$Sv \cdot Bq^{-1}$]

Kd_w is the distribution coefficient water/particles [$m^3 \cdot kg^{-1}$]

Part is the river suspended particle concentration [$kg \cdot m^{-3}$]

D_{ing_fish} is the dose due to fish consumption [$Sv \cdot y^{-1}$]:

$$D_{ing\_fish} = Q_{fish} \cdot C_{water} \cdot TF_{fish} \cdot DF_{ing}$$

where

Q_{fish} is the annual fish consumption rate [$kg \cdot y^{-1}$]

TF_{fish} is the concentration ratio for fish [$m^3 \cdot kg^{-1}$]

D_{ing_crop} is the dose due to crop consumption [$Sv \cdot y^{-1}$]:

$$D_{ing\_crop} = \sum_{root,green,grain} \left\{ Q_{crop} \left[C_{water} \frac{Irrig.Int}{Yield} + A_{soil} TF_{crop} \right] DF_{ing} \right\}$$

where

Q_{crop} is the annual crop consumption rate [$kg \cdot y^{-1}$]

Irrig is the depth of irrigation [$m \cdot y^{-1}$]

Int is the interception factor [-]

Yield is the crop yield [$kg \cdot m^{-2} \cdot y^{-1}$]

TF_{crop} is the soil to plant concentration factor for the crop [$Bq \cdot kg^{-1}$ fresh weight / $Bq \cdot kg^{-1}$ dry soil]

D_{ing_animal} is the dose due to animal product consumption [$Sv \cdot y^{-1}$]

$$D_{ing\_animal} = \sum_{beef,milk} \left\{ Q_{animal} \left[q_{water} C_{water} + q_{soil} A_{soil} + q_{pasture} A_{soil} TF_{pasture} \right] \times TF_{animal} DF_{ing} \right\}$$

where

Q_{animal} is the annual animal product consumption rate [$kg \cdot y^{-1}$]

q_{water} is the daily water intake [$m^3 \cdot day^{-1}$]

q_{soil} is the daily soil intake [$kg \cdot day^{-1}$]

$q_{pasture}$ is the daily pasture intake [$kg \cdot day^{-1}$]

$TF_{pasture}$ is the soil to plant concentration factor for pasture [$Bq \cdot kg^{-1}$ fresh weight / $Bq \cdot kg^{-1}$ dry soil];

TF_{animal} is the transfer coefficient to the animal product [$day \cdot kg^{-1}$]

C_{water} is derived from the source term calculations and the river characteristics

A_{soil} is governed by a first order differential equation:

$$\frac{d\ A_{soil}}{dt} = \frac{Irrig}{\rho_{soil} \cdot Th_{soil}} C_{water} - \lambda_{eff} A_{soil}$$

where

Irrig is the irrigation rate [$m \cdot y^{-1}$]

ρ_{soil} is the soil dry bulk density [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

λ_{eff} is an effective decay [y^{-1}]

$$\lambda_{eff} = \lambda + \frac{P_{eff}}{Th_{soil}(\omega_{soil} + \rho_{soil} Kd_s)} + \frac{TF_{plant} \cdot Yield_{plant}}{Th_{soil} \cdot \rho_{soil}} + \lambda_{erosion}$$

where

λ is the radionuclide decay constant [y^{-1}]

P_{eff} is the water infiltration rate through the soil [m y^{-1}]

ω_{soil} is the soil kinematic porosity [-]

Kd_s is the radionuclide distribution coefficient in the soil [$m^3 \cdot kg^{-1}$]

TF_{plant} is the soil to plant concentration factor for the plant [$Bq \cdot kg^{-1}$ fresh weight / $Bq \cdot kg^{-1}$ dry soil]

$Yield_{plant}$ is the annual crop yield [$kg \cdot m^{-2} \cdot y^{-1}$]

$\lambda_{erosion}$ is the soil erosion rate [y^{-1}]

In the case of simple screening calculations, a more simple treatment of A_{soil} can be found in [II.3]

II.5. SOLID RELEASE (SCE10A)

II.5.1. Source term modelling

The release rate of a radionuclide in dust, R_{dust} [$Bq \cdot h^{-1}$], is given by:

$$R_{dust} = f_{rel,dust} \cdot V_{dust} \cdot A_m \cdot \rho_{bd} / t_{dust}$$

where

$f_{rel,dust}$ is the release fraction for the radionuclide [-]

V_{dust} is the volume of the waste from which the dust is released [m^3]

A_m is the specific activity of the radionuclide in the trench [$Bq \cdot kg^{-1}$]

ρ_{bd} is the bulk density of the waste [$kg \cdot m^{-3}$]

t_{dust} is the duration of the dust release [h]

It is assumed that the associated plume is neutrally buoyant. Using an approach consistent with [II.3], the air concentration of a radionuclide at ground level, $C_{air,dust}$ [$Bq \cdot m^{-3}$], at a given distance for prevalent atmospheric conditions is given by:

$$C_{air,dust} = R_{dust} \cdot C_{\mathrm{integ},dust}$$

where

R_{dust} is the release rate of the radionuclide in dust [$Bq \cdot h^{-1}$]

$C_{integ,dust}$ is the time-integrated air concentration at ground level at the given distance for the prevalent atmospheric conditions [$Bq \cdot h \cdot m^{-3} \cdot Bq^{-1}$]

The above equation cautiously assumes that the plume is non-depleting during passage towards the exposed individual. Activity may be deposited on the ground by dry and wet deposition. The surface concentration of a radionuclide, $C_{surf,dust}$ [$Bq \cdot m^{-2}$], resulting from dry and wet deposition is given by:

$$C_{surf,dust} = C_{air,dust} \cdot t_{dep,dust} \cdot \left(V_{g,dust} + W_{out,dust} \cdot h_{dust}\right)$$

where

$C_{air,dust}$ is the concentration of a radionuclide in the dust [$Bq \cdot m^{-3}$]

$t_{dep,dust}$ is the time over which deposition occurs [s]

$V_{g,dust}$ is the dry deposition velocity [$m \cdot s^{-1}$]

$W_{out,dust}$ is the washout coefficient [s^{-1}]

h_{dust} is the plume height [m]

II.5.2. Dose assessment

II.5.2.1 Worker

The dose to a worker exposed to the dust plume can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{sub} + Dose_{inh}$$

where

$Dose_{sub}$ and $Dose_{inh}$ are the doses due to the external irradiation from submersion in the dust plume, and the dose from inhalation of particles in the plume [$Sv \cdot y^{-1}$].

$$Dose_{sub} = C_{dust} \cdot t_{out,dust} \cdot DF_{sub} \cdot Occ_{dust}$$

where

C_{dust} is the concentration of a radionuclide in the dust [Bq·m^{-3}]

$t_{out,dust}$ is the time spent exposed to the dust [h]

DF_{sub} is the dose factor for external irradiation from submersion in the dust [Sv·h^{-1}/Bq·m^{-3}]

Occ_{dust} is the number of dust releases per year [y^{-1}]

$$C_{dust} = A_m \cdot Dust$$

where

A_m is the specific activity of the radionuclide in the trench [Bq·kg^{-1}]

Dust is the dust level in the air from the solid release breathed by the worker [kg·m^{-3}]

$$Dose_{inh} = C_{dust} \cdot t_{out,dust} \cdot b_{r,dust} \cdot DF_{inh} \cdot Occ_{dust}$$

where

C_{dust} is the air concentration of a radionuclide in the dust [Bq·m^{-3}]

$t_{out,dust}$ is the time spent exposed to the dust [h]

$b_{r,dust}$ is the breathing rate of the worker [m^{3}·h^{-1}]

DF_{inh} is the dose factor for inhalation [Sv·Bq^{-1}]

Occ_{dust} is the number of dust releases per year [y^{-1}]

II.5.2.2. Public

The dose to a person living at the site boundary can be expressed as (in [Sv·y^{-1}]):

$$Dose = Dose_{sub} + Dose_{inh} + Dose_{ext} + Dose_{ing}$$

where

$Dose_{sub}$ is the dose due to the external irradiation from submersion in the dust plume [Sv·y^{-1}]

$Dose_{inh}$ is the dose from inhalation of particles in the plume [Sv·y^{-1}]

$Dose_{ext}$ is the dose from external irradiation from deposited activity both during and after the passing of the plume [Sv·y^{-1}]

$Dose_{ing}$ is the dose from the ingestion of activity deposited on leafy green vegetables [$Sv\cdot y^{-1}$].

$$Dose_{sub} = C_{air,dust} \cdot t_{out,dust} \cdot DF_{sub} \cdot Occ_{dust}$$

where

$C_{air,dust}$ is the air concentration of a radionuclide at ground level [$Bq\cdot m^{-3}$]

$t_{out,dust}$ is the time spent outside during the passage of the dust plume [h]

DF_{sub} is the dose factor for external irradiation from submersion in the plume [$Sv\cdot h^{-1}/Bq\cdot m^{-3}$]

Occ_{dust} is the number of dust releases per year [y^{-1}]

$$Dose_{inh} = C_{air,dust} \cdot t_{out,dust} \cdot b_{r,dust} \cdot DF_{inh} \cdot Occ_{dust}$$

where

$C_{air,dust}$ is the air concentration of a radionuclide at ground level [$Bq\cdot m^{-3}$]

$t_{out,dust}$ is the time spent outside during the passage of the dust plume [h]

$b_{r,dust}$ is the breathing rate of the member of the public [$m^3\cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv\cdot Bq^{-1}$]

Occ_{dust} is the number of dust releases per year [y^{-1}]

$$Dose_{ext} = C_{surf,dust} \cdot \left(\frac{1-e^{-\lambda t}}{\lambda t}\right)\left(sf \cdot t_{in} + t_{out}\right) \cdot DF_{ext,surf}$$

where

$C_{surf,dust}$ is the surface concentration of a radionuclide from one release [$Bq\cdot m^{-2}$]

λ is the radionuclide decay constant [y^{-1}]

t is the exposure duration [y]

sf is the indoor shielding factor [-]

t_{in} is the time spent indoors [$h\cdot y^{-1}$]

t_{out} is the time spent outdoors [$h\cdot y^{-1}$]

$DF_{ext,surf}$ is the external exposure dose factor [$Sv\cdot h^{-1}/Bq\cdot m^{-2}$]

$$Dose_{ing} = Q_{gveg} CF_{gveg} C_{surf,dust} \frac{\left(1-e^{-\lambda_{gveg} t}\right)}{\lambda_{gveg}} DF_{ing} \cdot Occ_{dust}$$

where

Q_{gveg} is the vegetable consumption rate [$kg \cdot d^{-1}$]

CF_{gveg} is the initial activity concentration of a radionuclide in green vegetables for unit a real concentration on soil [$Bq \cdot kg^{-1}/Bq \cdot m^{-2}$]

$C_{surf,dust}$ is the concentration of a radionuclide deposited following the dust release on the surface of the soil [$Bq \cdot m^{-2}$]

λ_{gveg} is the effective rate constant for removal of activity from the green vegetable [d^{-1}]

t is the time following the release over which the green vegetables are consumed [d]

Df_{ing} is the dose coefficient for ingestion [$Sv \cdot Bq^{-1}$]

Occ_{dust} is the number of dust releases per year [y^{-1}]

$$CF_{gveg} = \frac{Int_{gveg} \cdot Fract}{Yield_{gveg}}$$

where

Int_{gveg} is the effective interception factor [–]

Fract is the fraction of activity retained after processing [–]

$Yield_{gveg}$ is the yield of fresh green vegetables [$kg\ m^{-2}$]

$$\lambda_{gveg} = \lambda + \lambda_{weath}$$

where

λ is the decay constant of the radionuclide [d^{-1}]

λ_{weath} is the weathering rate of the radionuclide from the green vegetable [d^{-1}]

Note that it is cautiously assumed that the wind direction at the time of each release is towards the exposed member of the public.

II.6. FIRE (SCE12A)

II.6.1. Source term modelling

The release rate of a radionuclide from the fire, R_{fire} [$Bq \cdot h^{-1}$], is given by:

$$R_{fire} = f_{rel,fire} \cdot V_{fire} \cdot A_m \cdot \rho_{bd} / t_{fire}$$

where

$f_{rel,\ fire}$ is the release fraction for the radionuclide [-]

V_{fire} is the volume of the waste consumed in the fire [m^3]

A_m is the specific activity of the radionuclide in the trench [$Bq \cdot kg^{-1}$]

ρ_{bd} is the bulk density of the waste [$kg \cdot m^{-3}$]

t_{fire} is the duration of the fire [h]

Consistent with [II.4], it is cautiously assumed that the associated plume is neutrally buoyant. Using an approach consistent with [II.3], the air concentration of a radionuclide at ground level, $C_{air,fire}$ [$Bq \cdot m^{-3}$], at a given distance for prevalent atmospheric conditions is given by:

$$C_{air,fire} = R_{fire} \cdot C_{integ,fire}$$

where

R_{fire} is the release rate of the radionuclide from the fire [$Bq \cdot h^{-1}$]

$C_{int\,eg,fire}$ is the time-integrated air concentration at ground level at the given distance for the prevalent atmospheric conditions [$Bq \cdot h \cdot m^{-3}\ Bq^{-1}$]

The above equation cautiously assumes that the plume is non-depleting during passage towards the exposed individual.

Activity may be deposited on the ground by dry and wet deposition. The surface concentration of a radionuclide, $C_{surf,fire}$ [$Bq \cdot m^{-2}$], resulting from dry and wet deposition is given by:

$$C_{surf,fire} = C_{air,fire} \cdot t_{dep,fire} \cdot \left(V_{g,fire} + W_{out,fire} \cdot h_{fire} \right)$$

where

$C_{air,fire}$ is the air concentration of a radionuclide at ground level [$Bq \cdot m^{-3}$]

$t_{dep,fire}$ is the time over which deposition occurs [s]

$V_{g,fire}$ is the dry deposition velocity [$m \cdot s^{-1}$]

$W_{out,fire}$ is the washout coefficient [s^{-1}]

h_{fire} is the plume height [m]

II.6.2. Dose assessment

II.6.2.1. Worker

The dose to a worker fighting the fire can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{sub} + Dose_{inh}$$

where

$Dose_{sub}$ and $Dose_{inh}$ are the doses due to the external irradiation from submersion in the plume released by the fire, and the dose from inhalation of particles in the plume [$Sv \cdot y^{-1}$].

$$Dose_{sub} = C_{fire} \cdot t_{out,fire} \cdot DF_{sub} \cdot Occ_{fire}$$

where

C_{fire} is the assumed concentration of a radionuclide in the plume [$Bq.m^{-3}$]

$t_{out,fire}$ is the time spent fighting the fire [h]

DF_{sub} is the dose factor for external irradiation from submersion in the plume [$Sv \cdot h^{-1}/Bq \cdot m^{-3}$]

Occ_{fire} is the number of fires per year [y^{-1}]

$$C_{fire} = A_m \cdot Dust_{fire}$$

where

A_m is the specific activity of the radionuclide in the trench [$Bq \cdot kg^{-1}$]

$Dust_{fire}$ is the dust level in the air from the fire release breathed by the worker [$kg \cdot m^{-3}$]

$$Dose_{inh} = C_{fire} \cdot t_{out,fire} \cdot b_{r,fire} \cdot DF_{inh} \cdot Occ_{fire}$$

where

C_{fire} is the assumed concentration of a radionuclide in the plume [$Bq \cdot m^{-3}$]

$t_{out,fire}$ is the time spent fighting the fire [h]

$b_{r,fire}$ is the breathing rate of the worker [$m^3 \cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

Occ_{fire} is the number of fires per year [y^{-1}]

II.6.2.2. Public

The dose to a person living at the site boundary can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{sub} + Dose_{inh} + Dose_{ext} + Dose_{ing}$$

where

$Dose_{sub}$ is the dose due to the external irradiation from submersion in the plume released by the fire [$Sv \cdot y^{-1}$]

$Dose_{inh}$ is the dose from inhalation of particles in the plume [$Sv \cdot y^{-1}$]

$Dose_{ext}$ is the dose from external irradiation from deposited activity both during and after the passing of the plume [$Sv \cdot y^{-1}$]

$Dose_{ing}$ is the dose from the ingestion of activity deposited on leafy green vegetables [$Sv \cdot y^{-1}$]

$$Dose_{sub} = C_{air,fire} \cdot t_{out,fire} \cdot DF_{sub} \cdot Occ_{fire}$$

where

$C_{air,fire}$ is the air concentration of a radionuclide at ground level [$Bq \cdot m^{-3}$]

$t_{out,fire}$ is the time spent outside during the fire [h]

DF_{sub} is the dose factor for external irradiation from submersion in the plume [$Sv \cdot h^{-1}/Bq \cdot m^{-3}$]

Occ_{fire} is the number of fires per year [y^{-1}]

$$Dose_{inh} = C_{air,fire} \cdot t_{out,fire} \cdot b_{r,fire} \cdot DF_{inh} \cdot Occ_{fire}$$

where

$C_{air,fire}$ is the air concentration of a radionuclide at ground level [$Bq \cdot m^{-3}$]

$t_{out,fire}$ is the time spent outside during the fire [h]

$b_{r,fire}$ is the breathing rate of the member of the public [$m^3 \cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

Occ_{fire} is the number of fires per year [y^{-1}]

$$Dose_{ext} = C_{surf,fire} \frac{\left(1 - e^{-\lambda t}\right)}{\lambda t} \cdot DF_{ext,surf}$$

where

$C_{surf,fire}$ is the surface concentration of a radionuclide from one release [$Bq \cdot m^{-2}$]

λ is the radionuclide decay constant [y^{-1}]

t is the exposure duration [y]

$DF_{ext,surf}$ is the external exposure dose factor [$Sv \cdot h^{-1}/Bq \cdot m^{-2}$]

$$Dose_{ing} = Q_{gveg}\, CF_{gveg}\, C_{surf,fire} \frac{\left(1 - e^{-\lambda_{gveg} t}\right)}{\lambda_{gveg}} DF_{ing} \cdot Occ_{fire}$$

where

Q_{gveg} is the green vegetable consumption rate [kg·d^{-1}]

CF_{gveg} is the initial activity concentration of a radionuclide in green vegetables for unit a real concentration on soil [Bq·kg^{-1}/Bq·m^{-2}]

$C_{surf,fire}$ is the concentration of a radionuclide deposited following the fire on the surface of the soil [Bq·m^{-2}]

λ_{gveg} is the effective rate constant for removal of activity from the green vegetable [d^{-1}]

t is the time following the release over which the green vegetables are consumed [d]

Df_{ing} is the dose factor for ingestion [Sv·Bq^{-1}]

Occ_{fire} is the number of fires per year [y^{-1}]

$$CF_{gveg} = \frac{Int_{gveg} \cdot Fract}{Yield_{gveg}}$$

where

Int_{gveg} is the effective interception factor [–]

Fract is the fraction retained after processing [–]

$Yield_{gveg}$ is the yield of fresh green vegetables [kg m^{-2}]

$$\lambda_{gveg} = \lambda + \lambda_{weath}$$

where: λ is the decay constant of the radionuclide [d^{-1}]

λ_{weath} is the weathering rate of the radionuclide from the green vegetable [d^{-1}]

Note that it is cautiously assumed that the wind direction at the time of each release is towards the exposed member of the public.

II.7. DIRECT CONTACT (SCE14A)

II.7.1. Source term modelling

It is supposed that the waste accidentally tipped on the ground constitutes a semi-infinite slab for the exposed worker.

The effective dose rate in the vicinity of a semi-infinite medium with uniform activity can be calculated as follows:

β-radiation

At any point within a source with an infinite volume of specific activity A_m, the energy absorbed per gram equals the energy emitted per gram.

The β dose rate in air in Gy·h^{-1} in this medium will then be [II.2]:

$$H = 5.76 \cdot 10^{-7} \cdot A_m \cdot E_i \cdot Y_i$$

where

A_m is the specific activity [Bq·g^{-1}]

E_i is the radiation energy [MeV]

Y_i is the intensity of radiation with energy E_i

At the surface of a semi-infinite volume source the dose rate will be half the above value. The dose equivalent rate for an activity of 1 Bq·g^{-1} is then:

$$D = 2.88 \cdot 10^{-7} \cdot A_m \cdot \sum (E_i \cdot Y_i \cdot K_i)$$

γ-radiation

For γ radiation, the dose factor for a large thick source is given by the following equation:

$$D_r = \sum \left(\left(k_i \left(Df_i \cdot \rho / 8760 \right) \right) \cdot \left(1 - e^{((\gamma/\rho)\cdot \rho \cdot e)} \right) \right)$$

where

k_i is the effective dose factor for energy E_i [Sv·Gy^{-1}]

Df_i is the annual dose factor for a large source and for energy E_i [Gy·y^{-1}·Bq^{-1}·cm^3]

ρ is the apparent soil density [g·cm^3]

(γ/e) is the mass attenuation coefficient [cm^2·g^{-1}]

e is the source thickness [cm]

The annual large source dose factors for different soil thickness are taken from reference [II.5]. In the case of a semi-infinite source F g inf = Dinf (inf).

II.7.2. Dose assessment

The dose to worker due to a direct contact with the waste [Sv] is given by:

$$Dose_{inh} = Dose_{ir} + Dose_{cr} + Dose_{ing}$$

II.7.2.1. External β, γ exposure at a distance from a thick source

The effects of dose [Sv] from β, γ emission by direct irradiation is given by:

$$Dose_{ir} = A_m \cdot (D) \cdot (t_e) \cdot (F_{\gamma \inf} + F_{\beta \inf})$$

where

A_m is the specific activity [Bg·g^{-1}]

D is the dilution coefficient [-]

t_e is the exposure time [h]

$F_{\gamma_{inf}}$ is the γ dose factors for exposure in the vicinity of a semi-infinite with a uniform contamination [Sv·h^{-1}·Bq^{-1}·g] [II.6, II.7]

$F_{\beta_{inf}}$ is the dose factors for exposure in the vicinity of a semi-infinite medium with a uniform contamination [Sv·h^{-1}·Bq^{-1}·g] [II.6, II.7]

II.7.2.2. $\beta\gamma$ exposure due to contact with waste and remnant contamination

The βγ skin exposure by contact with waste and remnant contamination leads to an effecting dose [Sv] given by:

$$Dose_{cr} = A_s \cdot D_g \cdot D \cdot t_e \cdot (F_{\beta c} + F_{\gamma c})$$

with: : $A_s = A_m \cdot e \cdot \frac{d_a}{10}$

where

A_m is the specific activity [Bq·g^{-1}]

e is the thickness of the deposit [mm]

d_a is the apparent density [g·cm^{-3}]m

As is the surface radioactivity (Bq·cm^{-2})

Where

D is the dilution coefficient [-]

D_g is the contamination dilution coefficient [-]

t_e is the time exposure [h]

$F_{\beta c}$ is the γ dose factor at the basal epidermal layer due to a 1 Bq·cm^{-2} surface skin contamination [Sv·h^{-1}·Bq^{-1}·cm^2] [II.8, II.9]

$F_{\gamma_c i\gamma}$ is the dose factor at the basal epidermal layer due to a 1 Bq·cm^{-2} surface skin contamination [Sv·h^{-1}·Bq^{-1}·cm^2] [II.9]

II.7.2.3. Ingestion through to surface contamination of the fingers

It is assumed that part of the surface activity of waste has been transferred to the fingers and then ingested by licking certain areas of skin on the fingers:

$$Dose_{ing} = A_s \cdot (D_g \cdot D) \cdot (n_i \cdot s_i) \cdot F_{ing}$$

where:

$$A_s = A_m \cdot e \cdot \frac{d_a}{10}$$

where

A_m is the specific activity [Bq·g^{-1}]

D_g is the contamination dilution coefficient [-]

D is the dilution factor [-]

n_i is the number of finger lickings per year [-]

s_i is the licked surface area [cm^2]

F_{ing} is the dose factor for 1 Bq activity ingestion [Sv·Bq^{-1}]

II.7.2.4. Inhalation in dust-laden environment

The effective dose from inhalation of contaminated dust [Sv] is given by:

$$Dose_{inh} = H_{inh} = D \cdot 10^{-3} \cdot C_a \cdot d_r \cdot t_e \cdot F_{inh} \cdot A_m$$

where

A_m is the specific activity [Bq·g^{-1}]

D is the dilution factor [-]

C_a is the dust concentration in air [mg·m^{-3}]

d_r is the breathing rate [m^3·h^{-1}]

t_e is the exposure time [h]

F_{inh} is the dose factor for inhalation [Sv·Bq^{-1}]

II.8. BATHTUBBING (SCE16A)

II.8.1. Source term modelling

When a "bath-tub" effect scenario is considered it is necessary to evaluate the concentration of radionuclides in the overflowing leachate, C_{disp} [Bq·m^{-3}]:

$$C_{disp}(t)=e^{-\lambda t}\frac{A_{mi}}{V_{dispunit}\left(\omega_{cd}+\rho_{bd}Kd_{d}\right)}$$

where

$e^{-\lambda t}$ is the radioactive decay before the scenario [-]

A_{mi} is the initial activity in the disposal unit [Bq]

$V_{dispunit}$ is the volume of the disposal unit [m^3]

ω_{cd} is the moisture content of the disposal unit [-]

ρ_{bd} is the dry bulk density in the disposal unit [$kg \cdot m^{-3}$]

Kd_d is the distribution coefficient in the disposal unit [$m^3 \cdot kg^{-1}$]

II.8.2. Dose assessment

The dose due to the "bath-tub" effect can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{ext} + Dose_{inh} + Dose_{ing}$$

where

$Dose_{ext}$, $Dose_{inh}$, and $Dose_{ing}$ are the doses due to the external exposure, the inhalation, and the ingestion pathways [$Sv \cdot y^{-1}$]:

$$Dose_{ext}=\frac{OF}{\rho_{soil}\cdot Th_{soil}}C_{disp}\left(sf\cdot t_{in}+t_{out}\right)\cdot DF_{ext}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

Sf is the shielding factor [-]

t_{in} is the time spent indoor [$h \cdot y^{-1}$]

t_{out} is the time spent outdoor [$h \cdot y^{-1}$]

DF_{ext} is the external exposure dose factor [$Sv \cdot h^{-1} \cdot Bq^{-1} \cdot kg$]

$$Dose_{inh}=\frac{OF}{\rho_{soil}\cdot Th_{soil}}C_{disp}\left[dust_{in}br,_{in}t_{in}+dust_{out}br,_{out}t_{out}\right]\cdot DF_{inh}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

$dust_{in}$, $dust_{ou}$ are the indoor and outdoor dust levels [$kg \cdot m^{-3}$]

$br_{,in}$, $br_{,out}$ are the indoor and outdoor breathing rates [$m^3 \cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

$$Dose_{ing} = \frac{OF}{\rho_{soil} \cdot Th_{soil}} C_{disp} \cdot \left[TF_{vegt} Q_{veget} + Q_{soil} \right] DF_{ing}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

TF_{veget} is the soil to plant concentration factor for the vegetable [$Bq \cdot kg^{-1}$ fresh weight / $Bq \cdot kg^{-1}$ dry soil]

Q_{veget} is the annual vegetable consumption [$kg \cdot y^{-1}$]

Q_{soil} is the inadvertent soil ingestion rate [$kg \cdot y^{-1}$]

DF_{ing} is the dose factor for ingestion [$Sv\ Bq^{-1}$]

II.9. LEACHATE (SCE1B)

II.9.1. Source term modelling

The equation governing the evolution of the residual activity A_r in a disposal unit is:

$$\frac{dA_r}{dt} = (\lambda + F \times ALF) A_r$$

where

A_r is the residual activity of the radionuclide in a disposal unity as a function of time [Bq]

λ is the radionuclide decay constant [y^{-1}]

ALF is the annual leaching rate of the radionuclide [y^{-1}]

F is the distribution function of failure probability of the disposal unit [-]

The annual leaching rate (ALF) is the ratio of the activity lost by leaching during the year t, A(t), over the total activity remaining that year, $A_r(t)$. It is expressed as:

$$ALF = \frac{Inf}{H_d\left(\varpi_{cd} + \rho_{bd} Kd_d\right)}$$

where

Inf is the annual infiltration rate [$m.y^{-1}$] accounting for the water budget and the disposal unit's hydraulic conductivity

H_d is the disposal unit height [m]

ω_{cd} is the kinematic porosity under saturated conditions or the moisture content under unsaturated conditions [-]

ρ_{bd} is the dry bulk density of the disposal unit [$kg.m^{-3}$]

Kd_d is the radionuclide distribution coefficient in the disposal unit [$m^3.kg^{-1}$]

By multiplying both parts of the ratio expressing ALF by the area of the disposal unit, the following expression can be derived:

$$ALF = \frac{Q}{V_{dispunit}\left(\varpi_{cd} + \rho_{bd} Kd_d\right)}$$

where

Q is the annual water flow through the disposal unit [$m^3.y^{-1}$]

$V_{dispunit}$ is the volume of the disposal unit [m^3]

F effectively represents the fraction of the waste, which will be subject to the leaching. It is often expressed as a Gaussian law:

$$F = \int_{-\infty}^{t} N(\tau)\, d\tau \text{ with } N(\tau) = \frac{1}{\sqrt{2\pi}\sigma} \exp\left[-\frac{(t-\overline{\tau})^2}{2\sigma^2}\right]$$

where

$\overline{\tau}$ and σ [y] are the average structure failure time and its standard deviation corresponding to the physical properties of the containment structure.

The radioactivity which leaves a disposal unit and enters the geosphere is:

$$A(t) = A_r(t) \times F \times ALF \left[Bq \cdot y^{-1}\right]$$

For the purposes of the current study, F has been set to unity.

An analytical solution can be presented for this particular case, as:

$$A_r(t) = A_{t=0} e^{-(\lambda + F \cdot ALF)t}$$

II.9.2. Dose assessment

In order to undertake dose calculations, it is necessary to consider transport of radionuclides through the unsaturated zone below the disposal facility (if present), and the saturated zone.

The unsaturated zone can be simply modelled as a delay (time buffer used by a decay function $e^{-\lambda . t}$); the time necessary for a contaminant to travel vertically through the zone is given by:

$$t_{unsat} = \frac{H_{unsat}}{V_{unsat}}$$

where

t_{unsat} is the travel time in the unsaturated zone [y]

H_{unsat} is the unsaturated zone thickness [m]

V_{unsat} is the contaminant velocity in the unsaturated zone [m.y^{-1}]

$$V_{unsat} = \frac{Inf}{\theta_u + \rho_{bg} Kd_g}$$

where

Inf is the annual infiltration rate [m.y^{-1}] accounting for the water budget and the disposal unit's hydraulic conductivity

θ_u is the moisture content of the unsaturated zone [-]

ρ_{bg} is the dry bulk density of the geosphere [kg.m^{-3}]

Kd_g is the distribution coefficient [m^3.kg^{-1}] in the geosphere

The general advection-dispersion equation governing the transport through the saturated zone is (in two dimensions):

$$\frac{D_x}{\omega_c} \frac{\partial^2 C}{\partial x^2} + \frac{D_y}{\omega_c} \frac{\partial^2 C}{\partial y^2} - \frac{V_d}{\omega_c} \frac{\partial C}{\partial x} = R \frac{\partial C}{\partial t} + R \cdot \lambda \cdot C$$

where

x is the groundwater flow axis

y is the transverse axis

C is the concentration of contaminant in the groundwater [kg.m^{-3}]

D_x and D_y are the dispersion tensors [m^2.y^{-1}] in the geosphere

$$D_x = \omega d + \alpha_L \mid V_d \mid \approx \alpha_L \mid V_d \mid$$

$$D_y = \omega d + \alpha_T \mid V_d \mid \approx \alpha_T \mid V_d \mid$$

ω is the total porosity of the geosphere [-]

d is the molecular diffusion coefficient in the geosphere [m^2.y^{-1}]

α_L and α_T is longitudinal and transversal dispersivities in the geosphere [m]

V_d is Darcy's velocity [m.y^{-1}]

$$V_d = -K \frac{\partial H}{\partial x}$$

V_d/ω_c is sometimes considered as a whole and then called "real" velocity or pore water velocity [m.y^{-1}]: it is the actual water velocity in the porous medium

K is hydraulic conductivity of the geosphere [m.y^{-1}]

$\frac{\partial H}{\partial x}$ is head gradient in the geosphere [-]

ω_c is the kinematic porosity of the geosphere [-]

R is the retardation factor in the geosphere [-]

$$R = 1 + \frac{\rho_{bg} Kd_g}{\omega}$$

ρ_{bg} is the dry bulk density of the geosphere [kg.m^{-3}]

Kd_g is the distribution coefficient [m^3.kg^{-1}] in the geosphere

λ is the radioactive decay constant [y^{-1}]

There are various ways to solve the transport equation, from simple analytical solutions, to complex 3D-modelling. The purpose of this publication is not to develop these approaches. However, some simplified analytical treatments can be found for instance in [II.10].

The dose to a member of the critical group for the leaching scenario can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{inh} + Dose_{ext} + Dose_{ing}$$

where

$Dose_{inh}$, $Dose_{ext}$, and $Dose_{ing}$ are the doses due to the inhalation, the external exposure, and the ingestion pathways [$Sv \cdot y^{-1}$].

The dose due to inhalation is expressed as:

$$D_{inh} = A_{soil} \cdot b_r \cdot 8766 \cdot \left[dust_{act} \%_{occup} + dust_{norm} \left(1 - \%_{occup}\right)\right] DF_{inh}$$

where

D_{inh} is the dose due to inhalation [$Sv.y^{-1}$]

A_{soil} is the concentration of the radionuclide in the soil [$Bq.kg^{-1}$ of soil]

b_r is the breathing rate [$m^3.h^{-1}$]

8766 are the hours in a year [$h.y^{-1}$]

$dust_{act}$ and $dust_{norm}$ are the dust concentrations during ploughing and non-ploughing activities [$kg.m^{-3}$]

$\%_{occup}$ is the occupancy factor for ploughing activities [-]

DF_{inh} is the dose factor for inhalation [$Sv.Bq^{-1}$]

The dose due to external exposure is expressed as:

$$D_{ext} = A_{soil} \cdot 8766 \cdot DF_{ext}$$

where

D_{ext} is the dose due to external exposure [$Sv.y^{-1}$]

A_{soil} is the concentration of the radionuclide in the soil [$Bq.kg^{-1}$ of soil]

8766 is the exposure duration [$h.y^{-1}$]

DF_{ext} is the external exposure dose factor [$Sv.h^{-1}.Bq^{-1}.kg$]

The dose due to ingestion is expressed as:

$$D_{ing} = D_{ing\_water} + D_{ing\_fish} + D_{ing\_crop} + D_{ing\_animal}$$

where

D_{ing} is the dose due to ingestion [$Sv.y^{-1}$]

D_{ing_water} is the dose due to water ingestion [$Sv.y^{-1}$]

$$D_{ing\_water} = Q_{water} \cdot C_{water} \cdot \frac{1}{1 + Kd_w \cdot part} \cdot DF_{ing}$$

Q_{water} is the intake rate of drinking water [$m^3.y^{-1}$]

C_{water} is the concentration of radionuclides in water [$Bq.m^{-3}$]

DF_{ing} is the dose factor for ingestion [$Sv.Bq^{-1}$]

Kd_w is the distribution coefficient for water/particles [$m^3.kg^{-1}$]

part is the suspended particle concentration [$kg.m^{-3}$] in the water (assumed to be zero for well water)

D_{ing_fish} is the dose due to freshwater fish consumption [$Sv.y^{-1}$] (when relevant to the scenario)

$$D_{ing\_fish} = Q_{fish} \cdot C_{water} \cdot TF_{fish} \cdot DF_{ing}$$

where

Q_{fish} is the consumption rate of freshwater fish [$kg.y^{-1}$]

TF_{fish} is the concentration ratio for fish [$m^3.kg^{-1}$]

D_{ing_veget} is the dose due to vegetable consumption [$Sv.y^{-1}$]

$$D_{ing\_crop} = \sum_{root,green,grain} \left\{ Q_{crop} \left[C_{water} \frac{Irrig \cdot Int}{Yield} A_{soil} TF_{crop} \right] DF_{ing} \right\}$$

where

Q_{crop} is the consumption rate of crop [$kg.y^{-1}$]

Irrig is the irrigation rate [$m.y^{-1}$]

Int is the interception factor [-]

Yield is the crop yield [$kg.m^{-2}.y^{-1}$]

TF_{crop} is the soil to plant concentration factor for the crop [Bq kg^{-1} fresh weight / Bq kg^{-1} dry soil]

D_{ing_animal} is the dose due to animal product consumption [$Sv.y^{-1}$]

$$D_{ing\_animal} = \sum_{beef,milk} \left\{ Q_{animal} \left[q_{water} C_{water} + q_{soil} A_{soil} + q_{pasture} A_{soil} TF_{pasture} \right] \times TF_{animal} DF_{ing} \right\}$$

where

Q_{animal} is the annual animal product consumption rate [$kg.y^{-1}$]

q_{water} is the daily water intake [$m^3.day^{-1}$]

q_{soil} is the daily soil intake [$kg.day^{-1}$]

$q_{pasture}$ is the daily pasture intake [$kg.day^{-1}$]

$TF_{pasture}$ is the soil to plant concentration factor for the pasture [Bq kg^{-1} fresh weight/Bq kg^{-1} dry soil]

TF_{animal} is the transfer coefficient to the animal product [$day.kg^{-1}$]

C_{water} is given by the geosphere calculations and the interface features (well extraction or discharge to a river)

A_{soil} is governed by a first order differential equation:

$$\frac{dA_{soil}}{dt} = \frac{Irrig}{\rho_{soil} \cdot Th_{soil}} C_{water} - \lambda_{eff} A_{soil}$$

where

$Irrig$ is the irrigation rate [$m.y^{-1}$]

ρ_{soil} is the soil dry bulk density [$kg.m^{-3}$]

Th_{soil} is the soil thickness [m]

λ_{eff} is an effective decay [y^{-1}]

$$\lambda_{eff} = \lambda + \frac{P_{eff}}{Th_{soil}\left(\omega_{soil} + \rho_{soil} Kd_s\right)} + \frac{TF_{plant} \cdot Yield_{plant}}{Th_{soil} \cdot \rho_{soil}} + \lambda_{erosion}$$

where

λ is the radionuclide decay constant [y^{-1}]

P_{eff} is the water infiltration rate through the soil [$m.y^{-1}$]

ω_{soil} is the soil kinematic porosity [-]

Kd_s is the radionuclide distribution coefficient in the soil [$m^3.kg^{-1}$]

TF_{plant} is the soil to plant concentration factor for the plant [Bq kg^{-1} fresh weight/Bq kg^{-1} dry soil]

$Yield_{plant}$ is the annual crop yield [$kg.m^{-2}.y^{-1}$]

$\lambda_{erosion}$ is the soil erosion rate [y^{-1}]

II.10. BATHTUBBING (SCE4B)

II.10.1. Source term modelling

When a "bath-tub" effect scenario is considered it is necessary to evaluate the concentration of radionuclides in the overflowing leachate, C_{disp} [$Bq \cdot m^{-3}$]:

$$C_{disp}(t) = e^{-\lambda t} \frac{A_{mi}}{V_{dispunit}(\omega_{ed} + \rho_{bd} Kd_d)}$$

where

$e^{-\lambda t}$ is the radioactive decay before the scenario [-]

A_{mi} is the initial activity in the disposal unit [Bq]

$V_{dispunit}$ is the volume of the disposal unit [m^3]

ω_{cd} is the moisture content of the disposal unit [-]

ρ_{bd} is the dry bulk density in the disposal unit [$kg \cdot m^{-3}$]

Kd_d is the radionuclide distribution coefficient in the disposal unit [$m^3 \cdot kg^{-1}$]

II.10.2. Dose assessment

The dose due to the "bath-tub" effect can be expressed as (in [$Sv \cdot y^{-1}$]):

$$Dose = Dose_{ext} + Dose_{inh} + Dose_{ing}$$

where

$Dose_{ext}$, $Dose_{inh}$, and $Dose_{ing}$ are the doses due to the external exposure, the inhalation, and the ingestion pathways [$Sv \cdot y^{-1}$];

$$Dose_{ext} = \frac{OF}{\rho_{soil} \cdot Th_{soil}} C_{disp}(sf \cdot t_{in} + t_{out}) \cdot DF_{ext}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density of the soil [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

sf is the shielding factor [-]

t_{in} is the time spent indoors [$h \cdot y^{-1}$]

t_{out} is the time spent outdoors [$h \cdot y^{-1}$]

DF_{ext} is the external exposure dose factor [$Sv \cdot h^{-1} \cdot Bq^{-1} \cdot kg$]

$$Dose_{inh} = \frac{OF}{\rho_{soil} \cdot Th_{soil}} C_{disp} \left[dust_{in} br,_{in} t_{in} + dust_{out} br,_{out} t_{out} \right] \cdot DF_{inh}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density of the soil [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

$dust_{in}$, $dust_{out}$ are the indoor and outdoor dust levels [$kg \cdot m^{-3}$]

$br,_{in}$, $br,_{out}$ are the indoor and outdoor breathing rates [$m^3 \cdot h^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

$$Dose_{ing} = \frac{OF}{\rho_{soil} \cdot Th_{soil}} C_{disp} \left[TF_{vegt} Q_{vegt} + Q_{soil} \right] DF_{ing}$$

where

OF is the water overflow to the garden during one year [m]

ρ_{soil} is the soil dry bulk density of the soil [$kg \cdot m^{-3}$]

Th_{soil} is the soil thickness [m]

C_{disp} is the concentration of radionuclides in overflowing leachate [$Bq \cdot m^{-3}$]

TF_{veget} is the soil to plant concentration factor for the vegetable [Bq kg^{-1} fresh weight/Bq kg^{-1} dry soil]

Q_{veget} is the vegetable consumption rate [$kg \cdot y^{-1}$]

Q_{soil} is the inadvertent soil ingestion rate [$kg \cdot y^{-1}$]

DF_{ing} is the dose factor for ingestion [Sv Bq^{-1}]

II.11. ON-SITE RESIDENCE, SOIL (SCE6B-SOIL)

II.11.1. Source term modelling

The activity to which the on-site resident is exposed, A_{res} [Bq.kg^{-1} of waste], is given by:

$$A_{res} = A_m \cdot e^{-\lambda t_1} \cdot dil$$

where

A_m is the initial concentration of the radionuclide disposed [Bq.kg^{-1} of waste]

λ is the radioactive decay constant [y^{-1}] (if required other mechanisms contributing to diminishing the radioactivity could also be incorporated in an effective decay term (λ_{eff}))

t_1 is the time before exposure starts [y]

dil is the dilution factor [-]

II.11.2. Dose assessment

The dose due to the residence scenario can be expressed as (in [Sv.y^{-1}]):

$$Dose = Dose_{ext} + Dose_{inh} + Dose_{ing}$$

where

$Dose_{ext}$, $Dose_{inh}$, and $Dose_{ing}$ are the doses due to the external exposure, the inhalation, and the ingestion pathways [Sv·y^{-1}];

$$Dose_{ext} = A_{res} \cdot (sf \cdot t_{in} + t_{out}) \cdot DF_{ext}$$

A_{res} is the activity to which the on-site resident is exposed [Bq.kg^{-1} of waste]

sf is the shielding factor [-]

t_{in} is the time spent indoors [h.y^{-1}]

t_{out} is the time spent outdoors [h.y^{-1}]

DF_{ext} is the external exposure dose factor [Sv·h^{-1}·Bq^{-1}·kg]

$$Dose_{inh} = A_{res} \cdot [dust_{in} b_{r,in} \cdot t_{in} + dust_{out} b_{r,out} \cdot t_{out}] \cdot DF_{inh}$$

A_{res} is the activity to which the on-site resident is exposed [Bq.kg^{-1} of waste]

$dust_{in}$ and $dust_{out}$ are the indoor and outdoor dust levels [kg.m^{-3}]

$b_{r,in}$ and $b_{r,out}$ are the indoor and outdoor breathing rates [m^3.h^{-1}]

t_{in} is the time spent indoors [h.y^{-1}]

t_{out} is the time spent outdoors [$h.y^{-1}$]

DF_{inh} is the dose factor for inhalation [$Sv \cdot Bq^{-1}$]

$$Dose_{ing} = A_{res}\left(TF_{veget} Q_{veget} + Q_{soil}\right) DF_{ing}$$

A_{res} is the activity to which the on-site resident is exposed [$Bq.kg^{-1}$ of waste]

TF_{veget} is the soil to plant concentration factor for the vegetable [Bq kg^{-1} fresh weight/Bq kg^{-1} dry soil]

Q_{veget} is the vegetable consumption rate [$kg.y^{-1}$]

Q_{soil} is the inadvertent soil ingestion rate [$kg.y^{-1}$]

DF_{ing} is the dose factor for ingestion [Sv Bq^{-1}]

II.12. ON-SITE RESIDENCE, RADON (SCE6B-RADON)

II.12.1. Source term modelling

For ^{222}Rn, the flux of gas into a house, F_{gas} [$Bq \cdot m^{-2} \cdot y^{-1}$], can be derived using the equations used in [II.1]:

$$F_{gas} = \lambda \cdot A \cdot \rho_{bd} \cdot \tau \cdot H_1 \cdot e^{\frac{-h_2}{H_2}} \cdot e^{-\lambda_{Ra226} \cdot t_1}$$

where

λ is the decay constant of ^{222}Rn [y^{-1}]

A is the initial ^{226}Ra concentration disposed [$Bq \cdot kg^{-1}$ of waste]

ρ_{bd} is the bulk density of the material in the disposal unit [$kg \cdot m^{-3}$]

τ is the emanation factor, defined as the fraction of the radon atoms produced which escape from the solid phase of the waste into the pore spaces [-]

H_1 is the effective diffusion relaxation length for the waste [m]

h_2 is the thickness of the cover between the waste and the base of the house [m]

H_2 is the effective relaxation length of the cover [m]

λ_{Ra226} is the ^{226}Ra decay constant [y^{-1}]

t_1 is the time before the exposure starts [y]

II.12.2. Dose assessment

Based on [II.1], the effective dose equivalent, D [$Sv.y^{-1}$], corresponding to the residence during one year on the site can be calculated from the following equation:

$$D = K_1 \cdot b_{r,in} \cdot t \cdot f \cdot K_2 \cdot C_{Rn222}$$

where

K_1 is the effective dose equivalent corresponding to an absorbed energy of 1 joule [$Sv.J^{-1}$]

$b_{r,in}$ is the indoor breathing rate [$m^3.h^{-1}$]

t_{in} is the time spent indoors [$h.y^{-1}$]

f is the equilibrium factor [-]

K_2 is the potential α-energy of ^{222}Rn in equilibrium with its daughters [$J.Bq^{-1}$]

C_{Rn222} is the concentration of ^{222}Rn in a house [$Bq \cdot m^{-3}$]

C_{Rn222} can then be derived using equations in [II.1]

$$C_{Rn222} = F_{gas} \cdot \frac{S}{V} \cdot \frac{1}{\lambda_{house}}$$

where

F_{gas} is the flux of ^{222}Rn into the house [$Bq \cdot m^{-2} \cdot y^{-1}$]

S is the surface area of the house [m^2]

V is the volume of the house [m^3]

λ_{house} is the air renewal rate in the house [y^{-1}]

II.13. ROAD CONSTRUCTION (SCE7B)

II.13.1. Source term modelling

The activity to which the intruder is exposed, A_{int} [$Bq.kg^{-1}$ of waste], is given by:

$$A_{\text{int}} = A_m \cdot e^{-\lambda t_1} \cdot dil$$

where

A_m is the initial concentration of the radionuclide disposed [$Bq.kg^{-1}$ of waste]

λ is the radioactive decay constant [y^{-1}] (if required other mechanisms contributing to diminishing the radioactivity could also be incorporated in an effective decay term (λ_{eff}))

t_1 is the time before intrusion starts [y]

dil is the dilution factor [-]

II.13.2. Dose assessment

The dose due the road construction scenario can be expressed as (in [$Sv.y^{-1}$]):

$$Dose = A_{\mathrm{int}} \cdot \left[Q_{soil} DF_{ing} + DF_{ext} + b_r \cdot dust \cdot DF_{inh} \right] t_2$$

where

A_{int} is the activity to which the intruder is exposed [$Bq.kg^{-1}$ of waste]

Q_{soil} is the inadvertent soil ingestion rate of the intruder [$kg.h^{-1}$]

DF_{ing} is the dose factor for ingestion [$Sv.Bq^{-1}$]

DF_{ext} is the external exposure dose factor [$Sv.h^{-1}.Bq^{-1}.kg$]

b_r is the breathing rate of the intruder [$m^3.h^{-1}$]

dust is the dust level experienced by the intruder [$kg.m^{-3}$]

DF_{inh} is the dose factor for inhalation [$Sv.Bq^{-1}$]

t_2 is the exposure duration [h]

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Appendix III

SOURCE TERM AND HUMAN EXPOSURE DATA

These data were used for the derivation of the example values within the particular cases considered in this publication and should not be further applied without appropriate care. Data are derived from various sources. Radionuclide and element dependent data are provided in Appendix IV.

III.1. GAS RELEASE (SCE1A AND SCE11A)

III.1.1. Source term

- fraction of ^{3}H activity associated with the gas = 0.039 [-] [III.1]
- fraction of ^{14}C activity associated with the gas = 0.2 [-] [III.1]
- average timescale of generation of ^{3}H gas = 20 y (trench), 100 y (vault) (representative values)
- average timescale of generation of ^{14}C gas = 20 y (trench), 100 y (vault) (representative values)
- emanation factor, defined as the fraction of the radon atoms produced which escape from the solid phase of the soil into the pore spaces = 0.1 (trench), 0.03 (vault) [-] (consistent with [III.2])
- effective diffusion relaxation length for the waste = 0.2 m (trench), 0.5 m (vault) (consistent with [III.2])
- the thickness of the cover = 1.0 m (trench), 0.3 m (vault) (Section 5.2.2 and Appendix I-1.2 and I-1.3)
- the effective relaxation length of the cover = 0.2 m (trench), 0.5 m (vault) (consistent with [III.2])
- the bulk density of the material in the disposal unit = 500 $kg{\cdot}m^{-3}$ (trench), 1533 $kg{\cdot}m^{-3}$ (vault) (A-1.2 and A-1.3)
- the width of the source perpendicular to the wind direction = 15 m (width of a single disposal unit, Appendix I-1)
- the mean wind speed = 4 $m{\cdot}s^{-1}$ (representative value)
- the height for vertical mixing = 2 m (approximate height of human)

III.1.2. Human exposure

- the time spent in the gas plume by the worker = 1760 h y^{-1} (8 hours per day for 220 days)
- the time spent in the gas plume by the public = 4383 h y^{-1} (50% of year)
- the breathing rate of the worker = 1.2 $m^{3}{\cdot}h^{-1}$
- the breathing rate of the public = 1.0 $m^{3}{\cdot}h^{-1}$

III.2. DROP AND CRUSH (SCE2A)

III.2.1. Source term

— A waste package is supposed to be dropped and crushed on the ground. The concrete and the biological shielding are suposed to be destroyed. The waste spread is supposed to represent a $1m^2$.

III.2.2. Human exposure

— Exposure in the cab operator:

distance between the cab and the waste = 7m (minimum distance for point source modelling)

air shield density = $1.22\ 10^{-3}\ g{\cdot}cm^{-3}$

radiation due to waste = perpendicular to floor

exposure time = 0.0167 h

— Exposure during the operator exit (ladder + walkway):

average distance between the operator and the waste = 10 m

air shield density = $1.22\ 10^{-3}\ g{\cdot}cm^{-3}$

air shield thickness = 10 m

radiation due to the waste = perpendicular to floor

exposure time during exit = 0.0167 h

— Exposure to the worker on the ground level:

average distance between the operator and the waste = 10 m

air shield density = $1.22\ 10^{-3}\ g{\cdot}cm^{-3}$

air shield thickness = 10 m

radiation due to the waste = parallel to floor

exposure time during exit = 0.0333 h

III.3. DIRECT IRRADIATION (SCE8A)

III.3.1. Source term

— Source term for tipped waste on trench:

thickness of tipped waste = 6 m

surface of apparent tipped waste = infinite

surface of covered tipped waste = 0, infinite

density of tipped waste (homogeneous) = $0.5\ g{\cdot}cm^{-3}$

source term = infinite slab

activity repartition on slab = homogeneous

III.3.2. Human exposure

— Exposure above the covered waste:

distance between worker and uncontaminated ground = 1 m

soil/fill material shield density = 1.5 $g \cdot cm^{-3}$

soil/fill material shield thickness = 1 m

the spent time on the covered waste = 1760 h y^{-1} (8 hour per day for 220 days)

— Exposure on the uncovered waste:

distance between worker and uncontaminated waste = 1 m

air shield density = 1.22 10^{-3} $g \cdot cm^{-3}$

air shield thickness = 1 m

the spent time on the uncovered waste = 20 h (around 5 minutes per day for 220 days)

III.4. LIQUID RELEASE (SCE9A)

III.4.1. Source term

All source term data are provided in Appendix I, except for radionuclide decay constants and trench distribution coefficients, which are provided in Tables D2 and D4, respectively.

III.4.2. Human exposure

Broadly consistent with [III.3].

— The biosphere system is composed of a small group of individuals (less than 50) relying on the local resources in order to constitute an agricultural community. Due to its characteristics, the average amount of water required to sustain such a system is $10^4 m^3 y^{-1}$.

— Human behaviour:

breathing rate = 1 $m^3 \cdot h^{-1}$

intake rate of drinking water = 0.73 $m^3 \cdot y^{-1}$

consumption rate of freshwater fish (clay geosphere disposal system) = 2 $kg \cdot y^{-1}$

consumption rate of grain = 148 $kg \cdot y^{-1}$

consumption rate of root vegetables = 235 $kg \cdot y^{-1}$

consumption rate of green vegetables = 62 $kg \cdot y^{-1}$

consumption rate of cow milk = 330 $kg \cdot y^{-1}$

consumption rate of cow meat = 95 $kg \cdot y^{-1}$.

dust concentration during ploughing activities = 10^{-6} $kg \cdot m^{-3}$

occupancy factor for ploughing activities = 0.034.

— Plants:

irrigation rate per crop (temperate) = 0.3 m y^{-1} (no irrigation of pasture)

irrigation rate per crop (arid) = 0.6 m y^{-1} (no irrigation of pasture)

soil thickness = 0.25 m

interception factor = 0.33

yield of grain = 0.4 $kg \cdot m^{-2} \cdot y^{-1}$ (wet weight)

yield of root vegetables = 3.5 $kg.m^{-2} \cdot y^{-1}$ (wet weight)

yield of green vegetables = 3 $kg \cdot m^{-2} \cdot y^{-1}$ (wet weight)

yield of pasture = 1.7 $kg \cdot m^{-2} \cdot y^{-1}$ (wet weight)

pasture is assumed to be contaminated by root uptake from soil previously used to grow irrigated crops.

— Cattle:

daily water consumption = 0.06 $m^3 \cdot day^{-1}$

daily soil consumption = 0.6 $kg \cdot day^{-1}$

daily pasture intake (wet) = 55 $kg.day^{-1}$

average body weight = 500 kg

average milk production = 5500 $kg \cdot y^{-1}$

cattle density on agricultural land = 100 $animals \cdot km^{-2}$.

— Soil consistent with Appendix I-3:

thickness = 0.25 m

water infiltration rate through soil (temperate) = 0.6 m y^{-1}

water infiltration rate through soil (arid) = 0.144 m y^{-1}

kinetic porosity = 0.3

dry bulk density = 1800 $kg \cdot m^{-3}$

erosion rate (temperate) = 2 10^{-4} $m \cdot y^{-1}$

erosion rate (arid) = 2 10^{-3} $m \cdot y^{-1}$

— Atmosphere:

dust concentration during non-ploughing activities = 2.10^{-8} $kg \cdot m^{-3}$.

III.5. SOLID RELEASE (SCE10A)

III.5.1. Source term

- the dust level in the air breathed by the worker = 1E-5 $kg \cdot m^{-3}$ (order of magnitude higher than the average quoted in [III.4])
- the release fraction for the radionuclide = 0.1 [-] [III.5]
- the volume of the waste from which the dust is released = 0.09 m^3 (assuming dust is suspended from a 1 mm thickness of waste with width (15 m) and depth (6 m) equal to those of a trench)
- the duration of the dust release = 1 h (representative value)
- the time-integrated air concentration at ground level at the given distance for the prevalent atmospheric conditions = 3.24 $Bq \cdot h \cdot m^{-3} \cdot Bq^{-1}$ (value given in [III.6] for 100 m from ground release under Pasquill stability category D (neutral))
- the time over which deposition occurs = 3600 s (same duration as the duration of the release)
- the dry deposition velocity = 0.002 $m \cdot s^{-1}$ [III.7]
- the washout coefficient = 3 10^{-4} (upper end of range given in [III.8])
- the plume height = 10 m [III.6] the number of dust releases per year = 1 y^{-1} (illustrative value)

III.5.2. Human exposure

III.5.2.1. Worker

- the time spent exposed to the dust = 1 h (same duration as the duration of the release)
- the breathing rate of the worker = 1.2 $m^3 \cdot h^{-1}$

III.5.2.2. Public

- the time spent outside during the passage of the dust plume = 1 h (same as the duration of the release)
- the breathing rate of the member of the public = 1 $m^3 \cdot h^{-1}$(consistent with SCE6B-Soil)
- the indoor shielding factor = 0.1[-] (consistent with SCE6B-Soil)
- the time spent indoors = 6575 $h \cdot y^{-1}$ (consistent with SCE6B-Soil)
- the time spent outdoors = 2192 $h \cdot y^{-1}$ (consistent with SCE6B-Soil)
- the exposure duration = 1 y (annual dose is being calculated)
- the effective interception factor for each radionuclide on green vegetables = 0.3 [-] [III.9]
- the fraction of each radionuclide retained after processing = 0.3 [-] (upper end of range given in [III.10])
- the green vegetable consumption rate = 0.26 $kg \cdot d^{-1}$(assume that 75% of annual consumption given for SCE6B-Soil for small kitchen garden is consumed during a 90 day period)

— the time following the dust release over which the green vegetables are consumed = 90 d (representative value)

— the yield of fresh green vegetables = 3.0 kg·m^{-2} (consistent with SCE6B-Soil)

III.6. FIRE (SCE12A)

III.6.1. Source term

— The dust level in air breathed by the worker = 1E-3 kg·m^{-3}

— The release fraction for the radionuclide from the fire [-] [III.4]:

1.0	H, C , I
0.5	Pb
0.1	Zn, Ru, Sb, Cs
0.01	Na, Ag
0.001	All other elements

— The volume of the waste consumed in the fire = 90 m^3 (15 m width, 6 m depth and 1 m thickness of waste affected by fire)

— The duration of the fire = 1 h (representative value)

— The time-integrated air concentration at ground level at the given distance for the prevalent atmospheric conditions = 3.24 Bq·h·m^{-3}·Bq^{-1} (value given in [III.6] for 100 m from ground release under Pasquill stability category D (neutral))

— The time over which deposition occurs = 3600 s (same duration as the duration of the release)

— The dry deposition velocity = 0.002 m·s^{-1} [III.7]

— the washout coefficient = 3 10^{-4} (upper end of range given in [III.8])

— The plume height = 10 m [III.6]

— The number of fires per year = 1 y^{-1} (illustrative value to allow doses to be expressed in Sv·y^{-1} rather than Sv per fire thus facilitating comparison of results from different scenarios)

III.6.2. Human exposure

III.6.2.1. Worker

— the time spent fighting the fire = 1 h (same duration as the duration of the release)

— the breathing rate of the worker = 1.2 m^3·h^{-1}

III.6.2.2. Public

— the time spent outside during the fire = 1 h (same duration as the duration of the release)

— the breathing rate of the member of the public = 1.0 m^3·h^{-1}(consistent with SCE6B-Soil)

— the indoor shielding factor = 0.1[-] (consistent with SCE6B-Soil)

— the time spent indoors = 6575 $h.y^{-1}$ (consistent with SCE6B-Soil)

— the time spent outdoors = 2192 $h.y^{-1}$ (consistent with SCE6B-Soil)

— the exposure duration = 1 y (annual dose is being calculated)

— the green vegetable consumption rate = 0.26 $kg{\cdot}d^{-1}$(assume that 75% of annual consumption given for SCE6B-Soil for small kitchen garden is consumed during a 90 day period)

— the time following the fire release over which the green vegetables are consumed = 90 d (representative value)

— the yield of fresh green vegetables = 3.0 $kg{\cdot}m^{-2}$ (consistent with SCE6B-Soil)

III.7. DIRECT CONTACT (SCE14A)

III.7.1.External bg exposure at a distance from a thick source

III.7.1.1. Source term

— activity repartition = homogeneous

— dilution coefficient = 1

III.7.1.2. Human exposure

— contact time exposure = 0.5 h

III.7.2. βγ exposure due to contact with waste and remanent contamination

III.7.2.1. Source term

— activity repartition = homogeneous

— thickness of the deposit = 0.1 mm [III.4]

— apparent density = 1.4 $g.cm^{-3}$ [III.4]

— dilution coefficient = 1

— contamination dilution coefficient = 0.1 [III.4]

III.7.2.2. Human exposure

— remanent time exposure = 8 h

III.7.3. Ingestion through to surface contamination of the fingers

III.7.3.1. Source term

— activity repartition = homogeneous

— thickness of the deposit = 0.1 mm [III.4]

— apparent density = 1.4 $g.cm^{-3}$ [III.4]

— dilution factor = 1
— contamination dilution coefficient = 0.1 [III.4]

III.7.3.2. Human exposure

— number of finger lickings = 1
— licked surface area = 2 cm^2

III.7.4. Inhalation in dust-lader environment

III.7.4.1. Source term

— specific activity of waste = 1 $Bq.g^{-1}$
— dilution factor = 1
— dust concentration in air = 1 $mg \cdot m^{-3}$

III.7.4.2. Human exposure

— breathing rate = 1.2 $m^3.h^{-1}$
— exposure time = 0.5 h

III.8. BATHTUBBING (SCE16A)

This scenario involves the over-flow of leachate accumulated in a trench disposal unit leading to contamination of soil just beyond the site boundary. This scenario is limited to a trench disposal unit in the clay geosphere under temperate conditions.

III.8.1. Source term

— minimum time necessary to fill the trench disposal unit with water = 5 years (assuming failure of drains)

Other source term data are provided in Appendix I, except for radionuclide decay constants and trench distribution coefficients, which are provided in Tables D2 and D4, respectively.

III.8.2. Human exposure

— Water overflow to the garden = 0.1 $m \cdot y^{-1}$
— For external exposure, a shielding factor of 0.1 for indoor activities is assumed (consistent with SCE6B-Soil).
— Human behaviour (consistent with SCE6B-Soil):

breathing rate indoor = 0.75 $m^3 \cdot h^{-1}$

breathing rate outdoor = 1 $m^3 \cdot h^{-1}$

time spent indoor = 6575 $h \cdot y^{-1}$

time spent outdoor = 2191 $h \cdot y^{-1}$

consumption rate of root vegetables = 118 $kg \cdot y^{-1}$

consumption rate of green vegetables = 31 $kg \cdot y^{-1}$

inadvertent soil ingestion rate = 3 10^{-2} $kg \cdot y^{-1}$.

— Soil (consistent with Appendix I-3):

thickness = 0.25 m

dry bulk density = 1800 $kg \cdot m^{-3}$.

— Atmosphere (consistent with SCE6B-Soil):

indoor dust level = 1 10^{-8} $kg \cdot m^{-3}$

outdoor dust level = 2 10^{-8} $kg \cdot m^{-3}$.

III.9. LEACHATE (SCE1B)

III.9.1. Source term

All source term data are provided in Appendix I, except for radionuclide decay constants and disposal facility distribution coefficients that are provided in Tables D2 and D4, respectively.

III.9.2. Human exposure

— This system is composed of a small group of individuals (less than 50) relying on the local resources in order to constitute an agricultural community. Due to its characteristics, the average amount of water required to sustain such a system is 10^4 $m^3.y^{-1}$.

— Human behaviour (generally consistent with [III.3]):

average adult breathing rate = 1 $m^3.h^{-1}$

intake rate of drinking water = 0.73 $m^3.y^{-1}$

consumption rate of freshwater fish (clay geosphere disposal system) = 2 $kg.y^{-1}$

consumption rate of grain = 148 $kg.y^{-1}$

consumption rate of root vegetables = 235 $kg.y^{-1}$

consumption rate of green vegetables = 62 $kg.y^{-1}$

consumption rate of cow milk = 330 $kg.y^{-1}$

consumption rate of cow meat = 95 $kg.y^{-1}$

dust concentration during ploughing activities = 10^{-6} $kg.m^{-3}$

occupancy factor for ploughing activities = 0.034 .

— Plants (generally consistent with [III.3]):

irrigation rate per crop (temperate) = 0.3 $m.y^{-1}$ (no irrigation of pasture)

irrigation rate per crop (arid) = 0.6 $m.y^{-1}$ (no irrigation of pasture) interception factor = 0.33

yield of grain = 0.4 $kg.m^{-2}.y^{-1}$ (wet weight)

yield of root vegetables = 3.5 $kg.m^{-2}.y^{-1}$ (wet weight)

yield of green vegetables = 3 $kg.m^{-2}.y^{-1}$ (wet weight)

yield of pasture = 1.7 $kg.m^{-2}.y^{-1}$ (wet weight)

pasture is assumed to be contaminated by root uptake from soil previously used to grow irrigated crops.

— Cattle (generally consistent with [III.3]):

daily water consumption = 0.06 $m^3.day^{-1}$

daily soil consumption = 0.6 $kg.day^{-1}$

daily pasture intake (wet) = 55 $kg.day^{-1}$

average body weight = 500 kg

average milk production = 5500 $kg.y^{-1}$

cattle density on agricultural land = 100 $animals.km^{-2}$.

— Soil (consistent with Appendix I-3):

thickness = 0.25 m

water infiltration rate through soil (temperate) = 0.6 $m.y^{-1}$

water infiltration rate through soil (arid) = 0.144 $m.y^{-1}$

kinematic porosity = 0.3

dry bulk density = 1800 $kg.m^{-3}$

erosion rate (temperate) = $2\ 10^{-4}\ y^{-1}$

erosion rate (arid) = $2\ 10^{-3}\ y^{-1}$.

— Atmosphere (consistent with [III.3]):

dust concentration during non-ploughing activities = 2.10^{-8} $kg.m^{-3}$.

III.10. BATHTUBBING (SCE4B)

This scenario involves the over-flow of leachate accumulated in a vault disposal unit leading to the contamination of soil just beyond the site boundary. This scenario is limited to a vault disposal unit in the clay geosphere under temperate conditions.

III.10.1. Source term

Minimum time necessary to fill the vault disposal unit with water = 60 years.

Other source term data are provided in Appendix I, except for the radionuclide decay constants and vault distribution coefficients, which are provided in Tables D2 and D4, respectively.

III.10.2. Human exposure

— Water overflow to the garden during one year = 0.1 m
— For external exposure, a shielding factor of 0.1 for indoor activities is assumed.
— Human behaviour (consistent with SCE6B-Soil):

breathing rate indoor = 0.75 $m^3 \cdot h^{-1}$
breathing rate outdoor = 1 $m^3 \cdot h^{-1}$
time spent indoor = 6575 $h \cdot y^{-1}$
time spent outdoor = 2191 $h \cdot y^{-1}$
consumption rate of root vegetables = 118 $kg.y^{-1}$
consumption rate of green vegetables = 31 $kg.y^{-1}$
inadvertent soil ingestion rate = 3 10^{-2} $kg.y^{-1}$.

— Soil (consistent with Appendix I-3):

thickness = 0.25 m
dry bulk density = 1800 $kg.m^{-3}$.

— Atmosphere (consistent with SCE6B-Soil):

indoor dust level = 1 10^{-8} $kg.m^{-3}$
outdoor dust level = 2 10^{-8} $kg.m^{-3}$.

III.11. ON-SITE RESIDENCE, SOIL (SCE6B-SOIL)

III.11.1. Source term modelling

— since the foundations are assumed set at 3 m, this scenario concerns only the trench concept (the vault has a cap of 3 m – see Appendix I-1.3)
— it is assumed that the construction of building on the trench system can occur upon loss of institutional control (example times considered in this study are 30, 100 and 300 years – see Section 5.1.5)
— dilution factor is 0.3 (as for the road construction scenario (SCE7B)).

III.11.2. Dose assessment

— Broadly consistent with [III.11].
— For external exposure, a shielding factor of 0.1 for indoor activities is assumed.
— Human behaviour:

breathing rate indoor = 0.75 $m^3.h^{-1}$
breathing rate outdoor = 1 $m^3.h^{-1}$

time spent indoor = 6575 h y^{-1}

time spent outdoor = 2192 h y^{-1}

root vegetables consumption rate = 118 kg.y^{-1}

green vegetables consumption rate = 31 kg.y^{-1}

inadvertent soil ingestion rate = 3 10^{-2} kg.y^{-1}.

— Atmosphere:

indoor dust level = 1 10^{-8} kg.m^{-3}

outdoor dust level = 2 10^{-8} kg.m^{-3}.

III.12. ON-SITE RESIDENCE, RADON (SCE6B-RADON)

III.12.1. Source term modelling

- emanation factor, defined as the fraction of the radon atoms produced which escape from the solid phase of the soil into the pore spaces = 0.1 (trench), 0.03 (vault) [-] (consistent with [III.2])
- effective diffusion relaxation length for the waste = 0.2 m (consistent with [III.2])
- the thickness of the cover between the waste and the base of the house = 2.0 m (trench), 3.0 m (vault) (Appendix I-1.2 and I-1.3)
- the effective relaxation length of the cover = 0.2 m (consistent with [III.2])
- bulk density of material in disposal unit = 500 kg·m^{-3} (trench) 1533 kg·m^{-3} (vault) (consistent with Appendix I-1.2 and I-1.3)
- it is assumed that the construction of a house on a disposal unit can occur upon loss of institutional control (example times considered in this study are 30, 100 and 300 years – see Section 5.1.5)
- air renewal rate in the house = 8741 y^{-1}
- ratio of house surface area to house volume = 0.2 m^{-1}

III.12.2. Dose assessment

- broadly consistent with [III.2]
- indoor breathing rate = 0.75 m^3.h^{-1}
- time spent indoors = 6575 h.y^{-1}
- effective dose equivalent corresponding to an absorbed energy of 1 joule = 2 Sv.J^{-1}
- equilibrium factor = 0.8
- potential α-energy of ^{222}Rn in equilibrium with its daughters = 5.54 10^{-9} J.Bq^{-1}

III.13. ROAD CONSTRUCTION (SCE7B)

III.13.1. Source term modelling

— Minimum time before the construction of a road through the vault system = 500 y (it is assumed that prior to this time that the concrete of the vault will be intact and form an effective deterent to intrusion even if institutional control is lost.)

— For the trench, it is assumed that the construction of a road through the trench system can occur upon loss of institutional control (example times considered in this study are 30, 100 and 300 years – see Section 5.1.5)

— Dilution of the radioactive waste in the non-radioactive materials (cover, engineered features, surrounding soil) ; for generic calculations, one could assume an average specific activity of all material as the ratio total activity / total mass of materials:

dilution factor for trench, 3 m depth = 0.3

no contamination for a 3 m depth through the vault system

dilution factor for trench, 6 m depth = 0.7

dilution factor for vault, 6 m depth = 0.5

dilution factor for trench, 9 m depth = 0.7 (same than for 6 m)

dilution factor for vault, 9 m depth = 0.6

— For the treatment of hot spot calculations (i.e. casual encounters with high concentration material), dilution factor = 1.0.

III.13.2. Dose assessment

— Broadly consistent with [III.11].

— Exposure duration:

average work speed = 10 km in 6 months (20 km per year)

maximum time to cross the facility = 25 to 250 hours according to road direction

work time = 8h / day during 1 month of 20 days

exposure duration = 88 h (200 m of radioactive material require 0.01 y = 88 h at the average speed defined above).

— For the treatment of hot spot calculations (i.e. casual encounters with high concentration material), exposure duration = 4 h

— Breathing rate of the intruder = 1.2 $m^3\ h^{-1}$

— Inadvertent soil ingestion rate of the intruder = $3.4\ 10^{-5}$ kg h^{-1}

— Dust level experienced by the intruder = $1\ 10^{-6}$ $kg.m^{-3}$.

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Appendix IV

RADIONUCLIDE AND ELEMENT DEPENDENT DATA

The following data have been used for the derivation of the illustrative activity limits for the particular cases considered in Section 5 of this publication and should not be further applied without appropriate care.

TABLE IV.1. RADIONUCLIDES AND DECAY CHAINS CONSIDERED IN THIS STUDY

Parent	Daughters (1)
^{3}H	
^{10}Be	
^{14}C	
^{22}Na	
^{41}Ca	
^{54}Mn	
^{55}Fe	
^{59}Ni	
^{63}Ni	
^{60}Co	
^{65}Zn	
^{90}Sr	
^{93}Zr	^{93m}Nb
^{94}Nb	
^{99}Tc	
^{106}Ru	
^{110m}Ag	
^{121m}Sn	
^{125}Sb	^{125m}Te (2)
^{126}Sn	
^{129}I	
^{134}Cs	
^{137}Cs	
^{144}Ce	
^{147}Pm	^{147}Sm
^{151}Sm	
^{152}Eu	
^{154}Eu	
^{204}Tl	
^{210}Pb	^{210}Po
^{226}Ra (3)	^{210}Pb $\rightarrow$ ^{210}Po
^{227}Ac	
^{228}Ra	^{228}Th
^{232}Th	^{228}Ra $\rightarrow$ ^{228}Th
^{234}U	^{230}Th $\rightarrow$ ^{226}Ra $\rightarrow$ ^{210}Pb $\rightarrow$ ^{210}Po
^{235}U	^{231}Pa $\rightarrow$ ^{227}Ac
^{238}U	^{234}U $\rightarrow$ ^{230}Th $\rightarrow$ ^{226}Ra $\rightarrow$ ^{210}Pb $\rightarrow$ ^{210}Po
^{237}Np	^{233}Pa $\rightarrow$ ^{233}U $\rightarrow$ ^{229}Th
^{238}Pu	^{234}U $\rightarrow$ ^{230}Th $\rightarrow$ ^{226}Ra $\rightarrow$ ^{210}Pb $\rightarrow$ ^{210}Po
^{239}Pu	^{235}U $\rightarrow$ ^{231}Pa $\rightarrow$ ^{227}Ac
^{240}Pu	^{236}U $\rightarrow$ ^{232}Th $\rightarrow$ ^{228}Ra $\rightarrow$ ^{228}Th
^{241}Pu	^{241}Am $\rightarrow$ ^{237}Np $\rightarrow$ ^{233}Pa $\rightarrow$ ^{233}U $\rightarrow$ ^{229}Th
^{241}Am	^{237}Np $\rightarrow$ ^{233}Pa $\rightarrow$ ^{233}U $\rightarrow$ ^{229}Th

Notes:

(1) Decay chains have been simplified to include only daughters with a half-life greater than 25 days. The radiation effects of other, shorter lived, daughters have been included with those of the immediate parent, assuming secular equilibrium within each medium.

(2) Branching ratio of 2.28E-01.

(3) For the calculation of radon impacts for the on-site residence scenario the decay of ^{226}Ra to ^{222}Rn is considered.

TABLE IV.2. RADIONUCLIDE HALF-LIVES AND DECAY CONSTANTS

Radionuclide	Half Life (y) (1)	Decay Constant (y^{-1}) (2)
^{3}H	1.24E+01	5.59E-02
^{10}Be	1.60E+06	4.33E-07
^{14}C	5.73E+03	1.21E-04
^{22}Na	2.60E+00	2.66E-01
^{41}Ca	1.40E+05	4.95E-06
^{54}Mn	8.56E-01	8.10E-01
^{55}Fe	2.70E+00	2.57E-01
^{59}Ni	7.54E+04	9.19E-06
^{63}Ni	9.60E+01	7.22E-03
^{60}Co	5.27E+00	1.32E-01
^{65}Zn	6.68E-01	1.04E+00
^{90}Sr	2.91E+01	2.38E-02
^{93}Zr	1.53E+06	4.53E-07
^{93m}Nb	1.36E+01	5.10E-02
^{94}Nb	2.03E+04	3.41E-05
^{99}Tc	2.13E+05	3.25E-06
^{106}Ru	1.01E+00	6.86E-01
^{110m}Ag	6.84E-01	1.01E+00
^{121m}Sn	5.50E+01	1.26E-02
^{125}Sb	2.77E+00	2.50E-01
^{125m}Te	1.59E-01	9.94E-01
^{126}Sn	1.00E+05	6.93E-06
^{129}I	1.57E+07	4.41E-08
^{134}Cs	2.06E+00	3.36E-01
^{137}Cs	3.00E+01	2.31E-02
^{144}Ce	7.79E-01	8.90E-01
^{147}Pm	2.62E+00	2.65E-01
^{147}Sm	1.06E+11	6.54E-12
^{151}Sm	9.00E+01	7.70E-03
^{152}Eu	1.33E+01	5.21E-02
^{154}Eu	8.80E+00	7.88E-02
^{204}Tl	3.78E+00	1.83E-01
^{210}Pb	2.23E+01	3.11E-02
^{210}Po	3.79E-01	1.83E+00
^{222}Rn	1.05E-02	6.60E+01
^{226}Ra	1.60E+03	4.33E-04
^{228}Ra	5.75E+00	1.21E-01
^{227}Ac	2.18E+01	3.18E-02
^{228}Th	1.91E+00	3.63E-01
^{229}Th	7.34E+03	9.44E-05
^{230}Th	7.70E+04	9.00E-06
^{232}Th	1.40E+10	4.95E-11
^{231}Pa	3.28E+04	2.11E-05
^{233}Pa	7.39E-02	9.38E+00
^{233}U	1.59E+05	4.36E-06
^{234}U	2.45E+05	2.83E-06
^{235}U	7.04E+08	9.85E-10
^{236}U	2.34E+07	2.96E-08
^{238}U	4.47E+09	1.55E-10
^{237}Np	2.14E+06	3.24E-07
^{238}Pu	8.77E+01	7.90E-03
^{239}Pu	2.41E+04	2.88E-05
^{240}Pu	6.54E+03	1.06E-04
^{241}Pu	1.44E+01	4.81E-02
^{241}Am	4.32E+02	1.60E-03

Notes:

(1) Half life data taken from [1]

(2) Decay constant = $\dfrac{\ln 2}{half\ life}$

REFERENCE TO TABLE IV.2

[1] ICRP, Radionuclide Transformations Energy and Intensity of Emissions. International Commission on Radiological Protection, ICRP Publication 38, Pergamon Press, Oxford and New York (1983).

TABLE IV.3. DOSE COEFFICIENTS FOR INGESTION, INHALATION AND EXTERNAL IRRADIATION

	Dose coefficients (adults) (1)				
Radionuclide	Ingestion (Sv Bq^{-1}) (2)	Inhalation (Sv Bq^{-1}) (2)	External irradiation from soil ($Sv{\cdot}h^{-1}{\cdot}Bq^{-1}{\cdot}kg$) (3)	External irradiation from plume ($Sv.h^{-1}.Bq^{-1}.m^3$) (4)	External irradiation from soil ($Sv.h^{-1}.Bq^{-1}.m^2$) (4)
^{3}H	1.8E-11	2.6E-10 (5)	0.0E+00	1.2E-15	0.0E+00
^{10}Be	1.1E-09	3.5E-08	2.5E-14	4.0E-14	1.5E-15
^{14}C	5.8E-10	5.8E-09 (5)	0.0E+00	8.1E-16	5.8E-17
^{22}Na	3.2E-09	1.3E-09	4.7E-10	7.8E-10	7.6E-12
^{41}Ca	1.9E-10	1.8E-10	3.3E-16	0.0E+00	0.0E+00
^{54}Mn	7.1E-10	1.5E-09	1.8E-10	1.5E-10	2.9E-12
^{55}Fe	3.3E-10	7.7E-10	1.6E-20	0.0E+00	0.0E+00
^{59}Ni	6.3E-11	4.4E-10	1.1E-19	0.0E+00	0.0E+00
^{63}Ni	1.5E-10	1.3E-09	0.0E+00	0.0E+00	0.0E+00
^{60}Co	3.5E-09	3.1E-08	5.5E-10	4.5E-10	8.5E-12
^{65}Zn	3.9E-09	2.2E-09	1.3E-10	1.0E-10	2.0E-12
^{90}Sr	3.1E-08	1.6E-07	2.1E-12	7.1E-13	2.0E-14
^{93}Zr	1.1E-09	2.5E-08	0.0E+00	0.0E+00	0.0E+00
^{93m}Nb	1.2E-10	1.8E-09	5.8E-15	1.6E-14	3.4E-15
^{94}Nb	1.7E-09	4.9E-08	3.4E-10	2.8E-10	5.5E-12
^{99}Tc	6.4E-10	1.3E-08	3.8E-17	5.8E-15	2.8E-16
^{106}Ru	7.0E-09	6.6E-08	4.8E-11	3.7E-11	7.6E-13
^{110m}Ag	2.8E-09	1.2E-08	5.9E-10	5.0E-10	9.7E-12
^{121m}Sn	5.6E-10	4.7E-09	1.5E-12	2.2E-13	1.8E-14
^{125}Sb	1.1E-09	1.2E-08	8.3E-11	7.3E-11	1.5E-12
^{125m}Te	8.7E-10	4.2E-09	4.7E-13	1.6E-12	1.3E-13
^{126}Sn	5.1E-09	2.8E-08	4.0E-10	3.5E-10	7.1E-12
^{129}I	1.1E-07	3.6E-08	1.7E-13	1.4E-12	9.3E-14
^{134}Cs	1.9E-08	2.0E-08	3.3E-10	2.7E-10	5.5E-12
^{137}Cs	1.3E-08	3.9E-08	1.2E-10	9.8E-11	2.0E-12
^{144}Ce	5.3E-09	5.3E-08	1.2E-11	1.0E-11	2.1E-13
^{147}Pm	2.6E-10	5.0E-09	3.4E-16	2.5E-15	1.2E-16
^{147}Sm	4.9E-08	9.6E-06	0.0E+00	0.0E+00	0.0E+00
^{151}Sm	9.8E-11	4.0E-09	1.3E-17	1.3E-16	1.8E-17
^{152}Eu	1.4E-09	4.2E-08	3.0E-10	2.0E-10	4.0E-12
^{154}Eu	2.0E-09	5.3E-08	2.6E-10	2.2E-10	4.3E-12
^{204}Tl	1.2E-09	3.9E-10	8.1E-14	2.0E-13	5.3E-15
^{210}Pb	6.9E-07	5.7E-06	2.5E-13	3.2E-13	1.3E-14
^{210}Po	1.2E-06	4.3E-06	1.9E-15	1.5E-15	3.0E-17
^{226}Ra	2.8E-07	9.5E-06 (5)	5.7E-10	3.2E-10	6.0E-12
^{228}Ra	6.9E-07	1.6E-05	1.6E-10	1.7E-10	3.3E-12
^{227}Ac	1.2E-06	5.7E-04	6.0E-11	6.7E-11	1.4E-12
^{228}Th	1.4E-07	4.4E-05	3.2E-10	2.9E-10	5.1E-12
^{229}Th	6.1E-07	2.6E-04	4.9E-11	5.4E-11	1.1E-12
^{230}Th	2.1E-07	1.0E-04	2.4E-14	6.3E-14	2.7E-15
^{232}Th	2.3E-07	1.1E-04	9.4E-15	3.1E-14	2.0E-15
^{231}Pa	7.1E-07	1.4E-04	6.1E-12	6.2E-12	1.5E-13
^{233}Pa	8.7E-10	3.9E-09	2.7E-11	3.4E-11	7.0E-13
^{233}U	5.1E-08	9.6E-06	3.4E-14	5.9E-14	2.6E-15
^{234}U	4.9E-08	9.4E-06	6.7E-15	2.7E-14	2.7E-15

^{235}U	4.7E-08	8.5E-06	1.9E-11	2.8E-11	6.0E-13
^{236}U	4.7E-08	8.7E-06	2.8E-15	1.8E-14	2.3E-15
^{238}U	4.8E-08	8.0E-06	6.3E-12	4.9E-12	1.1E-13
^{237}Np	1.1E-07	5.0E-05	9.0E-12	3.7E-12	1.0E-13
^{238}Pu	2.3E-07	1.1E-04	1.2E-15	1.8E-14	3.0E-15
^{239}Pu	2.5E-07	1.2E-04	4.6E-15	1.5E-14	1.3E-15
^{240}Pu	2.5E-07	1.2E-04	1.4E-15	1.7E-14	2.9E-15
^{241}Pu	4.8E-09	2.3E-06	3.0E-16	7.9E-16	1.9E-17
^{241}Am	2.0E-07	9.6E-05	6.4E-13	2.9E-12	9.9E-14

Notes:

(1) Values include effects of short-lived daughters not explicitly listed, assuming secular equilibrium at time of intake or exposure.
(2) All data are taken from [1].
(3) The external irradiation dose coefficient is for exposure to soil contaminated to an infinite depth. All data are taken from [2], except ^{41}Ca, ^{121m}Sn, and ^{147}Sm, which are taken from [3] and ^{125m}Te which is taken from [4].
(4) All data are taken from [4].
(5) Dose coefficients for inhalation of gases:
— ^{3}H (methane) 1.8E-13 $Sv{\cdot}Bq^{-1}$ [1]
— ^{14}C (carbon dioxide) 6.2E-12 $Sv{\cdot}Bq^{-1}$ [1]
— ^{222}Rn (worker) 2.9E-9 $Sv{\cdot}Bq^{-1}$ [5] (assumes breathing rate is 1.2 $m^3{\cdot}h^{-1}$)
— ^{222}Rn (public) 2.4E-9 $Sv{\cdot}Bq^{-1}$ [5] (assumes breathing rate is 1.0 $m^3{\cdot}h^{-1}$)

REFERENCES TO TABLE IV.3

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, FOOD AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY, INTERNATIONAL LABOUR ORGANISATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, IAEA, Vienna (1996).

[2] ASSELLINEAU, J.M., GUÉTAT, P., RENAUD, P., BAEKELANDT, L., SKA, B., Proposed Activity Levels for Waste Disposal in Regulated Landfills in France and Belgium, European Commission Report EUR 15483EN, European Commission, Luxembourg (1995).

[3] ASHTON J., SUMERLING T.J., Biosphere database for assessments of radioactive waste disposals, UK DoE Report DOE/RW/88.083, Department of the Environment, London (1988).

[4] ECKERMAN, K.F., and RYMAN, J.C., External Exposure to Radionuclides in Air, Water and Soil. Exposure-to-Dose Coefficients for General Application, Based on the 1987 Federation Protection Guidance. Federal Guidance Report No. 12, , EPA 402-R-93-081 United States Environmental Protection Agency, Washington DC (1993).

[5] ICRP, Protection Against Radon-222 at Home and Work, ICRP Publication 65, Pergamon Press, Oxford and New York (1993).

TABLE IV.4. DISPOSAL FACILITY DISTRIBUTION COEFFICIENTS (m^3 kg^{-1})

Element	Trench with sand backfill (1)	Trench with clay backfill (1)	Vault (2)
H	1.0E-4	1.0E-4	0E+0
Be	2.4E-1	1.3E+0	1E-1 (3)
C	1.0E-1	1.0E-1	2E+0
Na	5.5E-2	2.7E-1	1E-3 (3)
Ca	9.0E-3	4.9E-1	2E+0
Mn	4.9E-2	1.8E-1	1E-1 (3)
Fe	2.2E-1	1.6E-1	1E-1
Ni	4.0E-1	6.7E-1	1E-1
Co	6.0E-2	5.4E-1	1E-1
Zn	2.0E-1	2.4E+0	1E-3 (3)
Sr	1.3E-2	1.1E-1	5E-3
Zr	6.0E-1	3.3E+0	2E+0
Nb	1.6E-1	9.0E-1	1E+0
Tc	1.4E-4	1.2E-3	2E+0
Ru	5.5E-2	4.0E-1	1E-1 (3)
Ag	9.0E-2	1.8E-1	1E-3
Sn	1.3E-1	6.7E-1	1E+0
Sb	4.5E-2	2.4E-1	1E+0 (3)
Te	1.5E-1	7.4E-1	1E-4 (3)
I	1.0E-3	1.8E-1	1E-3
Cs	2.7E-1	1.8E+0	1E-3
Ce	4.9E-1	2.0E+1	2E+0 (3)
Pm	2.4E-1	1.3E+0	2E+0 (3)
Sm	2.4E-1	1.3E+0	2E+0
Eu	2.4E-1	1.3E+0	2E+0 (3)
Tl	2.7E-1	5.4E-1	2E+0 (3)
Pb	2.7E-1	5.4E-1	2E+0
Po	1.5E-1	2.7E+0	2E+0
Ra	4.9E-1	9.0E+0	2E-1
Ac	4.5E-1	2.4E+0	2E+0
Th	3.0E+0	5.4E+0	2E+0
Pa	5.4E-1	2.7E+0	2E+0
U	3.3E-2	1.5E+0	2E+0
Np	4.1E-3	5.5E-2	5E+0
Pu	5.4E-1	4.9E+0	2E+0
Am	2.0E+0	8.1E+0	2E+0

Notes:

(1) Assumed to be applicable to the entire trench. Chemistry of the trench assumed to be the same as the surrounding soil and for the vault – alkaline environment of cementitious material. Data consistent with sand and clay values given in Table D5.
(2) Assumed to be applicable to the entire vault.
(3) In the absence of real data, data considered were based on [1], and [2].

REFERENCES TO TABLE IV.4

[1] EWART, F.T., PUGH, S.Y.R., WISBEY, S.J., WOODWARK, D.R., Chemical and Microbiological Effects in the Near Field: Current Status 1989, Nirex, Safety Series Report NSS/G111 (1989), Harwell, UK.

[2] ALLARD, B., HÖGLUND, L.O., SKAGIUS, K., Adsorption of Radionuclides on Concrete. Swedish Final Repository for Radioactive Waste, SKB Report SFR 91-02 (1991), Stockholm, Sweden.

[3] ALLARD, B., PERSSON, G., TORSTENFELT, B., Radionuclide Sorption on Concrete, Nagra Technical Report NTB 85-21 (1985), Wettingen, Switzerland.

TABLE IV.5. GEOSPHERE DISTRIBUTION COEFFICIENTS (m^3 kg^{-1})

Element (1)	**Sandy Geosphere** (2)	**Clay Geosphere** (2)
H	0.0E+0	0.0E+0
C	5.0E-3	1.0E-3
Ca	9.0E-3	4.9E-1
Fe	5.0E-3	8.0E-1
Ni	4.0E-1	6.0E-1
Co	1.5E-2	5.0E-1
Sr	1.5E-2	1.0E-1
Zr	5.0E-3	8.0E-1
Nb	3.4E-1	7.6E+0
Tc	1.0E-4	1.0E-3
I	1.0E-3	1.0E-3
Cs	3.0E-1	2.0E+0
Sm	1.1E+0	9.2E-2
Pb	3.0E-1	5.0E-1
Po	1.5E-1	3.0E+0
Ra	5.0E-1	9.0E+0
Ac	3.4E-1	7.6E+0
Th	3.0E+0	6.0E+0
Pa	3.4E-1	7.6E+0
U	5.6E-1	4.6E-2
Np	3.4E-1	7.6E+0
Pu	3.4E-1	7.6E+0
Am	3.4E-1	7.6E+0

Notes:

(1) Geosphere distribution coefficients are only requited for the post-closure leaching scenario (SCE1B). Given the more restricted list of radionuclides for post-closure assessment (Table 5.2), only a correspondingly limited number of elements need to be considered.

(2) Assumed to be applicable to the entire saturated and unsaturated geosphere. In the absence of real data, data considered were based on [1], and [2].

REFERENCES TO TABLE IV.5

[1] SMITH, G.M., FEARN, H.S., SMITH, K.R., DAVIS, J.P. and KLOS, R., Assessment of the Radiological Impact of Disposal of Solid Radioactive Waste at Drigg, Report NRPB-M148, National Radiological Protection Board, Chilton (1988).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Co-ordinated Programme on the Safety Assessment of Near Surface Radioactive Waste Disposal Facilities (NSARS) : Specification for Test Case 2C, IAEA, Vienna (1995).

TABLE IV.6.. BIOSPHERE DISTRIBUTION COEFFICIENTS ($m^3\ kg^{-1}$) (1)

Element	Sandy Soil	Clay Soil	Freshwater Sediment
H	1.0E-4	1.0E-4	3E-5
Be	2.4E-1	1.3E+0	2E-3
C	1.0E-1	1.0E-1	1E-1
Na	5.5E-2	2.7E-1	1E+0
Ca	9.0E-3	4.9E-1	1E+0
Mn	4.9E-2	1.8E-1	1E+0
Fe	2.2E-1	1.6E-1	5E+0
Ni	4.0E-1	6.7E-1	1E+1
Co	6.0E-2	5.4E-1	5E+0
Zn	2.0E-1	2.4E+0	5E-1
Sr	1.3E-2	1.1E-1	1E+0
Zr	6.0E-1	3.3E+0	1E+0
Nb	1.6E-1	9.0E-1	1E+1
Tc	1.4E-4	1.2E-3	5E-3
Ru	5.5E-2	4.0E-1	1E+0
Ag	9.0E-2	1.8E-1	2E+0
Sn	1.3E-1	6.7E-1	1E+1
Sb	4.5E-2	2.4E-1	1E+1
Te	1.5E-1	7.4E-1	3E+0
I	1.0E-3	1.8E-1	1E-2
Cs	2.7E-1	1.8E+0	1E+0
Ce	4.9E-1	2.0E+1	1E+1
Pm	2.4E-1	1.3E+0	5E+0
Sm	2.4E-1	1.3E+0	5E+0
Eu	2.4E-1	1.3E+0	5E-1
Tl	2.7E-1	5.4E-1	1E+1
Pb	2.7E-1	5.4E-1	1E+1
Po	1.5E-1	2.7E+0	1E+1
Ra	4.9E-1	9.0E+0	5E-1
Ac	4.5E-1	2.4E+0	1E+1
Th	3.0E+0	5.4E+0	1E+1
Pa	5.4E-1	2.7E+0	5E+0
U	3.3E-2	1.5E+0	5E-2
Np	4.1E-3	5.5E-2	1E-2
Pu	5.4E-1	4.9E+0	1E+2
Am	2.0E+0	8.1E+0	5E+0

Notes

(1) In the absence of real data, data considered were based on [1], and [2].

REFERENCES TO TABLE IV.6

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Temperate Environments, Technical Reports Series No. 364, IAEA, Vienna (1994).

[2] ASHTON, J., SUMERLING, T.J., Biosphere database for assessments of radioactive waste disposals (edition 1) UK DoE Report DOE/RW/88.083, Department of the Environment, London (1988).

TABLE IV.7. TRANSFER COEFFICIENTS TO COWS MEAT (days kg^{-1} fresh weight) AND MILK (days 1^{-1}) (1)

Element	Meat	Milk
H	2.9E-2	1.5E-2
Be	6.6E-4	2.6E-6
C	1.2E-1	1.0E-2
Na	8.0E-2	1.6E-2
Ca	2.0E-3	3.0E-3
Mn	5.0E-4	3.0E-5
Fe	2.0E-2	3.0E-5
Ni	5.0E-3	1.6E-2
Co	1.0E-2	3.0E-4
Zn	1.0E-1	1.0E-2
Sr	8.0E-3	2.8E-3
Zr	1.0E-6	5.5E-7
Nb	3.0E-7	4.1E-7
Tc	1.0E-4	2.3E-5
Ru	5.0E-2	3.3E-6
Ag	3.0E-3	5.0E-5
Sn	1.9E-3	1.0E-3
Sb	4.0E-5	2.5E-5
Te	7.0E-3	4.5E-4
I	4.0E-2	1.0E-2
Cs	5.0E-2	7.9E-3
Ce	2.0E-5	3.0E-5
Pm	5.0E-3	2.0E-5
Sm	5.1E-4	2.0E-5
Eu	4.7E-4	5.0E-5
Tl	4.0E-4 (2)	3.0E-4 (2)
Pb	4.0E-4	3.0E-4
Po	5.0E-3	3.4E-4
Ra	9.0E-4	1.3E-3
Ac	1.6E-4	4.0E-7
Th	2.7E-3	5.0E-6
Pa	5.0E-5	5.0E-6
U	3.0E-4	4.0E-4
Np	1.0E-3	5.0E-6
Pu	1.0E-5	1.1E-6
Am	4.0E-5	1.5E-6

Notes:

(1) Data compiled from a range of compilations including [1] and [2].
(2) Value assumed to be the same as for Pb.

REFERENCES TO TABLE IV.7

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Temperate Environments, Technical Reports Series No. 364, IAEA, Vienna (1994).

[2] ASHTON, J. and SUMERLING, T.J., Biosphere database for assessments of radioactive waste disposals UK DoE Report DOE/RW/88.083, Department of the Environment, London (1988).

TABLE IV.8. SOIL TO PLANT CONCENTRATION FACTORS (Bq kg^{-1} fresh weight/Bq kg^{-1} dry soil) FOR CROPS (1)

Element	Root Vegetables	Green Vegetables	Grain	Pasture
H	5E+0	5E+0	5E+0	5E+0
Be	1E-3	2E-3	2E-3	2E-3
C	1E-1	1E-1	1E-1	1E-1
Na	3E-2 (2)	3E-2 (2)	2E-2 (2)	3E-2 (2)
Ca	5E-1	5E-1	5E-1	5E-1
Mn	5E-1	5E-1	5E-1	3E-1
Fe	3E-4	2E-4	4E-4	4E-4
Ni	3E-2	3E-2	5E-2	2E-2
Co	3E-2	3E-2	3E-2	6E-3
Zn	4E-1	4E-1	4E-1	4E-1
Sr	9E-2	3E+0	8E-2	3E+0
Zr	5E-3	5E-3	5E-3	5E-3
Nb	1E-2	1E-2	1E-2	1E-2
Tc	1E+1	1E+1	1E+1	1E+1
Ru	1E-2	4E-3	4E-2	4E-2
Ag	2E-1	2E-1	2E-1	2E-1
Sn	1E-1	1E-1	2E-1	2E-1
Sb	1E-2	1E-2	1E-2	1E-2
Te	1E+0 (2)	1E+0 (2)	1E+0 (2)	1E+0 (2)
I	1E-1	1E-1	1E-1	1E-1
Cs	3E-2	3E-2	2E-2	3E-2
Ce	3E-3 (2)	3E-3 (2)	3E-3 (2)	3E-3 (2)
Pm	3E-3	3E-3	3E-3	3E-3
Sm	2E-3	2E-3	2E-3	2E-3
Eu	3E-3	3E-3	3E-3	3E-3
Tl	1E-2 (2)	1E-2 (2)	1E-2 (2)	1E-2 (2)
Pb	1E-2	1E-2	1E-2	1E-2
Po	2E-4	2E-4	2E-4	2E-4
Ra	4E-2	4E-2	4E-2	4E-2
Ac	1E-3	1E-3	1E-3	1E-3
Th	5E-4	5E-4	5E-4	5E-4
Pa	4E-2	4E-2	4E-2	4E-2
U	1E-3	1E-3	1E-4	1E-3
Np	1E-3	1E-2	3E-4	5E-3
Pu	1E-3	1E-4	3E-5	1E-3
Am	1E-3	1E-3	1E-5	5E-3

Notes:

(1) Data compiled from a range of compilations including [1] to [3].
(2) In the absence of real data, data considered were based on [1], [2] and [3].

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TABLE IV.9. CONCENTRATION RATIOS ($m^3\ kg^{-1}$) FOR FRESHWATER FISH (1)

Element	Concentration Ratio
H	1E-3
Be	1E-1
C	5E+1
Na	2E-2
Ca	2E-1
Mn	4E-1
Fe	2E-1
Ni	1E-1
Co	3E-1
Zn	1E+0
Sr	6E-2
Zr	3E-1
Nb	3E-1
Tc	2E-2
Ru	1E-2
Ag	5E-3
Sn	3E+0
Sb	1E-1
Te	4E-1
I	4E-2
Cs	2E+0
Ce	3E-2
Pm	3E-2
Sm	3E-1
Eu	5E-2
Tl	3E-1 (2)
Pb	3E-1
Po	5E-2
Ra	5E-2
Ac	3E-2 (2)
Th	1E-1
Pa	1E-2
U	1E-2
Np	3E-2
Pu	3E-2
Am	3E-2

Notes:

(1) Data compiled from [1] and [2].

(2) In the absence of real data, data considered were based on [1], and [2].

REFERENCES TO TABLE IV.9

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TABLE IV.10. WEATHERING RATES (d^{-1}) FOR GREEN VEGETABLES (1)

Element	Weathering rate
H	5.0E-2
Be	5.0E-2
C	5.0E-2
Na	5.0E-2
Ca	5.0E-2
Mn	5.0E-2
Fe	5.0E-2
Ni	5.0E-2
Co	5.0E-2
Zn	5.0E-2
Sr	5.0E-2
Zr	5.0E-2
Nb	5.0E-2
Tc	5.0E-2
Ru	5.0E-2
Ag	5.0E-2
Sn	5.0E-2
Sb	5.0E-2
Te	5.0E-2
I	5.0E-2
Cs	5.0E-2
Ce	5.0E-2
Pm	5.0E-2
Sm	5.0E-2
Eu	5.0E-2
Tl	5.0E-2
Pb	5.0E-2
Po	5.0E-2
Ra	5.0E-2
Ac	5.0E-2
Th	5.0E-2
Pa	5.0E-2
U	1.4E-1
Np	1.4E-1
Pu	1.4E-1
Am	5.0E-2

Notes:

(1) All data from [1].

REFERENCE TO TABLE IV.10

[1] SMITH, G.M., FEARN, H.S., SMITH, K.R., DAVIS, J.P., KLOS, R., Assessment of the Radiological Impact of Disposal of Solid Radioactive Waste at Drigg, NRPB-M148, National Radiological Protection Board, Chilton (1988).

Appendix V

COMPUTATIONAL TOOLS USED

V.1. OPERATIONAL PERIOD CALCULATIONS

Two sets of calculations were undertaken by two independent consultants:

— One consultant used the AMBER compartment modelling application [V.1] for all scenarios except the drop and crush scenario (SCE2A), the direct irradiation scenario (SCE8A) and the tipping accident scenario (SCE14A). Calculations were undertaken for the full set of radionuclides considered in the operational period calculations given in Table II of the report. For all scenarios modelled, other than the liquid release scenario (SCE9A), AMBER was used primarily as a spreadsheet to calculate doses (and associated disposal concentrations) for the relevant pathways using the appropriate expressions from Appendix II. For the liquid release scenario calculations, AMBER was used to simulate the migration of radionuclides from the disposal facility into the drainage system and thence into the biosphere by representing the system as a series of compartments linked by the transfer expressions given in Section II.4 of Appendix II. In each compartment instantaneous and uniform mixing was assumed. AMBER was then used to calculate doses (and associated disposal amounts) for the relevant pathways using the expressions given in Section II.4 of Appendix II.

— The other consultant undertook calculations for the drop and crush scenario, the direct irradiation scenario and the tipping accident scenario. Calculations were undertaken for the full set of radionuclides considered in the operational period calculations given in Table II of the report. For the three scenarios modelled, a combination of spreadsheets and the Microshield software tool [V.2] were used to calculate doses (and associated disposal concentrations) for the relevant pathways using the appropriate expressions from Appendix II.

V.2. POST-CLOSURE PERIOD CALCULATIONS

Two sets of calculations were undertaken by two independent consultants:

— One consultant used the AMBER compartment modelling application [V.1] for all scenarios. Calculations were undertaken for the full set of radionuclides considered in the post-closure period calculations given in Table 5.2 of the main text. For all scenarios, other than the leaching scenario (SCE1B), AMBER was used primarily as a spreadsheet to calculate doses (and associated disposal concentrations) for the relevant pathways using the expressions given in Sections II.10 to II.13 of Appendix II. For the leaching scenario calculations, AMBER was used to simulate the migration of radionuclides from the disposal facility into the geosphere and thence into the biosphere by representing the system as a series of compartments linked by the transfer expressions given in Appendix II.9. In each compartment instantaneous and uniform mixing was assumed. The geosphere was sub-divided into several compartments and AMBER was used to provide an approximate solution to the 1D version of the 2D advection-dispersion equation given in Section II.9.2 of Appendix II. Advection was represented as an advective flux from the upstream to downstream compartment. Dispersion was represented by specifying two diffusive/dispersive exchanges between the upstream and downstream compartments –

one from the upstream to the downstream compartment, the other from the downstream to the upstream compartment. Allowance was made for transverse dispersion by allowing the width of each geosphere compartment to increase with distance from the disposal facility, thus the calculations could be considered to be quasi two dimensional. It was assumed that the concentration in the well in the sandy geosphere was derived from the leaching of waste across the entire disposal facility. AMBER was then used to calculate doses (and associated disposal amounts) for the relevant pathways using the expressions given in Section II.9. of Appendix II.

— The other consultant performed the calculations by implementing the models as they are described in Sections II.9. to II 13. of Appendix II in a variety of software tools. Calculations were undertaken for a limited set of radionuclides (^{3}H, ^{14}C, ^{90}Sr, ^{129}I, ^{137}Cs, ^{226} Ra, ^{238}U and ^{241}Pu). For the treatment of all scenarios except the leaching scenario, models and data were simply processed by using a spreadsheet, since the proposed mathematical equations expressing the dose assessment do not require heavy computing capabilities. For assessing the leaching scenario, two fortran-based computer codes were used; GEOS version 2 [V.3] for the source-term and geosphere modelling, and ABRICOT version 2 [V.4] for the biosphere calculations. ABRICOT is the straightforward implementation of the biosphere equations prescribed in Section II.9. of Appendix II. In the case of the geosphere modelling, solving the advection-dispersion equation required in GEOS the use of a semi-analytic numerical solution: GEOS is indeed a 2D horizontal model for the treatment of the well scenarios, where the source-term is discretized on a plane, and it is a 1D model for the treatment of the river scenarios, where the source-term is discretized along a single line. In both cases, the unsaturated zone is merely modelled as a time buffer.

TABLE V.1. TOTAL ACTIVITIES (Bq) FOR THE POST-CLOSURE LEACHING SCENARIO FOR THE VAULT DISPOSAL SYSTEM FOR THE TWO DIFFERENT MODELS USED

Radionuclide	Sand Temperate		Sand Arid		Clay Temperate		Clay Arid	
	Model A	Model B	Model A	Model B	Model A	Model B	Model A	Model B
^{3}H	1.E+13	2.E+15	3.E+13	6.E+15	1.E+20	1.E+20	1.E+20	1.E+20
^{14}C	7.E+13	3.E+14	3.E+14	8.E+14	1.E+20	1.E+20	1.E+20	1.E+20
^{90}Sr	5.E+16	1.E+20	2.E+17	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20
^{129}I	3.E+09	6.E+09	2.E+09	3.E+09	6.E+14	3.E+15	3.E+13	2.E+14
^{137}Cs	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20
^{226}Ra	2.E+12	3.E+15	1.E+14	3.E+15	1.E+20	1.E+20	1.E+20	1.E+20
^{238}U	1.E+11	1.E+12	6.E+10	2.E+12	2.E+16	2.E+17	8.E+14	2.E+16
^{241}Pu	8.E+16	1.E+20	1.E+17	1.E+20	1.E+20	1.E+20	1.E+20	1.E+20

Note:

(1) Values calculated using a dose limit of 1 $mSv.y^{-1}$ for each radionuclide.

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