Final Report **Project UKRSR03**

Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: Case Study

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Research contractor
This document was produced by:

Galson Sciences
5 Grosvenor House
Melton Road
Oakham
Rutland LE15 6AX, UK

SNIFFER's project manager
SNIFFER's project manager for this contract is:

Karl Littlewood, Scottish Environment Protection Agency (SEPA)

SNIFFER'S research manager
SNIFFER's research manager for this contract is:

Fiona Mactaggart, SNIFFER

SNIFFER's project steering group members are:

John Titley, Environment Agency
Jim Gemmill, SEPA
Ian Hall, Scottish Executive
Robert Larmour, Environment and Heritage Service
Fiona Mactaggart, SNIFFER

SNIFFER
First Floor, Greenside House
25 Greenside Place
EDINBURGH EH1 3AA

www.sniffer.org.uk

EXECUTIVE SUMMARY

This report forms part of a research project aimed at establishing a framework for assessing the suitability of controlled landfills to accept disposals of solid low-level radioactive waste. The disposal of radioactive waste alongside other wastes at landfill sites is a disposal route aimed at small users rather than at the nuclear industry, and it is restricted to relatively low activity wastes.

The framework comprises the overall process for determining the suitability of landfill sites for accepting certain types of low-level radioactive waste. The framework comprises four principal stages:

- Initial screening for potentially suitable sites.
- Development of the assessment context and methodology.
- Calculation of specific doses and radiological capacity.
- Authorisation decision and conditions.

The framework is aimed at assessing new sites, or sites that have not previously accepted radioactive waste. For the purpose of this project, it has been assumed that all SPB disposals will be made to non-hazardous landfill sites. The framework therefore may not be applicable to inert and hazardous landfill sites.

Assessments of landfill sites in terms of their environmental impacts require the identification of the sources, pathways and receptors through which environmental harm could arise. A generic set of these that encompasses the activities and environmental setting of landfill sites has been identified and conceptual models have been developed. In addition to the generic elements of the assessment context, there are elements of the assessment context that must be established on a site-specific basis.

An Assessment Model implementing the framework for calculating specific doses and radiological capacities has been developed from mathematical models that describe the source-pathway-receptor linkages.

This report describes the results of a Case Study using the Assessment Model to calculate specific doses and to illustrate the derivation of radiological capacities. The generic information in the assessment framework has been supplemented by site-specific data for a typical landfill site (Site A).

In addition to calculations of specific doses for Site A based on reference values, this report includes results from a series of sensitivity studies aimed at building an understanding of system behaviour and highlighting areas that assessors must consider in deriving robust estimates of radiological capacity.

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1 Introduction

1. The Environment Agency for England and Wales (EA), the Scottish Environment Protection Agency (SEPA), and the Environment and Heritage Service, Northern Ireland (EHS) are responsible for the regulation of radioactive waste disposal in the UK. The Scotland and Northern Ireland Forum for Environmental Research (SNIFFER) has commissioned research on behalf of these UK regulatory agencies to establish a framework for assessing the suitability of controlled landfills to accept disposals of solid low-level radioactive waste from small users.

2. The key principles on which the framework are based are described in more detail in the Principles Document: Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: Principles Document (SNIFFER 2005). Details of the models that underpin the dose calculations and hence the calculation of the potential radiological capacity are presented in: Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: Technical Reference Manual (SNIFFER 2006a).

3. This document builds on the earlier reports and uses the assessment framework to conduct a case study for an illustrative landfill site. Details of this site, termed Site A, are made as realistic as possible by referring to data from descriptions and application documents for existing sites and UK and international standards.

1.1 Objective and Scope

4. The objective of this report is to illustrate use of the assessment framework developed for assessing sites for SPB disposals through a case study using input data for the model based on an illustrative landfill site.

5. The scope of this report covers the deterministic calculations using best-estimate values for all the parameters and a small sub-set of sensitivity studies to illustrate the issues which users must consider when undertaking radiological capacity calculations and authorising disposals.

1.2 Methodology

6. The aim of the Case Study is to demonstrate use of the Assessment Methodology for establishing the available radiological capacity for disposal of LLW at SPB sites.

7. The information required by the assessment methodology is described in detail in companion reports (SNIFFER 2006a, 2006b). There are four types of information required:

- Site-specific. Data describing the site and its setting.
- Scenario-dependent. Data describing the release scenarios being considered.

- Reference values. Data describing typical material properties and habits of exposed groups.
- Established values. Constants and literature data for radionuclide-dependent parameters.

8. The Assessment Methodology requires specification of values for the first two groups by the user. Default values are provided for the reference values, but these can be changed to investigate model sensitivities and to account for particular features of a specific site. There is normally no reason for a user to change the final group of parameter values.

9. For this Case Study, a set of site-specific and scenario-dependent values has been established to represent a generic site (Site A), based on a typical modern landfill for non-hazardous wastes. A description of this generic site and the associated data values are provided in Section 2.1, and values for all the potential release scenarios described in the Principles Document (SNIFFER 2005) are provided in Section 2.2. For the Reference Case, this information and the default set of reference values give the potential annual dose resulting from a 1 MBq disposal of each radionuclide considered (Section 2.3).

10. For the Reference Case calculations, a set of eighteen radionuclides has been used, covering a range of half-lives and other properties (Table 1.1).

Table 1.1: List of radionuclides used in the Reference Case calculations and sensitivity studies.

Radionuclide	Half-life (yrs)	Sensitivity Studies	Radionuclide	Half-life (yrs)	Sensitivity Studies
H-3	12.31	*	Sn-126	100021	
C-14	5728	*	I-129	1.6E+07	
Cl-36	301368		Cs-137	30.01	*
Fe-55	2.70		Pb-210	22.29	
Co-60	5.29		Ra-226	1601	*
Sr-90	29.12	*	Th-232	1.4E+10	
Tc-99	213276		U-238	4.5E+09	*
Ru-106	1.01	*	Pu-239	24068	*
Ag-108m	126.95		Am-241	433.22	

11. The assessment methodology described in the Principles Document (SNIFFER 2005) and Technical Reference Manual (SNIFFER 2006a) is based on the assumption that, for disposal of LLW in SPB sites, there is a linear relationship between the disposed inventory and dose. In other words, the methodology assumes that if the inventory were doubled the dose would also double. Using this assumption, the potential radiological capacity of a site can be calculated through the ratio between the dose from a 1 MBq disposal and the 20 μSv/yr constraint proposed for this type of disposal. Radiological capacity calculations are presented in Section 4.

12. In addition to the Reference Case calculations, a set of key sensitivities have also been investigated in this Case Study (Section 3). For these sensitivity studies, a sub-set of eight radionuclides was used, including both short-lived and long-lived radionuclides and those with the potential to generate radioactive gases (H-3, C-14 and Ra-226).

1.3 Structure

13. The report is structured as follows:

- Section 2 presents the Reference Case for Site A. This includes a summary description of the site and the associated site-specific parameter values, data for the release scenarios considered, and reference dose calculations based on unit (1 MBq) disposals for each radionuclide.
- Section 3 presents sensitivity analyses for Site A, using a selection of radionuclides from the Reference Case, including a description of the parameters selected and the methodology adopted in carrying out the analyses.
- Section 4 presents radiological capacity calculations for Site A, based on the Reference Case and sensitivity studies.

14. Appendix A presents the tables of results from the sensitivity studies carried out in Section 3.

2 Reference Case Calculations

15. As noted in the Introduction, the Assessment Methodology requires some parameters describing a potential SPB site and its surroundings to be specified on a site-specific basis. For the purpose of the Reference Case calculations described here, a generic site (Site A) has been used as the basis for these parameter values. Site A is based on a real landfill site accepting non-hazardous wastes and using established construction and operational practises.

16. Although the construction and operational aspects of the generic Site A are typical of many potential SPB sites, the geological and hydrogeological characteristics of Site A and the location of potential receptors cannot necessarily be regarded as applicable to other sites. The results of the Reference Case calculations should therefore not be regarded as generic, or used to determine radiological capacities at other sites. These results can, however, be used to indicate the relative importance of different radionuclides in different scenarios and to identify radionuclides that could have a significant effect on determining radiological capacities at a range of potential sites.

17. This section provides details of the information used to specify the Reference Case calculations. This includes information describing the site and its surroundings (Section 2.1) and the scenarios considered (Section 2.2). Reference values, which describe such aspects as consumption habits, were not changed from the default values provided in the assessment methodology (SNIFFER 2006a, 2006b). The information described here will be typically available for a landfill site from documents such as the Site Working Plan, the Environmental Statement, the Waste Management Licence and Operational Procedures. A brief description is provided for each type of information; further details are provided in the User Manual (SNIFFER 2006b).

18. The specific doses calculated for the Reference Case calculations are presented in Section 2.3. In addition to tabulated results, summary diagrams are used to illustrate the relative importance of different pathways and radionuclides. Illustrative radiological capacities for Site A based on these results are presented in Section 4.

2.1 Site Description

2.1.1 General Site Characteristics

19. The volume and areal extent of a landfill site help to determine the potential radiological capacity through the amount of leachate that can form. For operational reasons, a site may be divided into sub-areas that will be filled sequentially and progressively restored so that only a part of the site will be worked on at any one time. However, unless there is complete hydraulic isolation of particular cells, the assessment methodology assumes that there is mixing of leachate throughout the site and it is the overall volume that is used.

20. These aspects of the site must be defined on a site-specific basis. Using the information available in the Site Working Plan and Waste Management Licence for the landfill site on which Site A is based, it is assumed that the overall surface area for Site A is 85×10^4 m^2, but that only about half of this area (42.39×10^4 m^2) is designated for landfill purposes. The total volume of waste that the site will accommodate is assumed to be 4.0×10^6 m^3, with the yearly amount of waste to be disposed of being limited to 4.0×10^5 tonnes. The projected lifetime of Site A is assumed to be 13 years.

21. During the operation of each phase, it is assumed that bales will be used to divide the area into smaller disposal cells, and that completed cells will be capped by a clay layer having a minimum thickness of 1 m and a permeability not exceeding 1×10^{-7} m/s. It is assumed that, following closure and site restoration, the land will be returned to agricultural and recreational uses.

22. On the basis of this description, the following site-specific values have been used for the Reference Case calculations:

- Duration of landfill operations = 13 yr.
- Surface area of the landfill = 42.39×10^4 m^2.
- Volume of the landfill = 4.0×10^6 m^3.

2.1.2 Liner Design

23. Under current landfill regulations, non-hazardous waste sites must have a geological barrier or artificial equivalent to protect groundwater, soil, and surface water. The performance of the barrier must be equivalent to a layer with a hydraulic conductivity ≤ 1.0×10^{-9} m/s and a thickness ≥ 1 m.

24. Different approaches may be used to satisfy this requirement at different sites, but a common approach is to use of a 2 mm thick synthetic geomembrane, manufactured from high-density polyethylene (H.D.P.E), laid over a clay lining system. The specifications for the clay liner will normally be for the construction of a continuous clay layer, at least 1 m thick, with a maximum permeability of 1.0×10^{-9} m/s or less across each phase of the landfill site. Construction Quality Assurance (CQA) procedures are used to ensure that specified performance criteria are met. Liner CQA reports can be used to derive a site-specific average permeability.

25. In a phased development, each phase is fitted with a protective liner and associated drainage prior to waste being deposited. Under-liner drainage for clean water outflow will, as a rule, be provided for each phase to prevent potential instability of the liner on slopes arising from a water build-up. This drainage network will also provide a means of monitoring beneath the liner for any leakage. Clean waters discharge to local watercourses and are subject to the terms of the discharge by the regulator. Management of leachate collected from the landfill is described in the following section.

26. Above the liner, it is common practise for a protection layer of smooth gravel or similar to be emplaced. At the site on which Site A is based, a further protective layer of high-density bales was emplaced before construction of the operating cells. These layers are not explicitly considered in the assessment.

27. In the assessment methodology, the hydraulic properties of the effective geological barrier are assigned reference values that should be applicable for all sites:

- Hydraulic conductivity = 1.0×10^{-9} m/s.
- Bulk density = 2.0×10^3 kg/m^3.
- Porosity = 0.5.

28. The thickness of the effective geological barrier can, however, be specified for a particular site. In the case of the site used as the basis for Site A, this value has been taken as the thickness of the artificial mineral barrier lying under the liner, justified by the fact that this layer is assumed to be more resistant to leachate transport than the unsaturated zone. This

assumption is conservative, but a thicker effective geological barrier may be justifiable at other sites.

- Thickness of the effective geological barrier = 1 m.

2.1.3 Leachate Management Procedures

29. A landfill will need an efficient leachate collection and removal system to enable leachate to be removed from the site for disposal or recirculation. There is no explicit treatment of leachate management in the assessment methodology, but the management options at a site do help to determine whether the aerosol and spillage scenarios require consideration.

30. At the site used as the basis for the description of Site A, automatic leachate pumps are positioned at 1 m head above the liner base level to extract leachate, and the leachate management procedures followed comprise:

- Leachate treatment in two 1.5×10^3 m3 main aeration lagoons. Aeration with air blowers keeps the dissolved oxygen concentration at a minimum of 1-2 mg/l.
- Aerobic biological process using activated sludge.
- 10-day retention time.
- 750 m^3 discharge lagoon.
- 2 l/s discharge in sewer.
- Recirculation or spray irrigation to upper parts of the site if volumes/standards are not reached.
- Surface water (tested) discharged to marsh as required.

31. On the basis of these procedures, both the aerosol and spillage of leachate scenarios are considered in the Reference Case for Site A. At sites where aeration and spray irrigation are not practised it would be appropriate to discount the aerosol scenario. The spillage scenario should only be discounted for sites where there is minimal handling of leachate.

2.1.4 Gas Management Procedures

32. Landfill sites used for the disposal of non-hazardous wastes are likely to require the installation of a gas management system so as to:

- Minimise the risk of migration of landfill gas beyond the site perimeter.
- Prevent air ingress into the landfill and minimise the risk of underground fires.
- Minimise damage to soils and vegetation within the restored landfill area.

33. There is no explicit treatment of gas management in the assessment methodology, but the management options in use or proposed at a site do help to determine how the gas release pathway is modelled.

34. At the landfill site that Site A is based on, vertical gas venting chimneys with radial collector pathways are proposed for construction during active waste emplacement in each cell to counter the potential risk of off-site migration of landfill gas. Such chimneys provide passive venting and have typical centres of the order of 30 – 50 m.

35. In addition to passive venting during operations, the plans for this site identify a probable requirement for an active gas abstraction system during the restoration phase. This will consist of a network of horizontal collector pipes connected to vertical collection pipes under negative pressure induced by a gas booster. The gas can be flared or utilised in a generator according to the volume produced. Monitoring of landfill gas will be carried out until the wastes are stable.

36. Release of radioactive gases through the cap is a pathway included by default within the assessment methodology. The gas management practises at the site used as the basis for Site A indicate that this pathway should be considered in the Reference Case. The proposed development of the landfill gas management systems during the restoration phase indicates that it would also be appropriate to include the remediation / re-engineering scenario in the Reference Case.

2.1.5 Geology and Hydrogeology

37. The geological and hydrogeological characteristics of a potential SPB site are key controls on the radiological capacity because of their effect on the transport of leachate to receptors. These characteristics are site-specific, although the assessment methodology does provide generic parameter values for key rock types.

38. For the Case Study, the geological and hydrogeological characteristics are based on conditions at the landfill site on which Site A is based, where fractured volcanic rocks are overlain by drift deposits dominated by boulder clay. These characteristics of Site A cannot necessarily be regarded as applicable to other sites, and the specific doses calculated for the leachate pathways should therefore not be regarded as generic.

39. The description of the geology and hydrogeology at the site on which Site A is based notes that, although there are minor peats, silts and clays overlying the boulder clay in parts of the site, the majority of groundwater flow is in the volcanic rock. Where there is rockhead at the surface, the water table is assumed to be less than 3 m below surface. Elsewhere, the water table is assumed to be confined to beneath the boulder clay. The effective thickness of the groundwater flow system in the region of the site is assumed to be 30 m, and the hydraulic gradient within this system is estimated to range from 0.02 to 0.05.

40. Permeability tests conducted in boreholes suggest that there is a dual permeability system within the volcanic rock, with a low “primary” porosity and a variable amount of fracturing. Slug tests yield higher values than rising head tests, indicating a moderately heterogeneous system, with variable void integration.

41. On the basis of this description, the following parameter values have been used for the Reference Case calculations:

- Depth to groundwater = 3 m (the effective geological barrier and unsaturated zone are modelled as a single unit having a thickness of 3 m).
- Thickness of the unsaturated zone = 2 m (depth to groundwater minus thickness of the effective geological barrier – see Section 2.1.2).
- Thickness of groundwater zone = 30 m.
- Hydraulic gradient = 0.05.
- Groundwater-bearing rock:
  - Hydraulic conductivity = 1.0×10^{-5} m/s.
  - Bulk density = 2.3×10^{3} kg/m^{3}.
  - Porosity = 0.4.

42. An important process in determining the transport of radionuclides through the groundwater pathway is sorption. In the assessment methodology, sorption in different parts of the system is modelled through use of distribution or partition coefficients (K_ds). The assessment methodology includes generic values for distribution coefficients in different parts of the disposal system, including values for several different rock types. In the case of the waste itself, the generic distribution coefficient for all radionuclides is set to zero. A low K_d value ensures that contaminants are assumed to travel very quickly through the waste, so that decay of

radionuclides within the waste is not taken into account. This is in accord with the conservative approach to assessing specific doses and radiological capacity described in the Principles Document (SNIFFER 2005).

43. The assessment methodology includes a facility for entering site-specific K_d values if additional information is available. In the case of the site on which Site A is based, however, here is no additional information on distribution coefficients and the Case Study uses the generic values listed in Table 2.1.

Table 2.1: Distribution coefficient (K_d) values for different parts of the disposal system.

Radionuclide	K_d waste (m^3/kg)	K_d soil (m^3/kg)	K_d rock (m^3/kg)	Kd barrier (m^3/kg)
H-3	0	1.0×10^{-4}	1.0×10^{-6}	0
C-14	0	0.1	1.0×10^{-6}	1.0×10^{-3}
Cl-36	0	1.5×10^{-2}	1.0×10^{-6}	1.5×10^{-2}
Fe-55	0	0.22	0	0
Co-60	0	6.0×10^{-2}	1.0×10^{-2}	0.5
Sr-90	0	1.3×10^{-2}	1.0×10^{-6}	0.1
Tc-99	0	1.4×10^{-4}	0.19	1.0×10^{-3}
Ru-106	0	5.5×10^{-2}	0	0
Ag-108m	0	9.0×10^{-2}	9.0×10^{-2}	0.18
Sn-126	0	0.13	0.1	0.67
I-129	0	1.0×10^{-3}	1.0×10^{-6}	1.0×10^{-3}
Cs-137	0	0.27	0.1	2.0
Pb-210	0	0.27	3.2	0.5
Ra-226	0	0.49	1.0×10^{-3}	9.0
Th-232	0	3.0	0.5	6.0
U-238	0	3.3×10^{-2}	1.0×10^{-2}	4.6×10^{-2}
Pu-239	0	0.54	0.1	7.6
Am-241	0	2.0	3.2	7.6

2.1.6 Biosphere and Exposed Groups

44. For the purposes of assessing the potential for landfill sites to receive SPB wastes, five potentially exposed groups have been identified as the basis for dose calculations (SNIFFER 2005). These comprise two groups of workers and three groups of members of the public.

- Workers 1. This group comprises workers operating the site during the normal operations phase. The site operators will have the highest occupancy (i.e., period of time spent on the site), and so will receive the highest doses from the exposure pathways associated with the surface of the landfill. For the normal operations scenario, the pathways are external irradiation from the landfill surface, inhalation of aerosols from leachate and potentially inhalation of radioactive gases and dust or particles from fires.
- Workers 2. This group comprises workers engaged in site operations that may lead to the exposure of waste. Pathways include external irradiation from exposed waste and inhalation or ingestion of dust from contaminated material. At sites where there is landfill gas abstraction, this group may be exposed during normal site operations. At all sites exposure may occur during remediation or re-engineering. A group with similar habits may also be involved in inadvertent intrusion of the landfill after closure.

- Public 1. This group comprises members of the public living sufficiently close to the site to be affected directly by site operations. Members of this group may inhale aerosols from leachate treatment and gas from landfill gas utilisation. Spillage of leachate during treatment or handling may contaminate surface water and lead to exposure through ingestion of water or foodstuffs. Fires on the site may lead to exposure of this group through inhalation of dust or particles, ingestion of dust deposited on foodstuffs or irradiation from dust deposited on the ground.
- Public 2. This group comprises members of the public living at the point of groundwater discharge or surface water consumption where they will receive the highest dose associated with contaminated groundwater. Potential exposure pathways include drinking contaminated water, consumption of crops irrigated by contaminated water, consumption of fish and inhalation of dust from soil contaminated by groundwater discharge. The same groundwater pathways and exposed public apply for the normal post-closure scenario and to the failure of barrier and spillage of leachate scenarios.
- Public 3. This group comprises members of the public living on or in close proximity to the site after capping and closure. There are three sets of exposure pathways that could affect this group. The first relates to the continued, normal evolution of the site and comprises inhalation of radioactive gases. The second comprises the ingestion of soil and food contaminated during a bathtubbing incident. These two pathways could potentially occur at any time after closure. The third set of pathways could occur only after loss of control over site use and relates to contamination after an intrusion. It is assumed that the land will be levelled, and that the new soil layer may contain a component of the radioactive waste. Doses are calculated for a member of the public residing on this land and farming it for crops and livestock.

45. For the Case Study, the location and characteristics of the public groups are based on the biosphere and population around the landfill site on which Site A is based.

46. There are no occupied properties immediately adjacent to the site boundary, but there are a number of households in the area. Within 250 m of the site there are two occupied farms.

47. Wells and boreholes are present close to site, and are assumed to be in use for domestic/farm purposes. Sustainable yields from wells are generally less than 0.5 l/s and rarely exceed 2 l/s.

48. Site closure and remediation will result in runoff from the cap to a number of sub-catchments. The annual rainfall is of the order of 1.1 m and evapotranspiration is of the order of 0.45 m. Runoff is estimated at 45%, which implies an infiltration value of 0.155 m. It is assumed that the bulk of the surface water intercepted by the site will discharge to an adjacent marsh area with an overall area of some 1.0×10^3 m^3.

49. On the basis of this description, the following values have been used for the Reference Case calculations:

- Distance to water abstraction point = 250 m.
- Net water infiltration rate through soil = 0.155 m/yr.

2.2 Potential Release Scenarios

50. The Principles Document (SNIFFER 2005) describes a series of potential release scenarios that could result in workers and / or members of the public receiving doses from radionuclides disposed at an SPB site. These scenarios may occur during operations (Table 2.2) or after closure of the site (Table 2.3).

Table 2.2: Operational scenarios included in the framework and the associated hazards.

Scenario name	Description	Hazards
Normal operations	Expected operation of the landfill up to capping and closure, as approved by the relevant Agency. Doses to site workers and to the public are considered.	Gas Release
		Liquid release (leachate)
		Direct irradiation
Landfill gas abstraction	Workers expose waste during operations to install and remediate landfill gas system.	Solid release (dust while uncovered)
		Direct irradiation
Barrier failure	Failure of the artificial sealing liner and geological barrier during operations. Doses to the public are considered.	Liquid release (leachate)
Leachate spillage	Unintentional release of leachate to surface water. Doses to the public are considered.	Liquid release (leachate)
Site remediation or re-engineering	Workers expose waste during operations to remediate containment failure or to enlarge or otherwise re-engineer site.	Solid release (dust while uncovered)
		Direct irradiation
Fire	Fire releases radioactivity. Doses to site workers and to the public are considered.	Solid release (dust), gases and vapour

Table 2.3: Post-closure scenarios included in the framework and the associated hazards.

Scenario name	Description	Hazards
Normal post-closure evolution	During this time, the landfill engineering is assumed to gradually degrade. Doses to the public are considered.	Gas Release
		Liquid release (leachate)
		Direct irradiation (through cover)
Bathtubbing	Blockage of the drainage system causes overflow of leachate laterally from the landfill onto the soil. Doses to the public are considered.	Liquid release (leachate)
Inadvertent excavation	Waste is inadvertently excavated and re-distributed, e.g., during building or farming. Doses to the intruder and the subsequent user of the site are considered.	Direct irradiation
		Solid release (dust)
		Solid release (waste)

51. For the Reference Case, all of these scenarios have been considered.

52. The characteristics of Site A and its surroundings used in the Reference Case calculations are based on those of a typical landfill site (Section 2.1). Additional information required to characterise the release scenarios is based on reference values derived from previous studies and established sources (SNIFFER 2006a, 2006b) and on assumptions regarding the occurrence and extent of the release scenarios. The significance of several of these assumptions is assessed in the sensitivity studies described in Section 3.

53. During normal operations, doses may be received through direct irradiation, through inhalation of gas or aerosols (from leachate management), or through ingestion of foodstuffs (including drinking water) contaminated by releases to groundwater or surface water. Key assumptions made for the normal operations scenario are:

- Overall area of holes in the liner = 3.8×10^{-4} m^2.
- Volume of river compartment into which groundwater discharges = 2.0×10^4 m^3 (based on a length of 2×10^3 m and a cross-sectional area of 10 m^2).
- Soil thickness = 0.25 m.
- Proportion of time spent indoors and outdoors for resident in house on cap = 3 : 1.
- Number of leachate sprayings leading to release of aerosols = 1 per year.
- Time over which aerosols are released and public exposed = 1 h.

54. The barrier failure, leachate spillage and bathtubbing scenarios may lead to doses through ingestion of contaminated foodstuffs (including drinking water). Key assumptions made for these scenarios are:

- Volume of contaminated leachate spilt into surface waters = 10 m^3.
- Volume of leachate that overflows into the growing area through bathtubbing = 1000 m^3.

55. The fire scenario can lead to doses through inhalation of gas and dust from the fire or through ingestion of foodstuffs contaminated by fallout from the fire. Key assumptions made for the fire scenario are:

- Number of fires per year = 2.
- Volume of the waste consumed in each fire = 1000 m^3.
- Height of smoke and dust plume from fire = 10 m.
- Duration of each fire = 1 h.

56. In terms of pathways considered, the normal post-closure evolution scenario is similar to the operational scenario, except that there is no leachate management and hence no release of aerosols. Also, the only exposed group considered for this scenario is members of the pubic living on or close to the site. The key assumption for this scenario is the period over which the cap retains some effectiveness in terms of controlling infiltration:

- Time of cap failure = 100 yr.

57. The landfill gas abstraction, site remediation and inadvertent excavation scenarios can all lead to doses through direct irradiation or ingestion of dust. The inadvertent excavation scenario can also lead to doses from foodstuffs grown on contaminated soil. Key assumptions made for these scenarios include:

- Time of excavation = 20 yr.
- Time the excavator is exposed to the material = 88 h/yr (0.01 yr).

- Proportion of waste containing activity = 2.5 x 10^{-6} [SPB disposals to the landfill are assumed to be concentrated into a small volume (10 m^3) of waste within the overall site volume of 4.0 x 10^{-6} m^3].

2.3 Reference Case Results

58. For the Reference Case, specific doses (dose from a 1Mbq inventory of each radionuclide considered) have been calculated for all of the potentially exposed groups. Results are presented in Table 2.4 and illustrated in Figures 2.1 to 2.12.

Table 2.4: Doses per unit disposal (μSv yr^{-1} MBq^{-1}) for the Reference Case for Site A. Aggregate doses are given for each exposed group, together with doses from contributing pathways. Events in *italics* are not certain to occur, other events are part of the normal evolution scenario.

	H-3	C-14	Cl-36	Fe-55	Co-60	Sr-90	Tc-99	Ru-106	Ag-108m	Sn-126	I-129	Cs-137	Pb-210	Ra-226	Th-232	U-238	Pu-239	Am-241
Worker 1	6.11E-08	3.37E-06	9.86E-09	1.04E-09	1.89E-06	2.16E-07	1.76E-08	1.07E-06	1.15E-07	3.78E-08	5.44E-06	5.27E-08	4.27E-04	1.29E-05	1.49E-04	1.08E-05	1.62E-04	1.30E-04
Gas (inhalation)	2.18E-08	2.49E-06	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	7.36E-08	N/A	N/A	N/A	N/A
Aerosol (inhalation, external irradiation)	3.12E-10	6.96E-09	8.76E-09	9.24E-10	3.76E-08	1.92E-07	1.56E-08	7.92E-08	4.47E-08	3.36E-08	4.32E-08	4.68E-08	6.72E-06	1.14E-05	1.32E-04	9.60E-06	1.44E-04	1.15E-04
External irradiation	0.00E+00	3.98E-41	9.45E-16	0.00E+00	1.84E-06	9.14E-22	7.98E-29	0.00E+00	1.41E-08	4.59E-24	6.00E-87	5.65E-20	6.15E-57	2.80E-16	1.73E-27	9.92E-61	1.42E-17	3.79E-36
Fire (inhalation, external irradiation)	*3.90E-08*	*8.70E-07*	*1.10E-09*	*1.16E-10*	*4.70E-09*	*2.40E-08*	*1.95E-09*	*9.90E-07*	*5.58E-08*	*4.20E-09*	*5.40E-06*	*5.85E-09*	*4.20E-04*	*1.43E-06*	*1.65E-05*	*1.20E-06*	*1.80E-05*	*1.44E-05*
Worker 2	1.16E-05 8.71E-07	3.54E-04 3.52E-04	1.50E-03 1.50E-03	1.54E-04 1.13E-09	6.28E+00 1.52E-02	1.48E-02 4.96E-03	5.43E-04 5.42E-04	4.12E-03 8.14E-17	3.74E+00 2.91E+00	5.97E-02 5.96E-02	5.26E-02 5.26E-02	6.50E-03 2.25E-03	3.80E-01 9.10E-02	2.75E-01 2.70E-01	1.76E+00 1.76E+00	1.40E-01 1.40E-01	1.92E+00 1.91E+00	1.55E+00 1.44E+00
Site operations (inhalation, external irradiation, ingestion)	*1.16E-05*	*3.54E-04*	*1.50E-03*	*1.54E-04*	*6.28E+00*	*1.48E-02*	*5.43E-04*	*4.12E-03*	*3.74E+00*	*5.97E-02*	*5.26E-02*	*6.50E-03*	*3.80E-01*	*2.75E-01*	*1.76E+00*	*1.40E-01*	*1.92E+00*	*1.55E+00*
Inadvertent excavation (inhalation, external irradiation, ingestion)	*8.71E-07*	*3.52E-04*	*1.50E-03*	*1.13E-09*	*1.52E-02*	*4.96E-03*	*5.42E-04*	*8.14E-17*	*2.91E+00*	*5.96E-02*	*5.26E-02*	*2.25E-03*	*9.10E-02*	*2.70E-01*	*1.76E+00*	*1.40E-01*	*1.91E+00*	*1.44E+00*
Public 1	2.65E-07	1.18E-04	1.32E-05	3.45E-06	4.78E-05	2.68E-04	4.95E-06	7.36E-05	1.76E-05	1.30E-04	1.38E-03	6.72E-04	9.72E-03	2.42E-03	1.98E-03	3.51E-04	1.98E-03	1.69E-03
Gas (inhalation)	2.84E-08	3.24E-06	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	9.59E-08	N/A	N/A	N/A	N/A
Aerosol (inhalation, external irradiation, ingestion)	2.96E-10	6.97E-09	4.36E-08	1.42E-09	4.36E-07	2.17E-07	1.43E-08	7.96E-08	3.27E-07	4.64E-08	2.61E-07	6.57E-08	6.99E-06	1.01E-05	1.10E-04	8.03E-06	1.20E-04	9.64E-05
Fire (inhalation, external irradiation, ingestion)	*3.93E-08*	*9.46E-07*	*2.68E-09*	*2.20E-10*	*3.01E-08*	*3.07E-08*	*1.87E-09*	*1.08E-06*	*2.34E-07*	*5.85E-09*	*4.67E-05*	*9.86E-09*	*4.81E-04*	*1.29E-06*	*1.38E-05*	*1.01E-06*	*1.50E-05*	*1.21E-05*
Spillage (inhalation, external irradiation, ingestion)	*1.97E-07*	*1.14E-04*	*1.31E-05*	*3.45E-06*	*4.74E-05*	*2.67E-04*	*4.93E-06*	*7.25E-05*	*1.70E-05*	*1.30E-04*	*1.34E-03*	*6.72E-04*	*9.23E-03*	*2.41E-03*	*1.85E-03*	*3.42E-04*	*1.85E-03*	*1.58E-03*
Public 2	7.59E-07	4.20E-05	1.45E-05	2.36E-06	5.56E-16	1.08E-05	4.80E-08	5.86E-06	6.30E-13	1.38E-07	5.16E-03	5.18E-19	5.43E-34	3.35E-06	7.96E-11	5.42E-05	4.63E-07	3.61E-22
Groundwater (inhalation, external irradiation, ingestion)	2.79E-08	2.60E-05	6.58E-06	4.69E-10	4.43E-19	1.24E-06	4.40E-08	2.56E-12	1.96E-13	5.61E-08	3.16E-03	6.95E-20	5.12E-35	1.33E-06	2.98E-11	2.98E-05	1.83E-07	1.32E-22
Barrier failure (inhalation, external irradiation, ingestion)	*7.59E-07*	*4.20E-05*	*1.45E-05*	*2.36E-06*	*5.56E-16*	*1.08E-05*	*4.80E-08*	*5.86E-06*	*6.30E-13*	*1.38E-07*	*5.16E-03*	*5.18E-19*	*5.43E-34*	*3.35E-06*	*7.96E-11*	*5.42E-05*	*4.63E-07*	*3.61E-22*
Public 3	1.34E-06	2.06E-05	5.35E-04	5.79E-09	8.78E-06	8.42E-04	3.94E-04	8.87E-10	9.83E-05	4.41E-05	1.74E-03	2.21E-05	1.48E-04	7.32E-04	1.40E-05	2.83E-06	1.04E-05	1.06E-05
Gas (inhalation)	3.88E-08	9.22E-06	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	2.71E-07	N/A	N/A	N/A	N/A
External irradiation	0.00E+00	1.61E-46	3.84E-21	0.00E+00	1.36E-12	2.73E-27	3.24E-34	0.00E+00	5.33E-14	1.86E-29	2.44E-92	1.70E-25	1.67E-62	1.13E-21	7.03E-33	4.03E-66	5.77E-23	1.51E-41
Inadvertent excavation (inhalation, external irradiation, ingestion)	*4.57E-07*	*8.84E-06*	*4.15E-04*	*4.18E-12*	*3.87E-07*	*5.17E-04*	*3.06E-04*	*4.55E-19*	*7.32E-05*	*3.42E-05*	*1.35E-03*	*1.37E-05*	*8.20E-05*	*5.67E-04*	*1.09E-05*	*2.20E-06*	*8.10E-06*	*8.12E-06*
Bathtubbing (inhalation, external irradiation, ingestion)	*8.41E-07*	*2.55E-06*	*1.19E-04*	*5.79E-09*	*8.39E-06*	*3.25E-04*	*8.78E-05*	*8.87E-10*	*2.52E-05*	*9.83E-06*	*3.87E-04*	*8.43E-06*	*6.57E-05*	*1.65E-04*	*3.12E-06*	*6.33E-07*	*2.33E-06*	*2.46E-06*

59. The results presented in this section are mainly in terms of doses per unit disposal (μSv yr^{-1} MBq^{-1}), and show that, for the same level of disposed activity, there is a marked difference in terms of dose both between scenarios and between radionuclides. The key radionuclides for each pathway and potentially exposed groups are identified below, together with the reasons for differences between pathways.

60. Figure 2.1 shows the calculated doses per unit disposal for all exposed groups for the radionuclides included in the Reference Case. The highest doses arise through external irradiation, with Co-60 and Ag-108m the dominant radionuclides (Figure 2.2). Doses from this pathway are principally to workers, as a result of remediation or re-engineering (Worker 2). Workers who inadvertently intrude into the waste after closure (Inadvertent excavation pathway) may also receive doses through external irradiation, but for this exposed group, the dominant radionuclide is Ag-108m (Figure 2.2). The short half-life of Co-60 (5 yr) means that the Co-60 inventory is significantly reduced through decay during the operational period and the period when controls prevent inappropriate site use after closure. For workers operating the site during the normal operational period (Worker 1), doses can mainly be received through aerosol releases during leachate management or fires releasing radionuclides (Figure 2.3). The main mode of exposure is via the inhalation pathway, with the highest dose arising from Pb-210, which has a high dose coefficient for inhalation and, more importantly, a high release fraction (0.5).

Figure 2.1: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for all exposed groups (NB. Logarithmic scale).

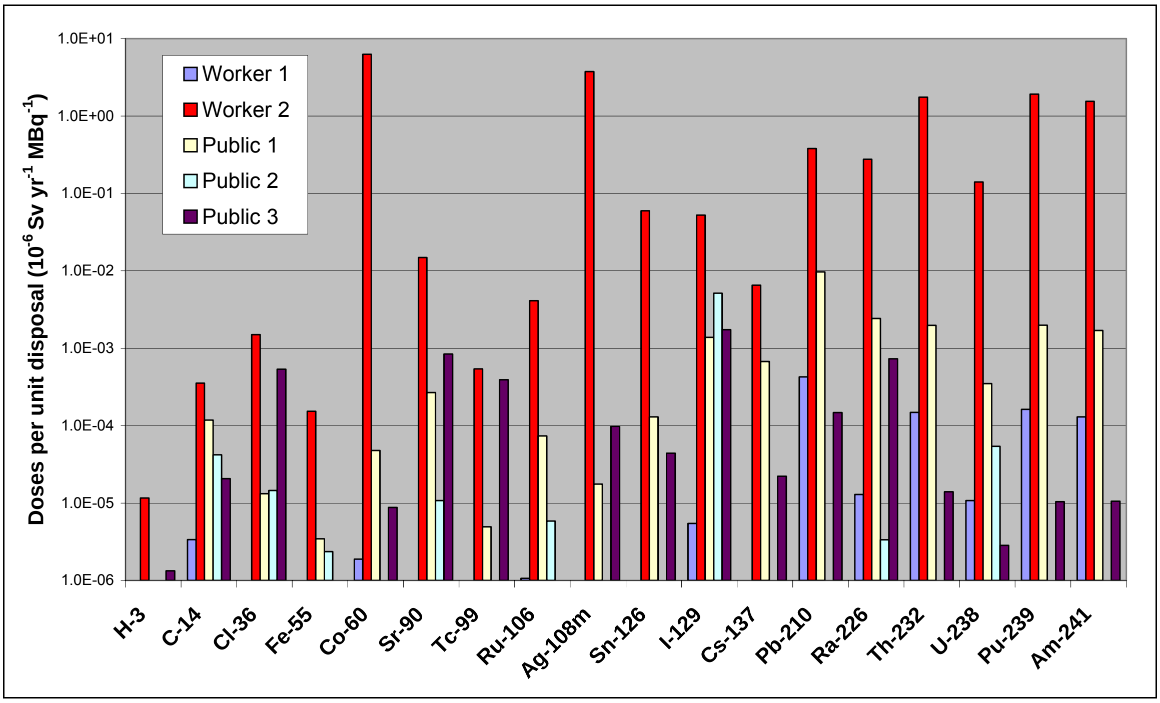


Figure 2.2: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for different pathways to the Worker 2 exposed group.

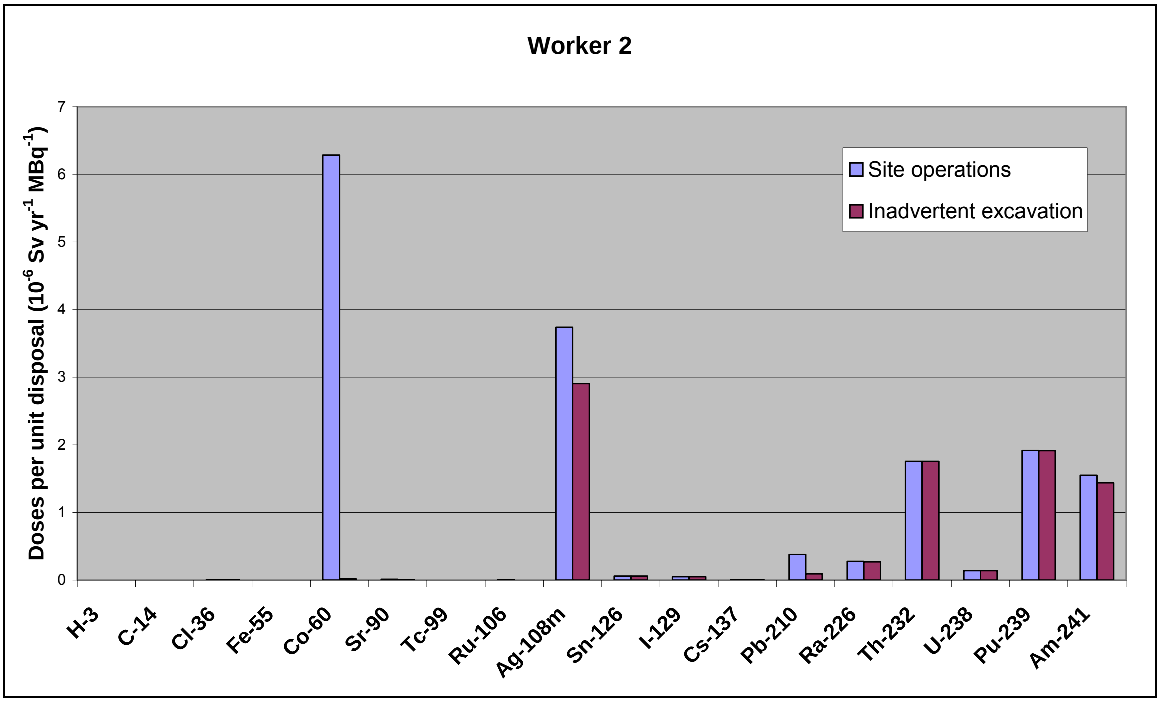
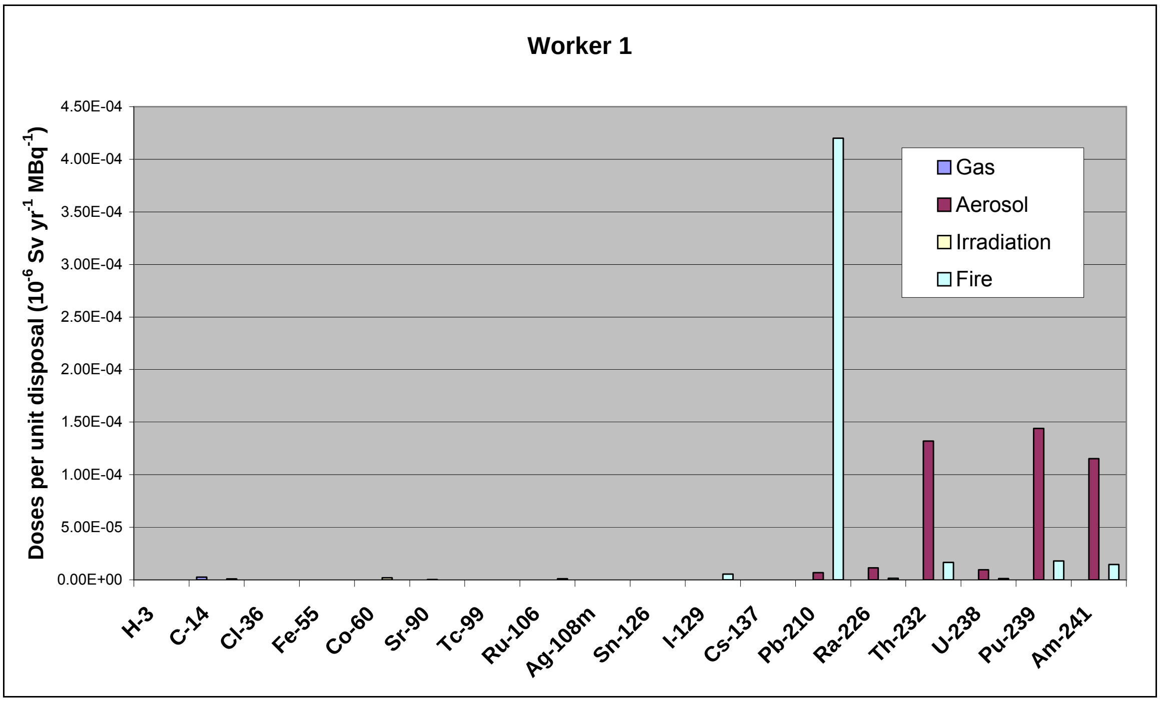


Figure 2.3: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for different pathways to the Worker 1 exposed group.



61. The highest doses per unit disposal for members of the public for the radionuclides included in the Reference Case are via the spillage of leachate scenario (Figures 2.4, 2.5 and 2.6), with Pb-210 dominating the calculated doses. The Reference Case assumes that leachate released via this accidental scenario contaminates a surface water body used by members of the public for drinking, fishing, and for watering livestock. Pb-210 has a high dose coefficient for ingestion, giving a high calculated dose from drinking contaminated water. Specific doses resulting from the spillage scenario for most other radionuclides are also dominated by the drinking water pathway.

Figure 2.4: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for different pathways to the Public 1 exposed group.

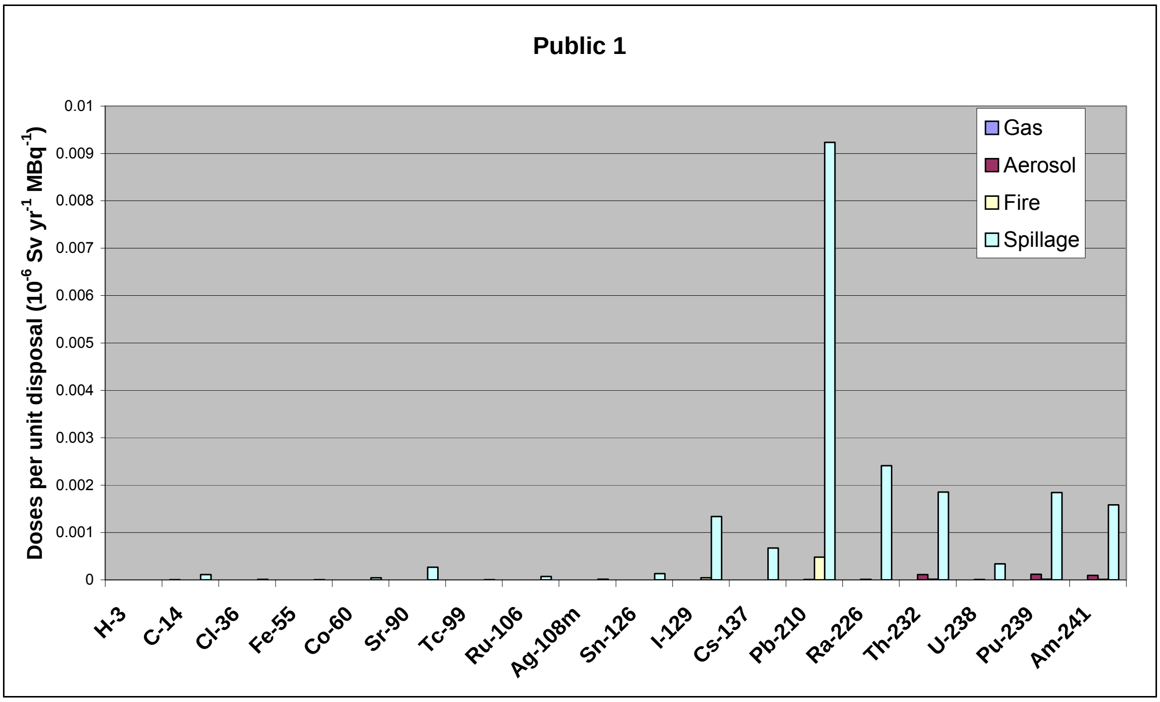


Figure 2.5: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for different pathways to the Public 2 exposed group.

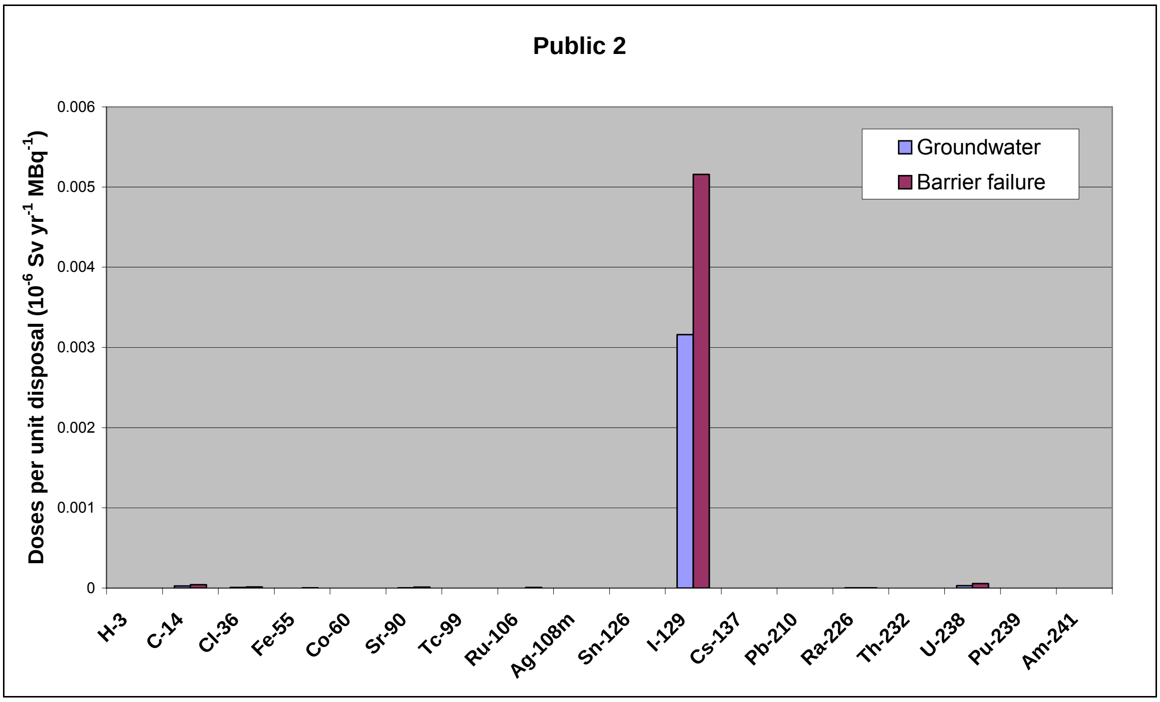
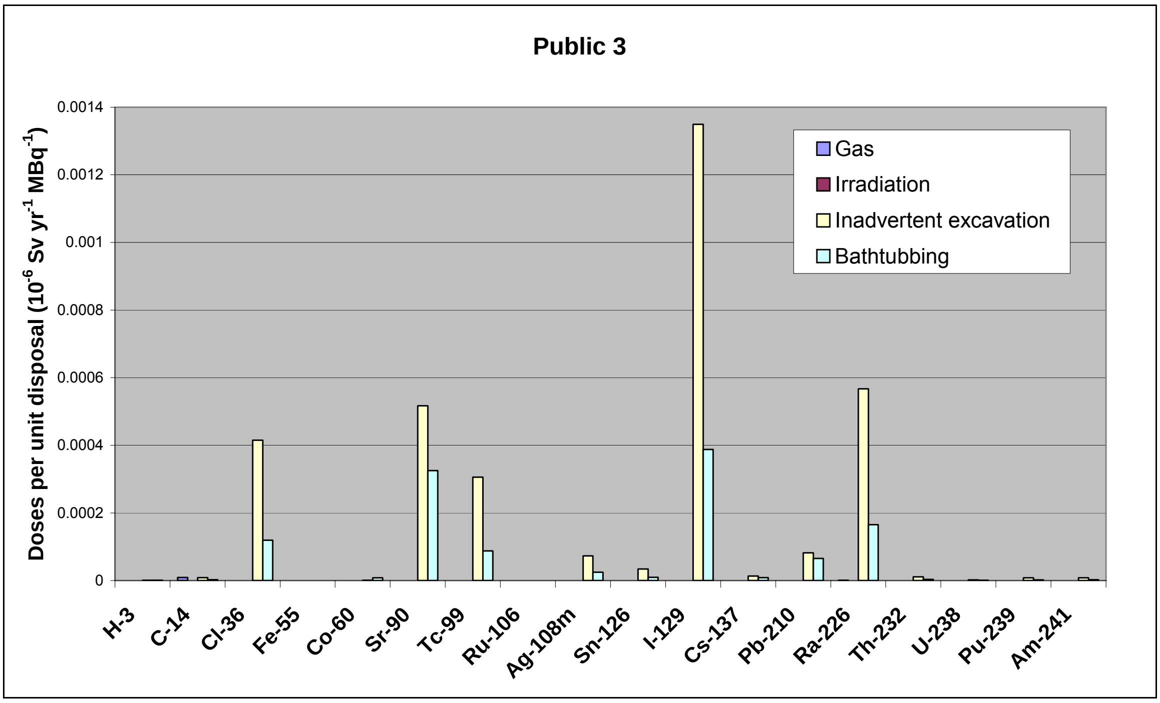


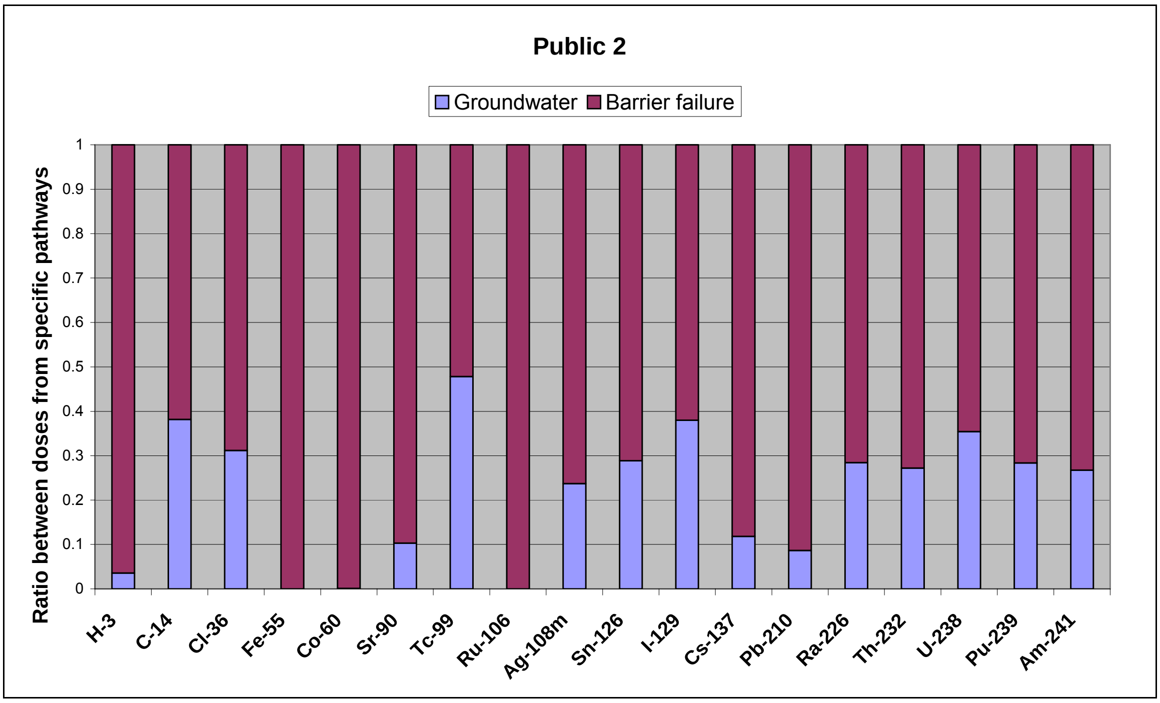
Figure 2.6: Site A Reference Case - doses per unit disposal (μSv yr^{-1} MBq^{-1}) for different pathways to the Public 3 exposed group.



62. For the operational and post-closure scenarios, doses to members of the public may arise through the gas, external irradiation, groundwater, and aerosol pathways. The highest doses per unit disposal via these pathways for the radionuclides included in the Reference Case are for I-129 through the groundwater pathway (Figure 2.5). In the Reference Case, groundwater is used for drinking, irrigation, watering livestock and fishing. All of these pathways contribute to doses, with consumption of crops irrigated with contaminated water the most important for the majority of radionuclides considered.

63. The groundwater pathway is assessed both as part of the normal evolution scenario and as the barrier failure scenario. The specific dose as a result of barrier failure is greater than that for the normal evolution release (Figure 2.7), because a greater amount of leachate is assumed to be released from the landfill as a result of such failure. Also, barrier failure is assumed to occur earlier than normal releases, so that decay has a greater effect in reducing specific doses for normal evolution than for barrier failure. This is particularly marked for Fe-55, Co-60 and Ru-106, whose short half-lives mean that there are negligible doses from the normal evolution groundwater pathway.

Figure 2.7: Site A Reference Case – contribution of different pathways to the Public 2 exposed group for each radionuclide.



64. Both the leachate spillage and barrier failure scenarios consider the use of water resources as pathways leading to doses. There is a marked difference between these two scenarios in terms of important radionuclides because sorption of radionuclides on geological media serves to decrease the rate of migration of many radionuclides sufficiently for decay to be important in determining doses via groundwater. Sorption is not an important process for the spillage scenario where leachate is released directly to surface water. I-129, which is not significantly sorbed, is therefore relatively more important in terms of specific dose for the groundwater pathway (Figure 2.5) in comparison to the spillage pathway (Figure 2.4).

65. In summary, the relative importance of different pathways differs between radionuclides because of differences in half-life and other properties. The key pathways for each exposed group are:

- Worker 1 (Figure 2.8): For the majority of the radionuclides considered, the most significant contributor to calculated doses is release of aerosols. Releases during fires is the most significant pathway for H-3, Ru-106, I-129, and Pb-210 due mostly to their relative high release fractions compared to the other radionuclides. External irradiation is the most significant contributor to the calculated dose for Co-60. This is because Co-60 is a gamma emitter with one of the highest dose conversion factors for irradiation amongst the radionuclides considered, which is only partially attenuated by the cover thickness.
- Worker 2 (Figure 2.9): Calculated doses to this exposed group from remediation or re-engineering works are generally higher than those that would arise from inadvertent excavation following the end of operations. This is because, at the time of intrusion, decay will have reduced the inventory of all radionuclides relative to that assumed for remediation. This is particularly marked for short-lived radionuclides such as Fe-55 and Co-60.
- Public 1 (Figure 2.10): For the exposed group living near the site, the highest calculated doses arise from the spillage of leachate. The assessment calculations assume that the generation of a significant volume of leachate and hence its potential release would only occur towards the end of the operational period, and doses for radionuclides with half-lives less than one year are not calculated.
- Public 2 (Figure 2.7): For the exposed group living at the point of groundwater discharge, a barrier failure accident results in higher calculated doses than the normal evolution releases to groundwater for all radionuclides, because a greater amount of leachate is assumed to be released during barrier failure and the release is also assumed to occur earlier.
- Public 3 (Figure 2.11): For the exposed group living on the site after closure, the relative contribution of different pathways to calculated doses is more varied. The inventory of short-lived radionuclides such as Fe-55, Co-60 and Ru-106 is sufficiently reduced through decay during operations and site control that calculated doses from inadvertent excavation after loss of control are low. Similarly, for the remaining radionuclides considered the ratio between specific doses for the bathtubbing and inadvertent excavation scenarios is a function of half-life – the difference in inventory between the assumed time for bathtubbing and the assumed time for excavation is proportionately less for longer-lived radionuclides. This also explains why inhalation of radioactive gases is an important contributor to the calculated dose for C-14, but not H-3.

Figure 2.8: Site A Reference Case – contribution of different pathways to the Worker 1 exposed group for each radionuclide.

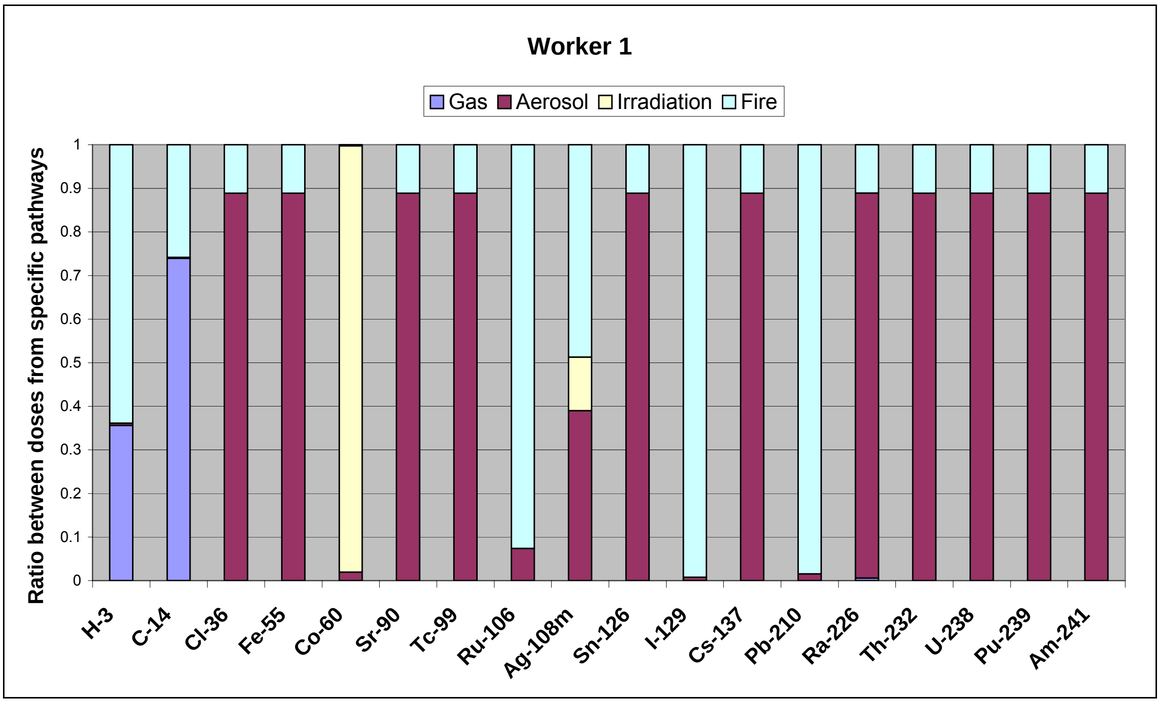


Figure 2.9: Site A Reference Case – contribution of different pathways to the Worker 2 exposed group for each radionuclide.

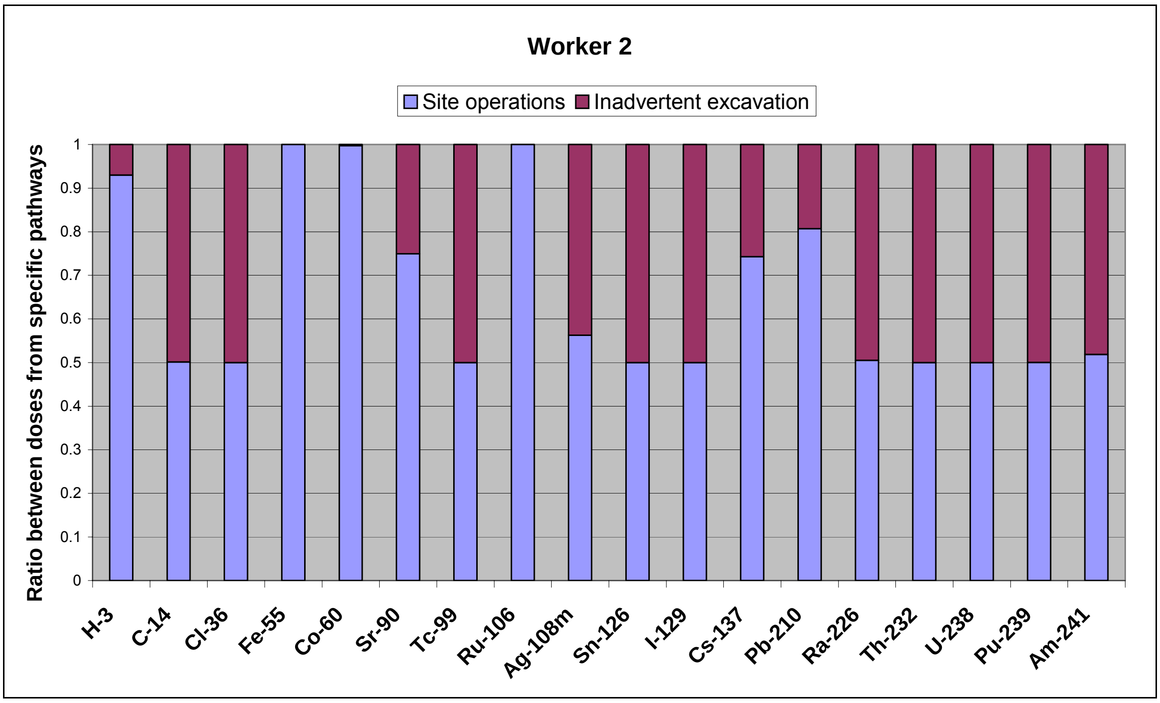


Figure 2.10: Site A Reference Case – contribution of different pathways to the Public 1 exposed group for each radionuclide.

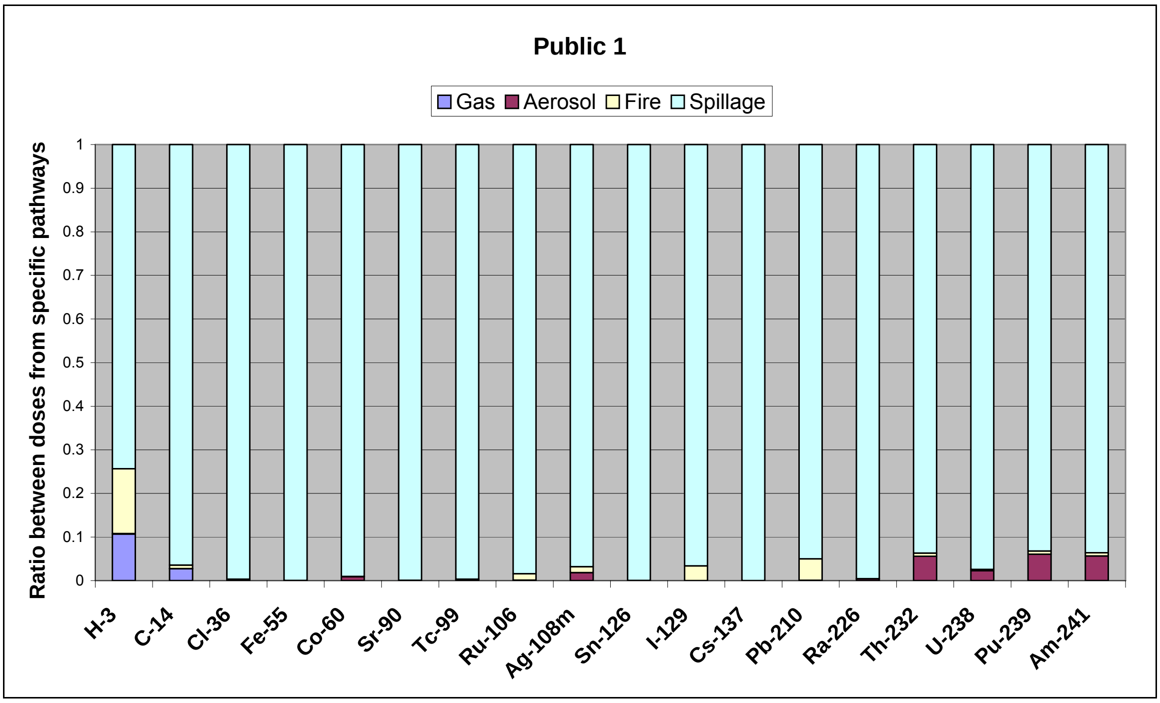
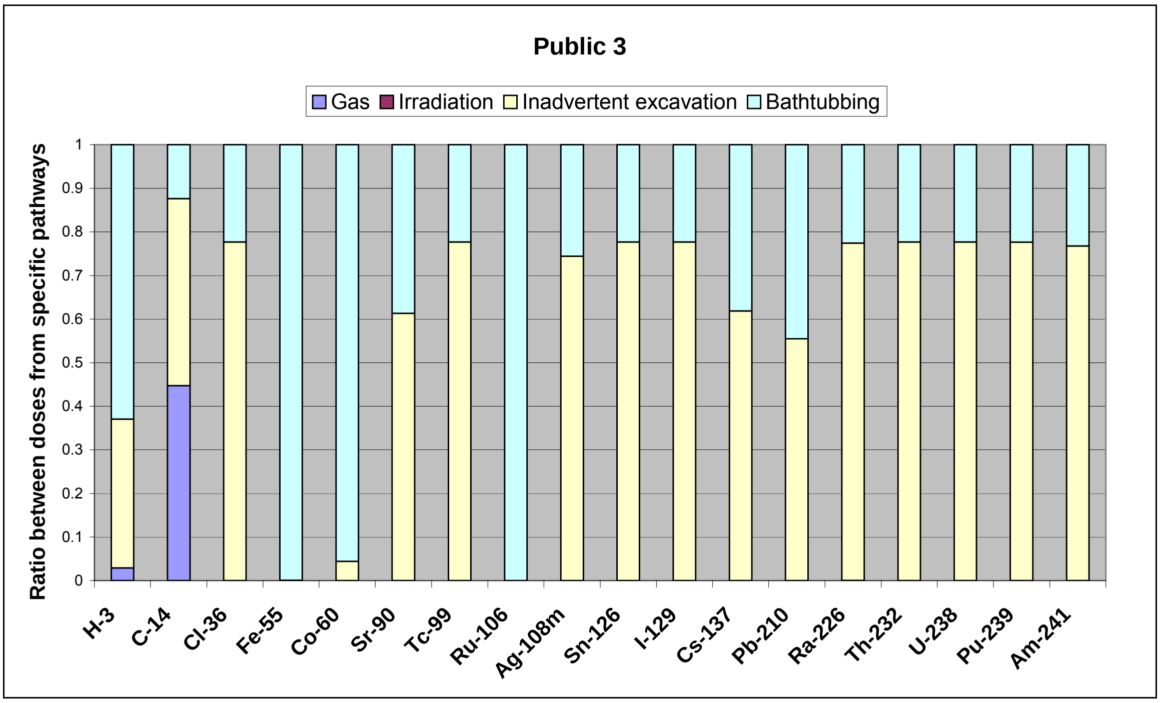


Figure 2.11: Site A Reference Case – contribution of different pathways to the Public 3 exposed group for each radionuclide.



66. The implication of these results in terms of determining radiological capacity for an SPB site is discussed in Section 4.

3 Sensitivity Studies

67. The results for the Reference Case presented in Section 2 are based on parameter values derived from a typical landfill site and from reference values (SNIFFER 2006a, 2006b). In addition, assumptions have been made about the occurrence and extent of the release scenarios. The importance of these assumptions in determining specific doses, and hence in controlling radiological capacity, has been assessed through a series of sensitivity studies.

68. The sensitivity studies have been conducted using a sub-set of eight radionuclides, representative of the range of radionuclides assessed in the Reference Case: H-3, C-14, Sr-90, Ru-106, Cs-137, Ra-226, U-238, and Pu-239.

69. This Section provides a summary of the parameters and parameter values selected and the overall results of the sensitivity calculations. Tabulated results are provided in Appendix A.

3.1 Parameters for Sensitivity Studies

70. Varying input parameter values can give rise to three types of behaviour in terms of calculated doses:

- The change in the input value results in a change of similar magnitude (linear relationship) in the calculated doses compared with the reference results across the range of radionuclides.
- The change in the input value results in a non-linear change (generally larger) in the calculated doses compared with the reference results for some of the radionuclides.
- The change in the input value results in a non-linear change (generally smaller) in the calculated doses compared with the reference results across the range of radionuclides.

71. The ten parameters selected for the sensitivity studies reported in this Case Study (Table 3.1) include examples of all three categories. Although most parameters fall into the final category, particular emphasis has been placed on the first two categories since these have most effect on calculated doses.

72. For each sensitivity test, the model was run twice with one parameter value changed from the Reference Case. In most cases, the revised input values were set to half and twice that of the original value (Table 3.1).

Table 3.1: Sensitivity studies included in the Case Study for Site A.

Id	Scenario	Parameter [unit]	Reference Case	Run 1	Run 2
4	Fire	Plume height [m]	10.0	5.0	20.0
5	Aerosol	Time public spends outside during aerosol releases [hr]	1	0.5	2
12	Inadvertent excavation scenario	Volume of the excavated waste in which the activity is contained [m^3]	10	5	20
13		Time of excavation [yr]	20	10	40
14	Groundwater	Time of cap failure [yr]	100	50	200
31		Average area of holes in liner [m^2]	3.75×10^{-4}	1.875×10^{-4}	7.5×10^{-4}
17		River length river over which radionuclides are assumed to be discharged from the geosphere [m]	2.0×10^3	1.0×10^3	3.0×10^3
36		Soil thickness [m]	0.25	0.15	0.4
22	Gas	Thickness of the cap [m]	1.5	1.0	3.0
24		Occupancy of house	0.75	0.65	0.85

3.2 Results of Sensitivity Studies

73. The results of the sensitivity studies are presented in Figures 3.1 to 3.20. These figures show the percentage difference between the results for the sensitivity test and those for the Reference Case in terms of doses to one or more exposed groups.

74. In general, doses to exposed groups represent the sum of doses from more than one release scenario. Some of the parameters selected for sensitivity studies affect doses via several scenarios, but others affect only a single scenario that is a small contributor to the overall dose in the Reference Case. The results of the sensitivity studies are presented in a series of tables in Appendix A (Tables A1.1 to A10.2), showing the effect of varying the parameter values (Table 3.1) on both overall doses and on doses from particular scenarios.

3.2.1 Fire Scenario

75. Both workers and members of the public living near the site may receive doses as a result of fires releasing radionuclides. The assessment methodology includes several pathways for exposure following a fire, including breathing smoke and dust and irradiation from smoke particles. The height of the smoke plume is a factor in determining the concentration of smoke particles deposited on the ground, and hence affects doses to the public through irradiation by dust deposited on the ground and ingestion of crops contaminated by dust. Doses via these pathways are, however, only a small contributor to overall doses to the Public 1 exposed group in the Reference Case (Figure 2.4).

76. Calculated doses for all of the radionuclides considered show sensitivity to plume height for the fire scenario (Tables A1.1 and A1.2). In general, overall doses are not sensitive to changes in plume height (Figures 3.1 and 3.2), because the fire scenario is a small contributor to overall dose (Figure 2.10). This contribution mainly arises from radionuclides with relatively high release fractions during fires, such as H-3, C-14 and Ru-106. Hence, the overall dose shows some sensitivity to plume height for these radionuclides (Figures 3.1 and 3.2).

Figure 3.1: Sensitivity of overall doses to plume height (5 m – Table 3.1, Id 4, Run 1).

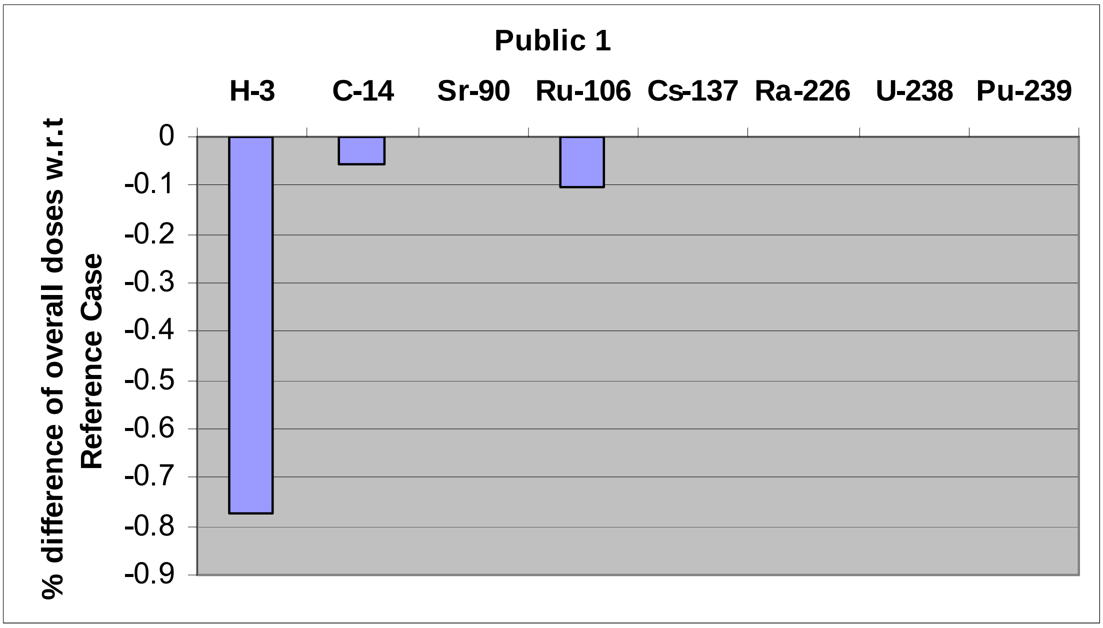
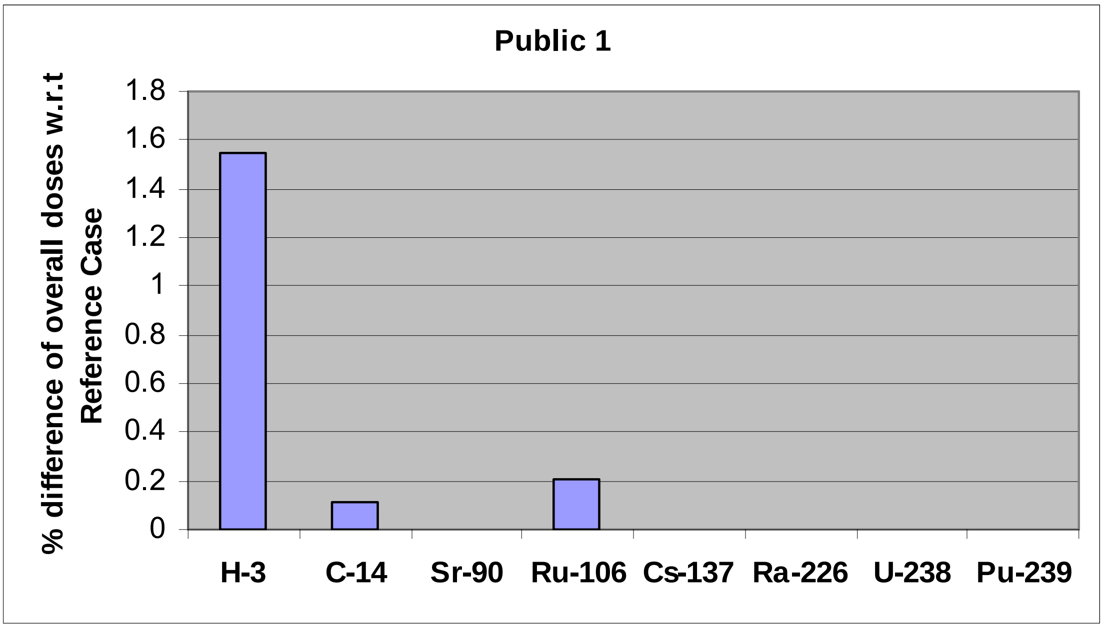


Figure 3.2: Sensitivity of overall doses to plume height (20 m – Table 3.1, Id 4, Run 2).



3.2.2 Aerosol Scenario

77. Both workers and members of the public living near the site may receive doses as the result of aerosol releases during leachate management. The time that members of the public spend outdoors during the aerosol release is a factor in determining doses via this pathway. Doses via this pathway are, however, only a small contributor to overall doses to the Public 1 exposed group in the Reference Case (cf. Figure 2.10).

78. In general, overall doses to the public are not sensitive to the time spent outdoors during aerosol releases (Figures 3.3 and 3.4) because the aerosol pathway is a small contributor to the overall dose.

79. The contribution of the aerosol dose to the overall dose differs between radionuclides because the dose coefficients for inhalation and irradiation from cloudshine differ between radionuclides. Doses from these components are directly related to the time spent in the aerosol plume. Doses from deposited aerosols and other components of the overall dose are also radionuclide-dependent but are not affected by the time spent in the plume.

80. The combined effect of these factors means that varying the time spent in the aerosol plume leads to a non-linear increase in the magnitude of the calculated doses from aerosol exposure compared with the Reference Case (Tables A2.1 and A2.2).

Figure 3.3: Sensitivity of overall doses to time public spends outside during aerosol releases (0.5 hr – Table 3.1, Id 5, Run 1).

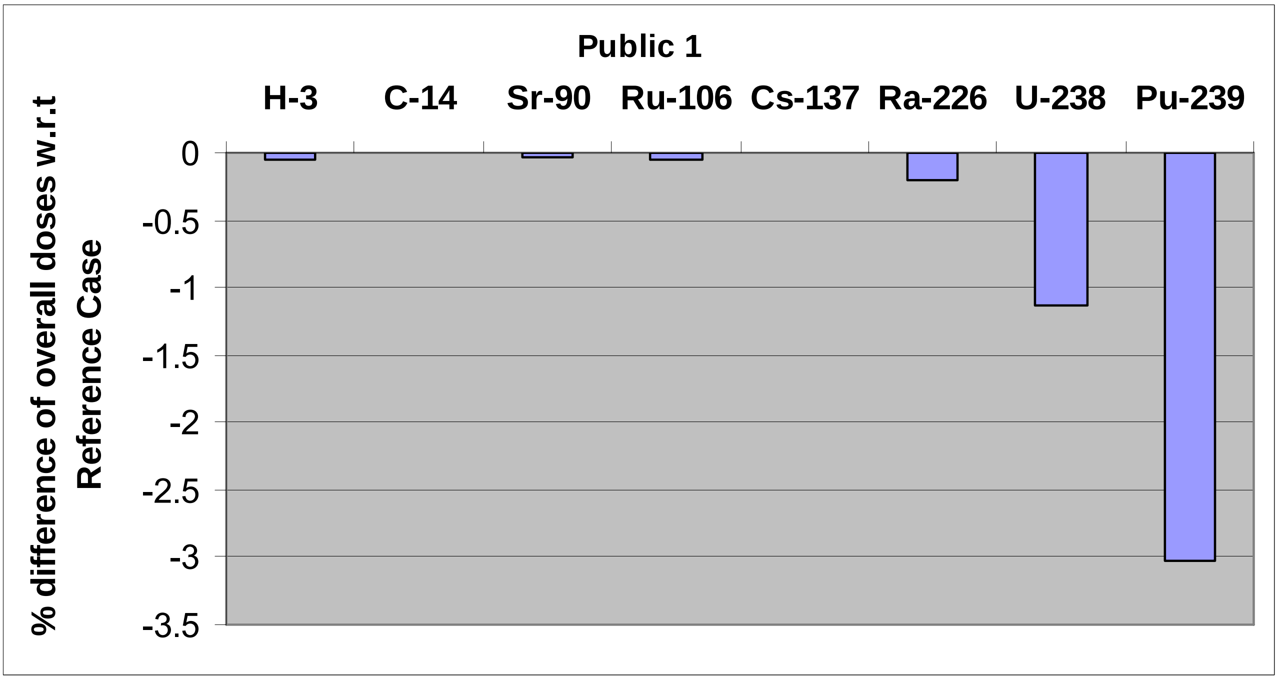
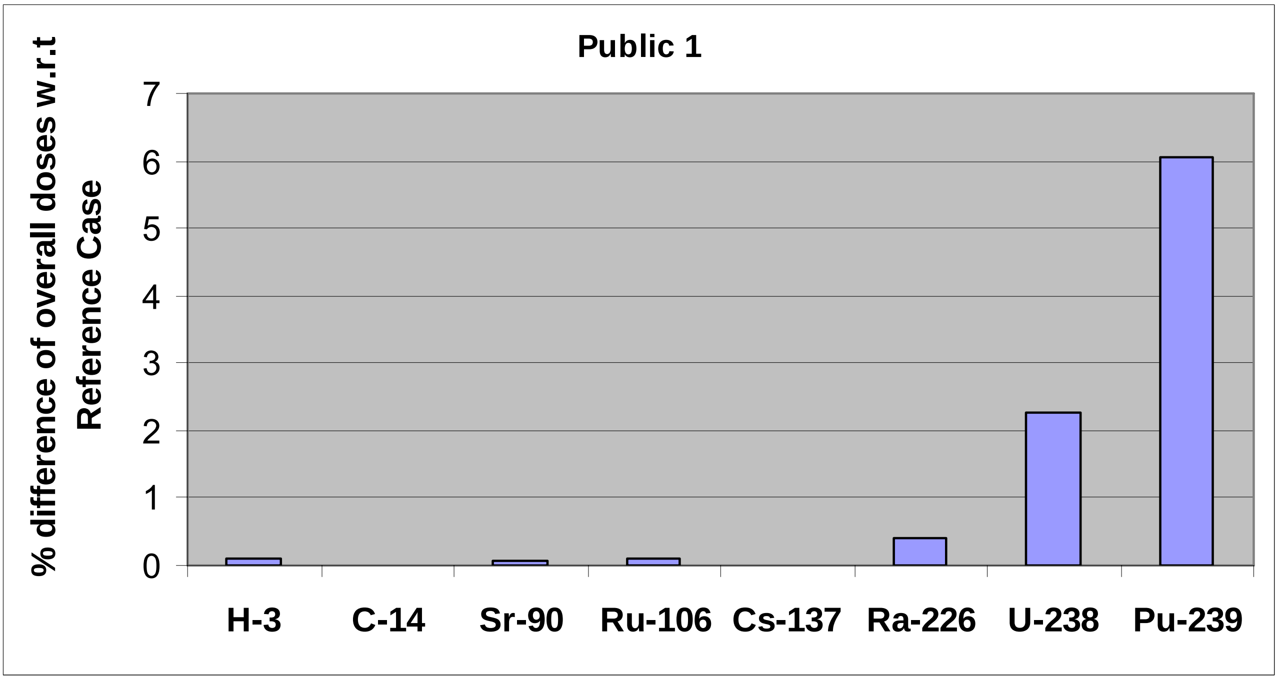


Figure 3.4: Sensitivity of overall doses to time public spends outside during aerosol releases (1 hr – Table 3.1, Id 5, Run 2).



3.2.3 Inadvertent Excavation Scenario

81. Both workers involved in excavating material from the landfill and members of the public living on and / or farming excavated material may receive doses as a result of inadvertent excavation. Sensitivity studies have examined the effect of the volume in which activity is contained and the time of excavation.

82. For the majority of release scenarios, the assessment methodology assumes that the inventory is uniformly distributed throughout the landfill and / or that there is thorough mixing of the leachate so that there is no variation in leachate concentration across the site. In practise, however, it is likely that SPB disposals will be restricted to particular parts of the landfill. There is a possibility that workers may inadvertently intrude into such a location, and therefore be exposed to higher concentrations of radionuclides than would be given by the assumption of homogeneity.

83. Doses to workers involved in inadvertent excavations display the expected sensitivities to changes in the volume containing SPB disposals (Figures 3.5 and 3.6, Tables A3.1 and A3.2). If the volume containing the inventory is doubled, concentrations and hence doses are halved. This is an example where the change in the input value results in a linear change in the calculated doses across the range of radionuclides.

Figure 3.5: Sensitivity of the overall doses to volume of the excavated waste in which the activity is contained (5 m^3 – Table 3.1, Id 12, Run 1).

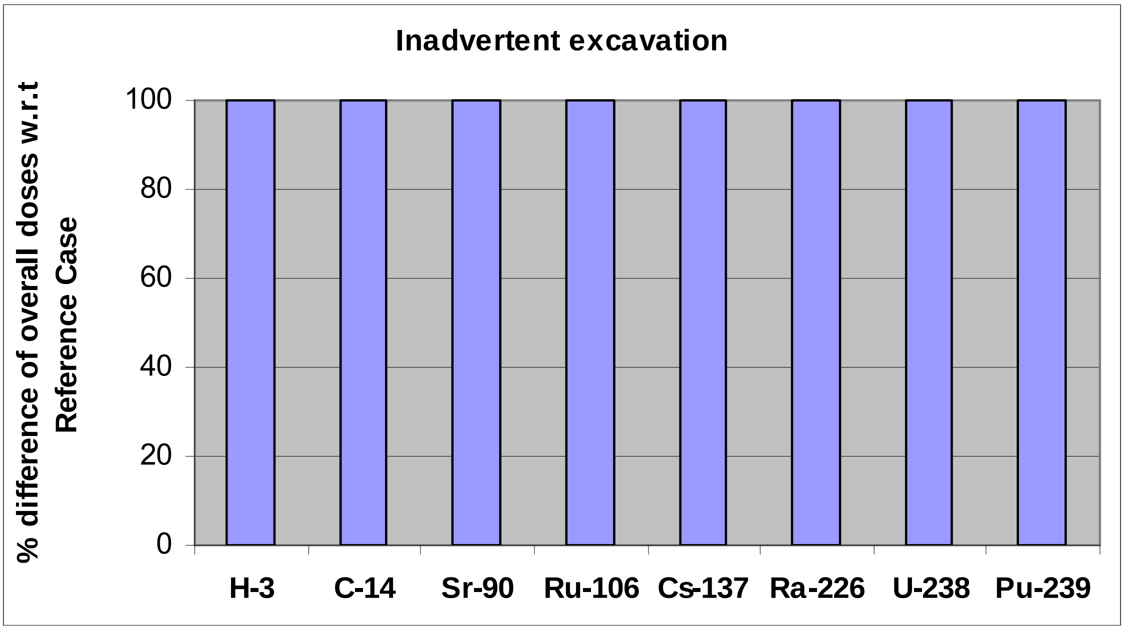
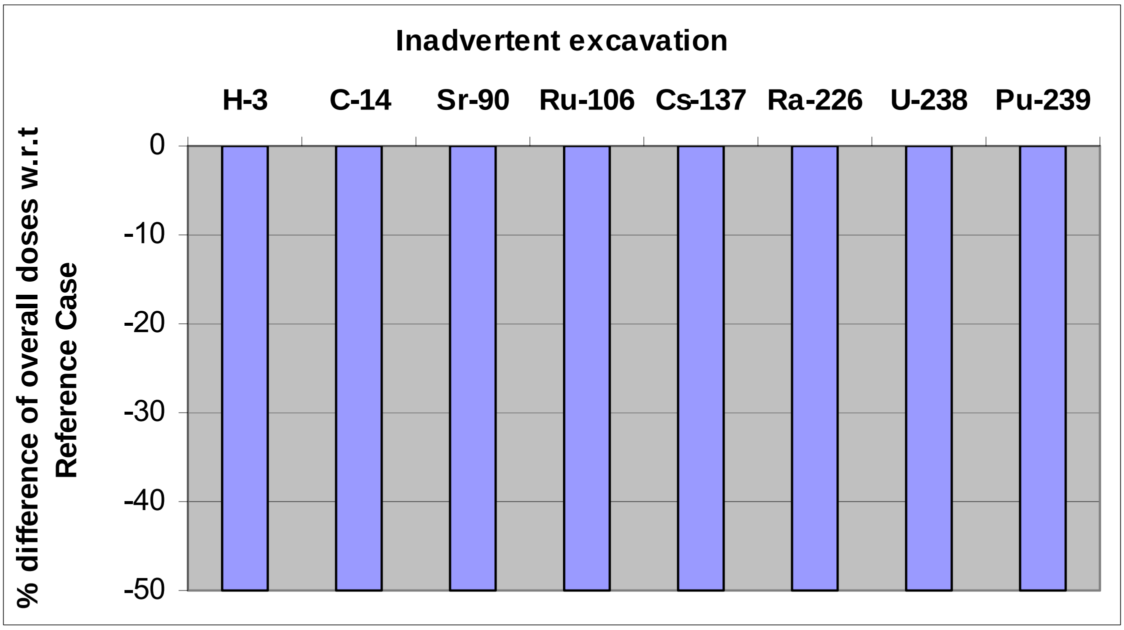


Figure 3.6: Sensitivity of the overall doses to volume of the excavated waste in which the activity is contained (20 m^3 – Table 3.1, Id 12, Run 2).



84. The assessment methodology assumes that inadvertent intrusion does not occur for some period after site closure and capping. This period represents the time during which knowledge of the site is retained and planning conditions prevent activities that would damage the site or lead to exposures. The sensitivity studies assess the effect of changing this period on doses to intruders and to members of the public using the site after intrusion.

85. The effect of the timing of site excavation on doses to the excavator is significant (Figures 3.7 and 3.8). For example, excavating the site 10 yr prior to the date originally selected (20 yr

after closure) results in the dose for H-3 that is nearly 76% higher than the original value (Table A4.1). On the other hand, excavating the site 20 yr after the date selected in the original run results in doses that are nearly 70% lower than the original value (Table A4.2). Changing the time of excavation has a greater effect on the calculated dose for radionuclides with shorter half-lives, because the inventory of these in the landfill at the time of excavation changes more over the interval concerned. While the effect on the dose for Ru-106 seems very significant (change from the original dose of the order of -100% to 95,000% for Runs 1 and 2, respectively), the contribution of this short-lived radionuclide to the overall Worker 2 doses is in fact negligible (Figure 2.5).

86. The timing of site excavation also has a significant effect on calculated doses to members of the public (Public 3) using or occupying land contaminated by excavated material (Figures 3.7 and 3.8: Tables A4.1 and A4.2). The magnitude of the effect is the same for the inadvertent intrusion pathway for the public as it is for the intruder, with the same relationship to radionuclide half-life. In terms of overall doses to the Public 3 exposed group, however, changing the time of excavation has less of an effect on calculated doses, and the magnitude of the effect is not as strongly related to radionuclide half-life. This is because there are other pathways contributing to the overall dose for this exposed group (gas, irradiation, and bathtubbing).

87. Because other release scenarios contribute to the overall doses to the Public 3 exposed group, the timing of inadvertent intrusion is a parameter for which a change to the input value results in a non-linear (generally smaller) change in the calculated doses across the range of radionuclides.

Figure 3.7: Sensitivity of the overall doses to time of excavation (10 yr – Table 3.1, Id 13, Run 1).

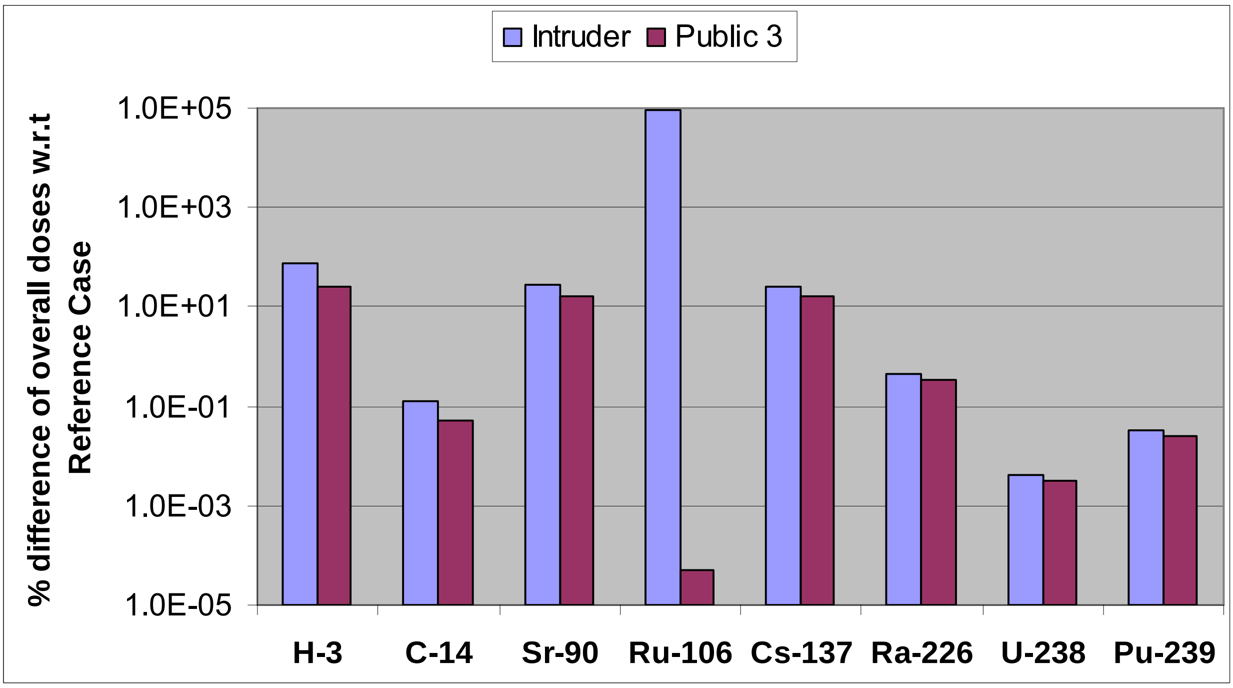
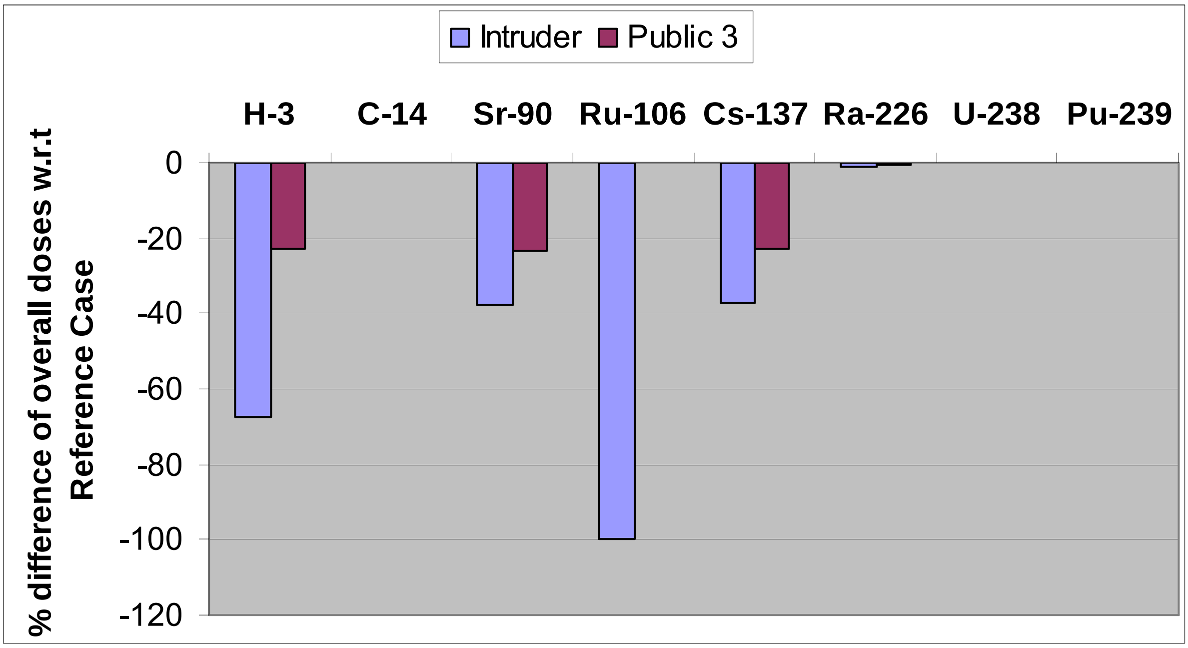


Figure 3.8: Sensitivity of the overall doses to time of excavation (40 yr – Table 3.1, Id 13, Run 2).



3.2.4 Groundwater Scenario

88. As part of the normal evolution scenario, the assessment methodology assumes that leachate will be released from the landfill into an aquifer. Contaminated groundwater in this aquifer or discharged into a water body may lead to doses through a number of pathways, including drinking contaminated water, consumption of fish from contaminated water, and consumption of crops and irradiation from soil following irrigation with contaminated water.

89. Sensitivity studies have examined the effect on calculated doses to the Public 2 exposed group of changes to parameters that affect the release of leachate (time of cap failure and changes to the size of holes in the liner) and radionuclide concentrations in the environment (river length and soil thickness). Changes to the amount of leachate released to groundwater affect the inventory remaining in the landfill and hence the potential dose from other release scenarios. The sensitivity studies therefore also include the effect of changes to these parameters on calculated doses to other exposed groups.

90. Calculated doses via the groundwater pathway show a complex sensitivity to the time of cap failure (Tables A5.1 and A5.2). An increase or decrease in this parameter value leads to both increases and decreases in calculated dose as a result of the change in inventory remaining in the landfill following cap failure, and sorption and half-lives of the radionuclides. For example, earlier cap failure leads to earlier infiltration of water in the landfill and hence earlier seepage through the geological barrier to the groundwater. As expected, since cap failure occurs after the time of the assumed inadvertent intrusion (20 years), calculated doses to intruders and site residents (Public 3) are not affected (Figures 3.9 and 3.10). For relatively short-lived radionuclides that are not significantly sorbed, such as H-3 and Sr-90, earlier release to groundwater results in significantly increased calculated doses via the groundwater pathway (Figure 3.9). For radionuclides that are more strongly sorbed in the groundwater system the time of cap failure does not have a significant effect on calculated doses via this pathway.

91. If the time of cap failure is increased to 200 yr (Figure 3.10), the opposite effects are seen. Later cap failure and hence later releases to groundwater mean that the inventory of short-lived radionuclides reaching the groundwater pathway is reduced, with a consequent reduction in calculated doses.

Figure 3.9: Sensitivity of the overall doses to time of cap failure (50 yr – Table 3.1, Id 14, Run 1).

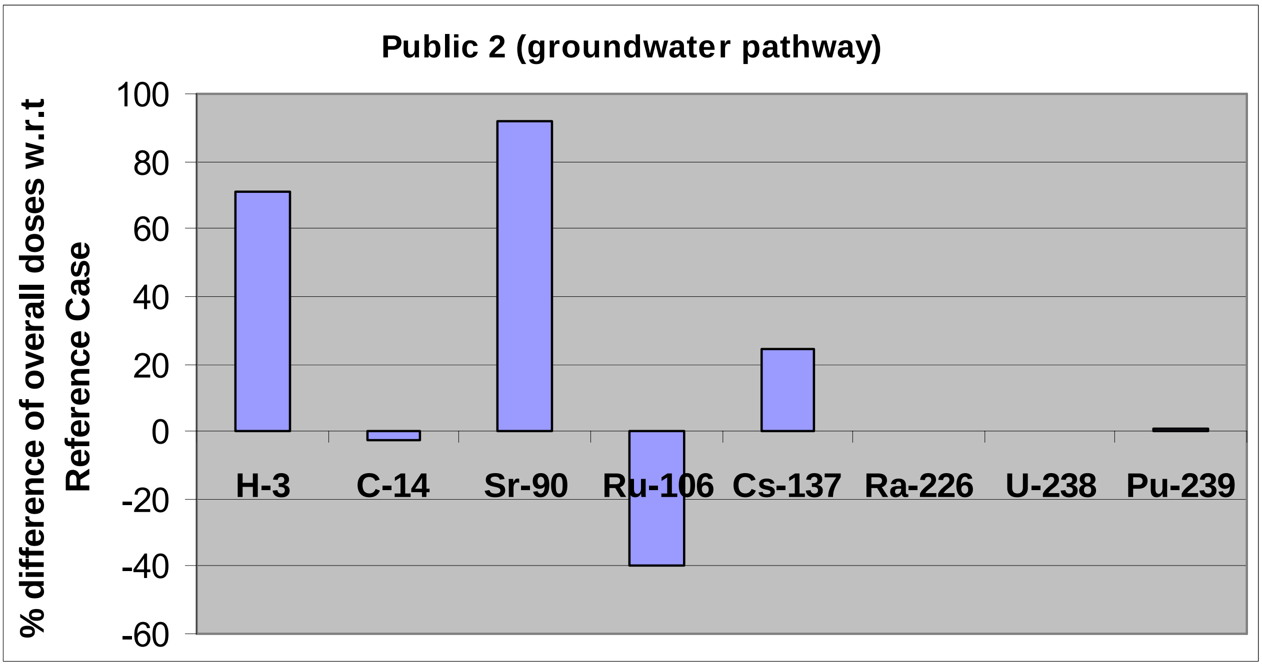
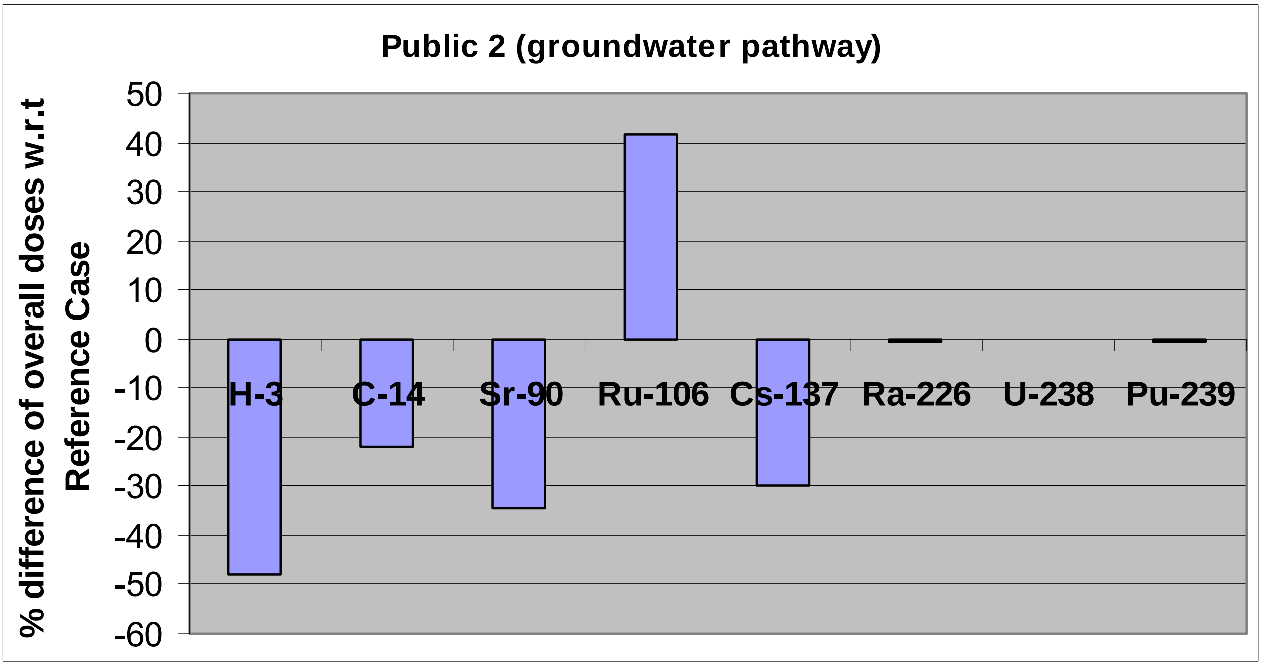


Figure 3.10: Sensitivity of the overall doses to time of cap failure (200 yr – Table 3.1, Id 14, Run 2).



92. Changes to the average area of holes in the liner have an effect on doses via several pathways (Figures 3.11 and 3.12), with a similar pattern of behaviour to changes to the time of cap failure. Decreasing the rate of leachate loss decreases calculated doses for short-lived, poorly sorbed radionuclides, but increases the inventory remaining in the landfill and hence increases calculated doses to intruders and site residents. The magnitude of the changes in dose from changes in the area of the holes for both individual pathways and overall (Tables A6.1 and A6.2) is negligible.

Figure 3.11: Sensitivity of the overall doses to average area of holes in liner (1.875×10^{-4} m^2 – Table 3.1, Id 31, Run 1).

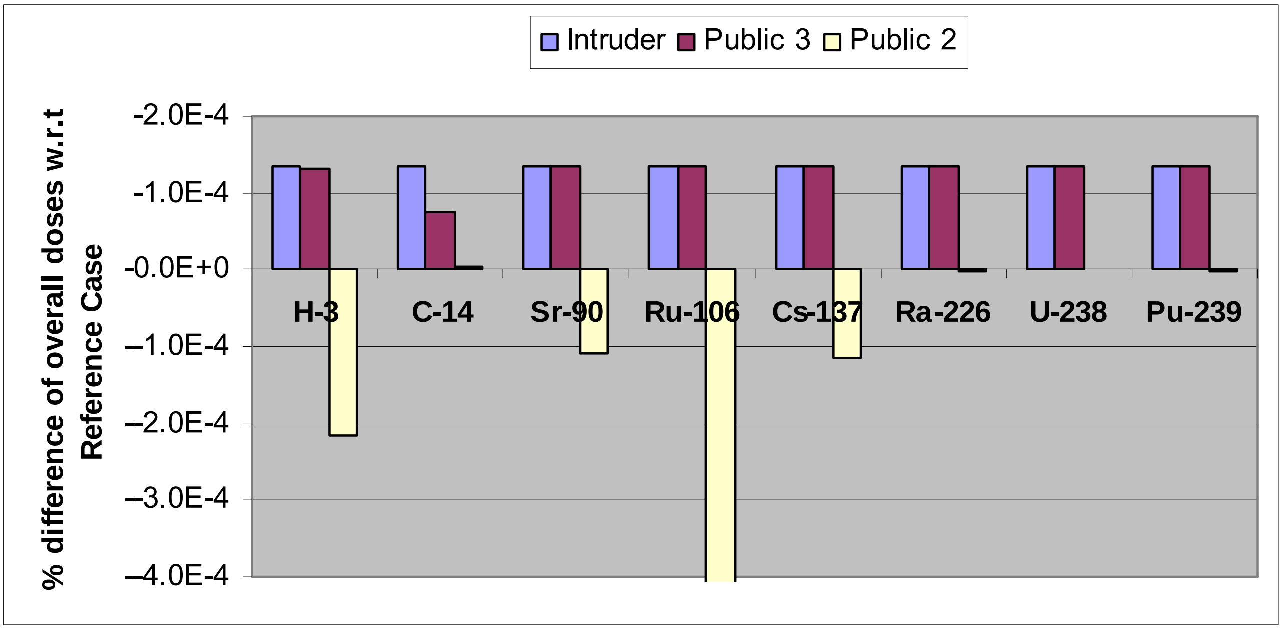
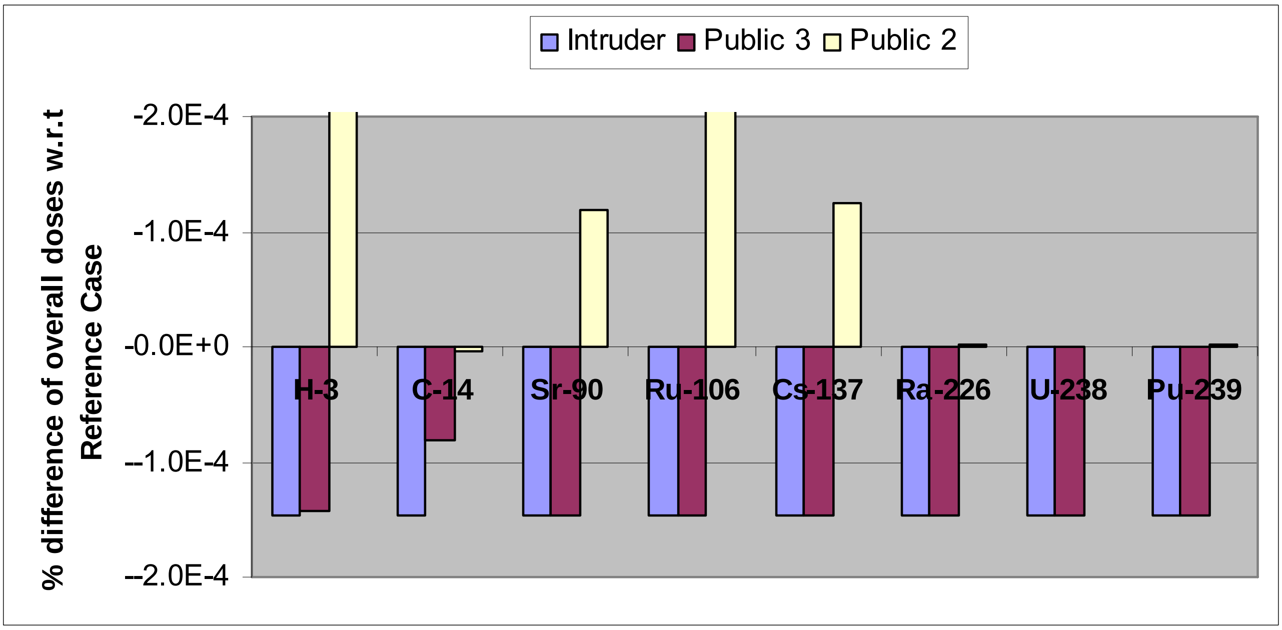


Figure 3.12: Sensitivity of the overall doses to average area of holes in liner (7.5×10^{-4} m^2 – Table 3.1, Id 31, Run 2).



93. The assessment methodology allows for groundwater to discharge to the sea or to a river/lake, and in both cases the groundwater is diluted by an amount proportional to the size of the compartment assumed. The Reference Case calculations assume that discharge is to a river and the amount of dilution is determined by the assumed length of the river.

94. Changes to the length of the river over which radionuclides are assumed to discharge from the geosphere have direct effects on calculated doses from both normal evolution groundwater pathway and the barrier failure release scenario (Tables A7.1 and A7.2). Halving this parameter value results in a doubling of the river water concentration, such that calculated doses are also increased in proportion (Figure 3.13). Similarly, increasing the magnitude of the length of the river by one-third compared to the Reference Case results in calculated doses that are 33% lower (Figure 3.14).

Figure 3.13: Sensitivity of the overall doses to river length over which radionuclides are assumed to be discharged from the geosphere (1.0×10^3 m – Table 3.1, Id 17, Run 1).

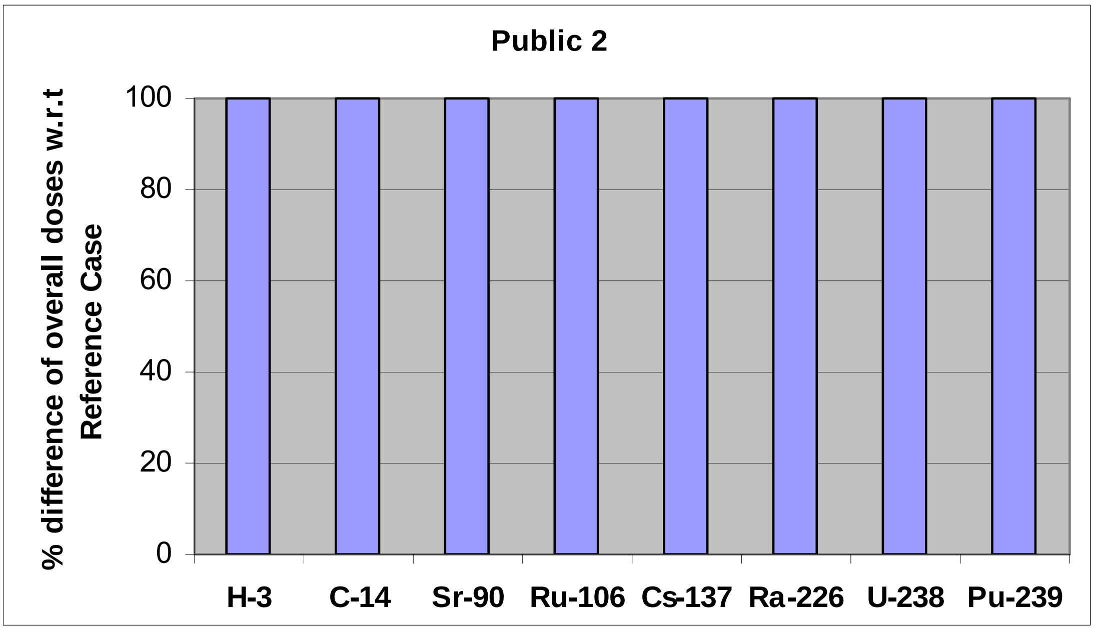
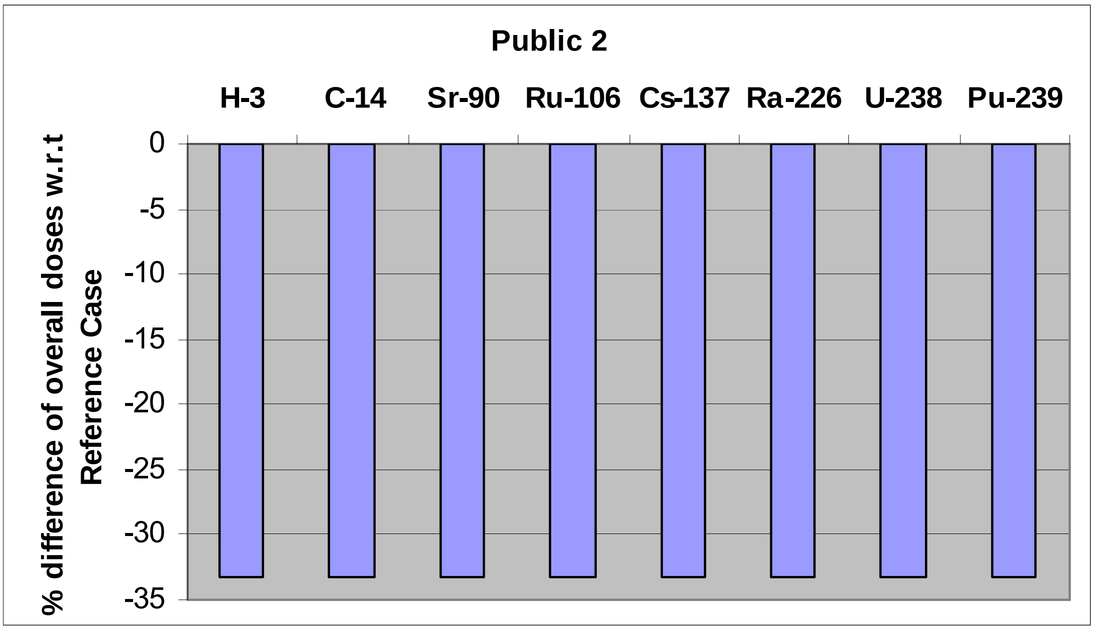


Figure 3.14: Sensitivity of the overall doses to river length over which radionuclides are assumed to be discharged from the geosphere (3.0×10^3 m – Table 3.1, Id 17, Run 2).



95. The assessment methodology allows for a range of uses for contaminated groundwater and surface water, including use for irrigation of crops. The concentration of radionuclides available for uptake by crops is a function of several parameters including the thickness of the soil. Crops may also take up radionuclides following a release of leachate directly to the soil through a bathtubbing event. The Reference Case assumes that surface water contaminated by groundwater releases is used for irrigation but surface water contaminated by direct leachate spillage is not used for this purpose. The sensitivity studies for soil thickness therefore consider only the groundwater and bathtubbing pathways.

96. Because releases of leachate to groundwater for the barrier failure scenario are assumed to take place before releases for the normal evolution scenario, the effect of varying soil thickness is greater for this scenario (Tables A8.1 and A8.2), particularly for relatively short-lived radionuclides.

97. For the bathtubbing scenario, consumption of crops from contaminated soil is the principal pathway. There is therefore a linear relationship between soil thickness and calculated dose (Tables A8.1 and A8.2), with a 37.5% increase in soil thickness reducing calculated doses by 37.5% for all the radionuclides considered. Bathtubbing is, however, only one component of the potential doses to the Public 3 exposed groups so that the effect of changes in soil thickness on overall dose (Figures 3.15 and 3.16) are non-linear and differ between radionuclides.

98. Doses calculated for the groundwater and barrier failure scenarios have several components, including drinking water and fish consumption, which are not affected by soil thickness. Different dose coefficients for different pathways mean that the relative importance of these pathways differs between radionuclides. Changes in soil thickness affect the overall calculated dose for these scenarios (Figures 3.15 and 3.16), but the influence of the other contributing doses reduces the sensitivity to soil thickness.

Figure 3.15: Sensitivity of the overall doses to soil thickness (0.15 m – Table 3.1, Id 36, Run 1).

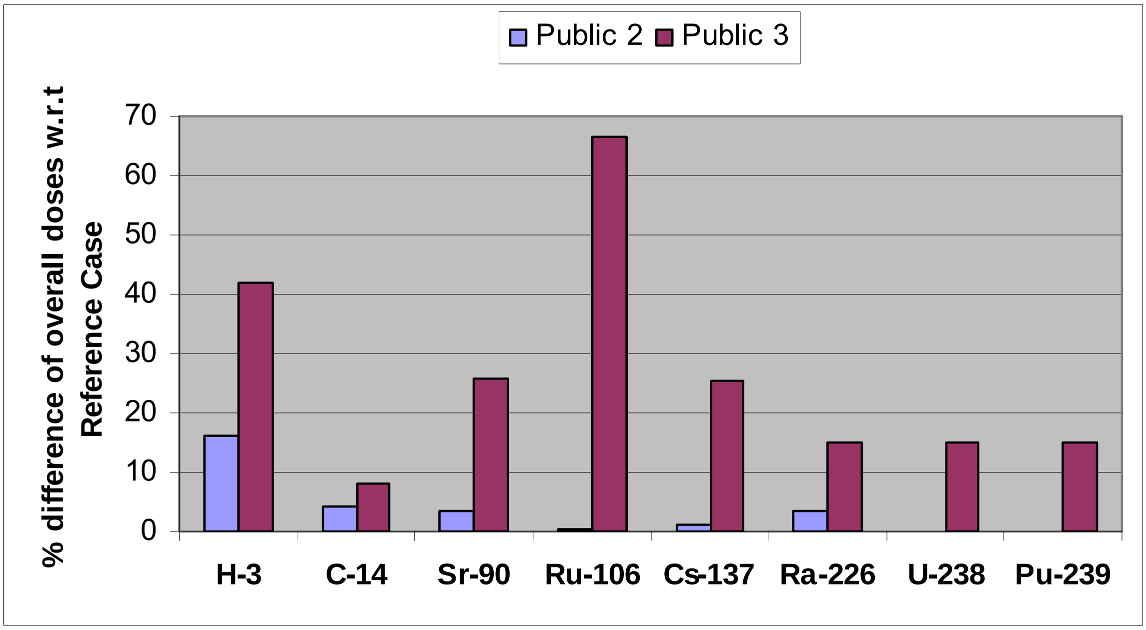
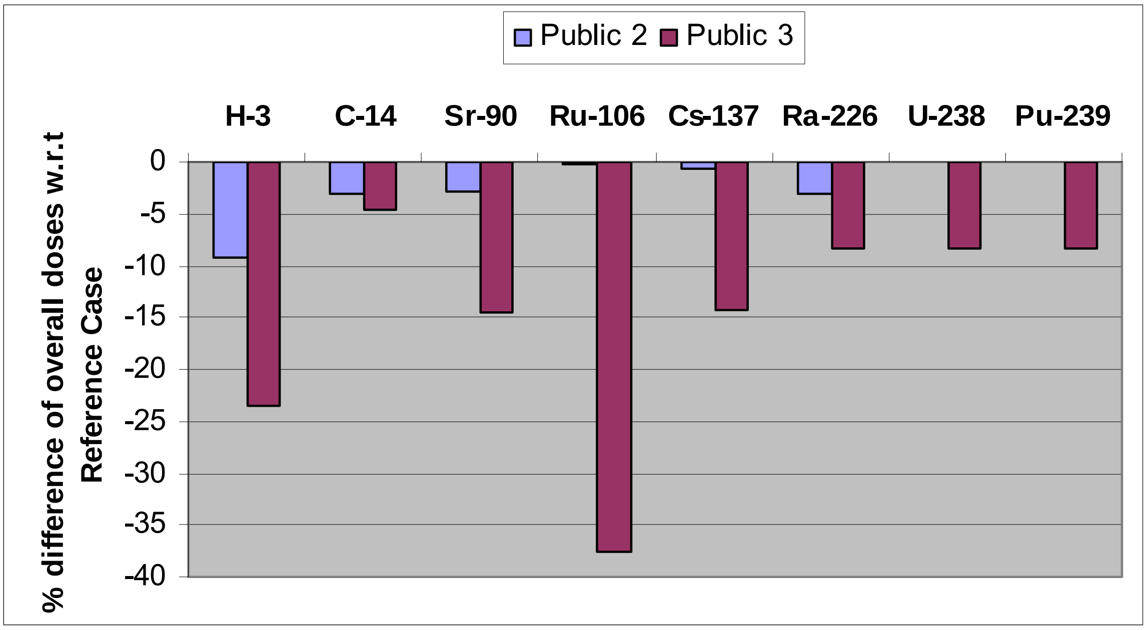


Figure 3.16: Sensitivity of the overall doses to soil thickness (0.4 m – Table 3.1, Id 36, Run 2).



3.2.5 Gas Scenario

99. Gas releases from a landfill may occur as a consequence of the degradation of organic material in the waste or through radioactive decay of certain radionuclides. The assessment methodology assumes that H-3 and C-14 in the inventory may be released in landfill gas and that radon (Rn-222) release is a function of the inventory of Ra-226 and not of any higher members of the decay chain.

100. Several exposed groups may receive a dose from gas releases. During operations, site workers (Worker 1) and members of the public living near the site (Public 1) may receive doses. After closure, site occupants (Public 3) may receive doses from continuing gas releases.

101. Sensitivity studies have been carried out for two parameters that affect doses from gas release. Because of the short half-life of radon, factors that affect the transport distance can affect doses. The thickness of the cap therefore has an effect on radon doses. For all radioactive gases, the amount inhaled is affected by how much time is spent indoors or outdoors. The parameter explicitly used in the assessment methodology is the proportion of time spent indoors, with the time spent outdoors calculated from this.

102. Overall calculated doses are not sensitive to the thickness of the cap (Figures 3.17 and 3.18), because doses via the gas pathway are a small proportion of the overall dose for all of the exposed groups. Tables A9.1 and A9.2, however, show that calculated doses from Ra-226 are very sensitive to decreases in the thickness of the cover (up to 1000%), but are much less sensitive to increase in cover thickness. This is because of the exponential rate at which decay occurs, so that increasing the transit time of radon through the cap has much less of an effect on concentration than decreasing the transit time by the same amount.

103. The thickness of the cover has no effect on calculated doses from the other radioactive gases (H-3 and C-14) because the conservative assumption in the assessment methodology to neglect the effect of the cap in mitigating gas migration.

Figure 3.17: Sensitivity of the overall doses to thickness of the cover (1 m – Table 3.1, Id 22, Run 1).

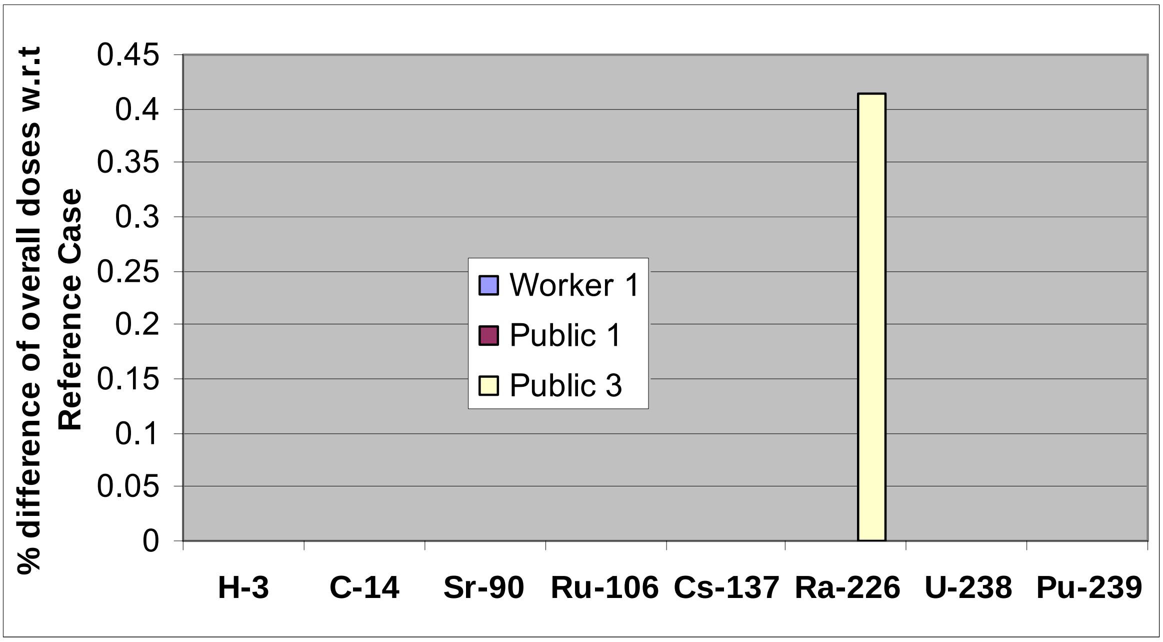
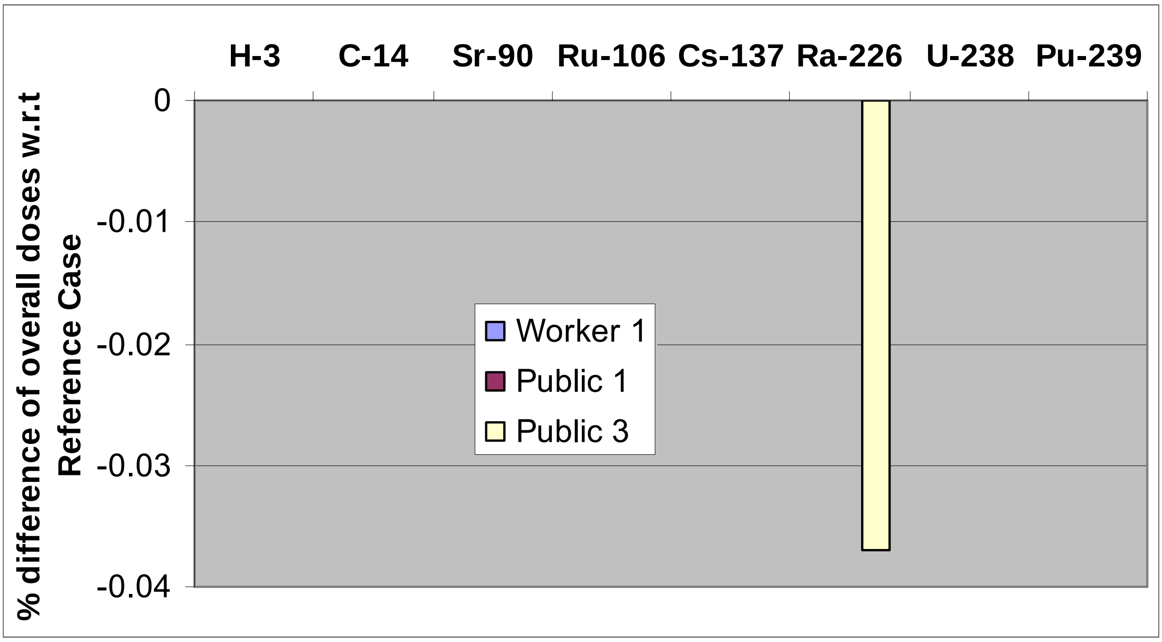


Figure 3.18: Sensitivity of the overall doses to thickness of the cap (3 m – Table 3.1, Id 22, Run 2).



104. Calculated doses to members of the public for the gas pathway are sensitive to the amount of time spent indoors (Tables A10.1 and A10.2), with the expected linear response between time and dose for all three radionuclides of interest. The gas pathway is a small contributor to overall doses, but the overall dose for H-3 is sensitive to this parameter. In the case of C-14 and Ra-226 (parent for radon), there is greater sensitivity to this parameter not because of effects on the gas pathway, but because of effects on the irradiation and bathtubbing pathways that are also affected by the time spent indoors. These effects are even more marked for U-238 and Pu-239, for which the shielding effects of a house affect irradiation from excavated waste.

Figure 3.19: Sensitivity of overall doses to occupancy of house (0.65 – Table 3.1, Id 24, Run 1).

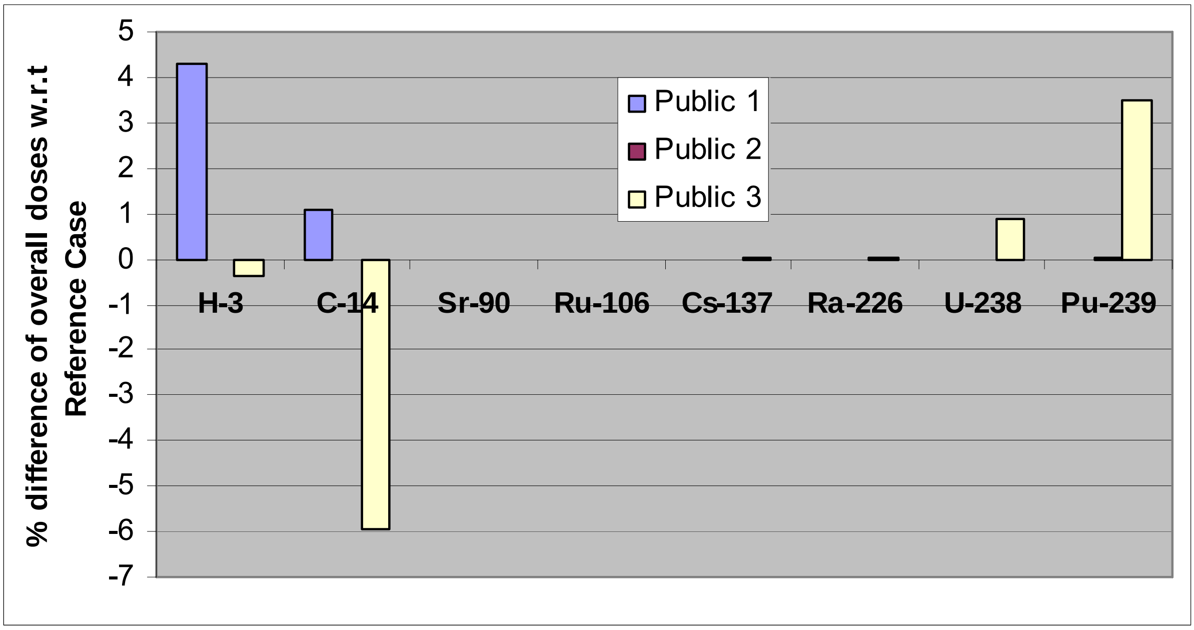
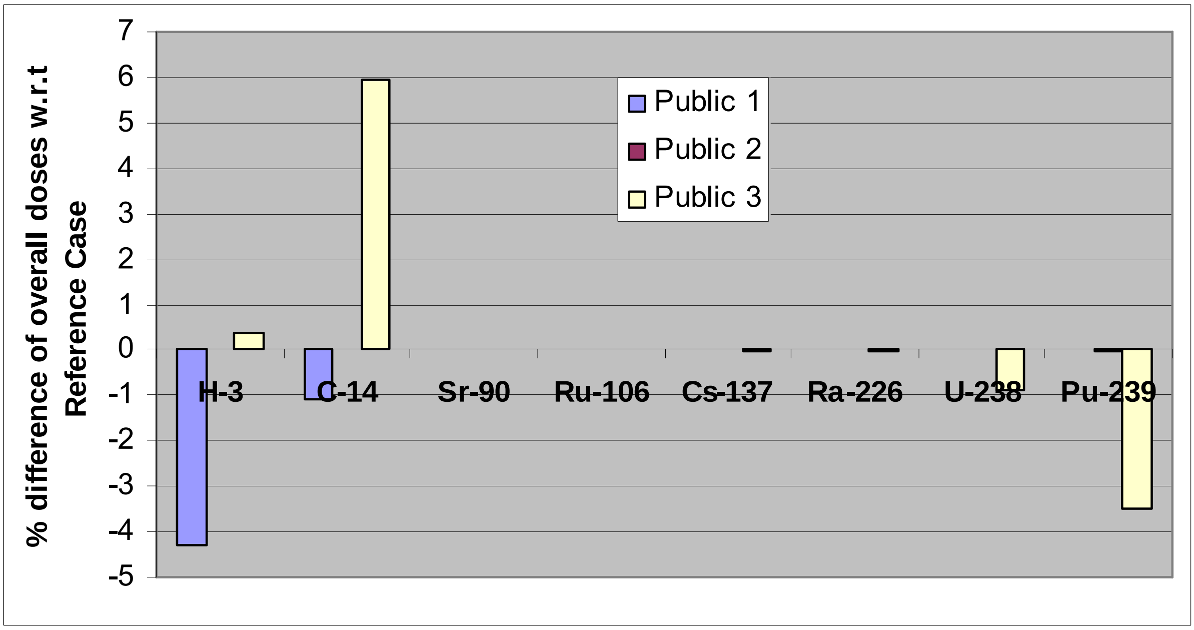


Figure 3.20: Sensitivity of overall doses to occupancy of house (0.85 – Table 3.1: Id 24, Run 2).



4 Radiological Capacity

4.1 Introduction

105. The calculations presented in earlier sections are in the form of specific doses or doses per unit disposal (μSv yr^{-1} MBq^{-1}) for each radionuclide. In this section, these results are used to develop illustrative radiological capacities for the Reference Case for Site A.

106. Radiological capacity is the amount of radioactive material that can be consigned to a site such that the site remains within the appropriate radiological protections criterion. All of the radiological capacity calculations presented here are based on the 20 μSv / yr constraint discussed in the Principles Document (SNIFFER 2005).

107. Because different radionuclides give different specific doses, and because the relative contribution of different radionuclides to doses differs between pathways, there is no single radiological capacity. Rather, there is a range of radiological capacities for a site depending on the mix of radionuclides intended for disposal and on the pathways and scenarios considered.

108. The radiological capacity for a particular waste stream can be calculated from the specific doses and the activity ratios between different radionuclides in the waste stream. However, because SPB disposal is aimed at small users, there is likely to be a number of different users consigning wastes to a particular site so that, in addition to waste stream compositions, it will be necessary to know the relative amounts of different waste streams intended for disposal. The relative amounts of different waste streams can be obtained from existing authorisations and applications to dispose of solid radioactive waste and, where possible, previous disposal records.

109. For the purposes of this Case Study, there is no information available on the composition or relative amounts of different waste streams. Radiological capacities for the Case Study are therefore calculated on a radionuclide-specific basis (i.e., as if all of the LLW for consignment comprised a single radionuclide). Examples of hypothetical mixtures of radionuclides on capacity are presented below.

4.2 Specific Radiological Capacities

110. The maximum radiological capacity for a site is determined by the expected or normal evolution of the site (i.e., those events or scenarios that will occur during site operations or after closure). If the consequences of uncertain events and scenarios are taken into account, the radiological capacity may be lowered. Determining whether these uncertain events should play a role in determining the radiological capacity requires an assessment of their likelihood. The use of a dose criterion of 20 μSv / yr is intended to allow for any reasonable activity to take place on or around the site without the need for control or regulatory oversight after closure (SNIFFER 2005). It may not, however, be reasonable to consider all the possible but unlikely events in determining capacity, or to take account of all the conservatisms inherent in the assessment approach (SNIFFER 2006a).

111. Table 4.1 presents the calculated radiological capacity for each individual radionuclide considered in the Reference Case for the normal evolution scenarios. In order to demonstrate the relative importance of the work and public exposed groups, separate capacities are presented. In each case, the radiological capacity is based on the pathway that leads to the highest specific dose. For example, the radiological capacity for Sr-90 is determined by the irradiation pathway for workers and the groundwater pathway for members of the public. In

practise, because the radiological protection criterion developed for SPB disposals applies to both members of the public and workers who may not be aware of the hazard, the overall radiological capacity is based on the highest specific dose for any of the normal evolution pathways (see paragraph 65).

Table 4.1: Calculated radiological capacity for individual radionuclides for the Reference Case for Site A, based on normal evolution scenarios.

Radionuclide	Radiological Capacity		
	Worker	Public	Overall
H-3	918 TBq	515 TBq	515 TBq
C-14	8 TBq	770 GBq	770 GBq
Cl-36	2,283 TBq	3 TBq	3 TBq
Fe-55	21,645 TBq	14,036 TBq	14,036 TBq
Co-60	11 TBq	46 TBq	11 TBq
Sr-90	104 TBq	16 TBq	16 TBq
Tc-99	1,282 TBq	455 TBq	455 TBq
Ru-106	253 TBq	251 TBq	251 TBq
Ag-108m	448 TBq	61 TBq	61 TBq
Sn-126	595 TBq	357 TBq	357 TBq
I-129	463 TBq	6 GBq	6 GBq
Cs-137	427 TBq	305 TBq	305 TBq
Pb-210	3 TBq	3 TBq	3 TBq
Ra-226	2 TBq	2 TBq	2 TBq
Th-232	152 GBq	181 GBq	181 GBq
U-238	2 TBq	671 GBq	671 GBq
Pu-239	139 GBq	166 GBq	139 GBq
Am-241	174 GBq	207 GBq	174 GBq

112. Table 4.1 does not include capacities for short-lived radionuclides such as Fe-55. The only release events assumed to take place sufficiently soon after emplacement to result in a dose from such radionuclides are fires, which are not certain to occur and therefore not included in the normal evolution scenario. If there are no other operational releases within the first few years of disposal, then there is an almost unlimited capacity for these short-lived radionuclides.

113. Table 4.2 presents the calculated radiological capacity for each individual radionuclide considered in the Reference Case for the events and scenarios that are uncertain to occur. As in Table 4.1, separate capacities are presented for the worker and public exposed groups. In this case, the worker exposed group includes both site workers involved in activities such as remediation and workers involved in intrusion after closure and withdrawal of control.

Table 4.2: Potential radiological capacity for individual radionuclides for the Reference Case for Site A, based on uncertain release scenarios.

Radionuclide	**Radiological Capacity**					
	Worker		**Public**		**Overall**	
H-3	2	TBq	24	TBq	2	TBq
C-14	57	GBq	176	GBq	57	GBq
Cl-36	13	GBq	48	GBq	13	GBq
Fe-55	130	GBq	6	TBq	130	GBq
Co-60	3	MBq	422	GBq	3	MBq
Sr-90	1	GBq	39	GBq	1	GBq
Tc-99	37	GBq	65	GBq	37	GBq
Ru-106	5	GBq	276	GBq	5	GBq
Ag-108m	5	MBq	273	GBq	5	MBq
Sn-126	335	MBq	154	GBq	335	MBq
I-129	380	MBq	4	GBq	380	MBq
Cs-137	3	GBq	30	GBq	3	GBq
Pb-210	53	MBq	2	GBq	53	MBq
Ra-226	73	MBq	8	GBq	73	MBq
Th-232	11	MBq	11	GBq	11	MBq
U-238	143	MBq	59	GBq	143	MBq
Pu-239	10	MBq	11	GBq	10	MBq
Am-241	13	MBq	13	GBq	13	MBq

114. The overall radiological capacities presented in Table 4.2 are generally the most pessimistic capacities for Site A, because they take account of events that might not occur but that might have high consequences. Radionuclides for which the capacity based on these events and scenarios is less than that for the normal evolution scenario are highlighted in Table 4.2. Because the doses to workers are calculated for earlier times than doses to excavators, specific doses to the latter group will be reduced as a result of radioactive decay. The same effect is apparent for other radionuclides but the longer half-lives reduces the magnitude of the difference.

115. Table 4.3 combines the results presented in Tables 4.1 and 4.2 and presents overall radiological capacities for individual radionuclides from all release scenarios.

Table 4.3: Potential radiological capacity for individual radionuclides for the Reference Case for Site A, based on all release scenarios.

Radionuclide	**Radiological Capacity**
H-3	2 TBq
C-14	57 GBq
Cl-36	13 GBq
Fe-55	130 GBq
Co-60	3 MBq
Sr-90	1 GBq
Tc-99	37 GBq
Ru-106	5 GBq
Ag-108m	5 MBq
Sn-126	335 MBq
I-129	380 MBq
Cs-137	3 GBq
Pb-210	53 MBq
Ra-226	73 MBq
Th-232	11 MBq
U-238	143 MBq
Pu-239	10 MBq
Am-241	13 MBq

4.3 Overall Radiological Capacity

116. It is important to stress that the capacities presented in Section 4.3 are for individual radionuclides and are not additive. If more than one radionuclide is present in the wastes to be consigned, the capacities for individual radionuclides will be decreased.

117. The approach used to determine radiological capacities for a mixed waste stream is described in SNIFFER (2006a). Because there is no information on radionuclide ratios for the Case Study, the following examples are hypothetical, and restricted to six key radionuclides.

118. If all the radionuclides are present in the same amount (by activity) then the capacities are reduced in proportion to the number of radionuclides considered. In the case of six radionuclides, the capacities are one-sixth of the values in Table 4.3 (see Table 4.4). It is unlikely that this would apply in practise, but the same principle is used what ever the ratios. Table 4.4 includes three examples of how capacity varies with waste stream composition.

Table 4.4: Illustrative radiological capacity calculations for the Reference Case for Site A, based on all release scenarios and differing proportions of six radionuclides.

Radionuclide	H-3	Co-60	Sr-90	Cs-137	Pb-210	Ra-226	U-238
Nuclide-specific Radiological Capacity	1.7 TBq	3.2 MBq	1.3 GBq	3.1 GBq	52.6 MBq	72.6 MBq	142.8 MBq
Radionuclide Ratios	1	1	1	1	1	1	1
Radiological Capacity	2.8 MBq	2.8 MBq	2.8 MBq	2.8 MBq	2.8 MBq	2.8 MBq	2.8 MBq
Radionuclide Ratios	1	0.1	1	1	1	1	1
Radiological Capacity	13.8 MBq	1.4 MBq	13.8 MBq	13.8 MBq	13.8 MBq	13.8 MBq	13.8 MBq
Radionuclide Ratios	1	0.01	5	2	1	2	1
Radiological Capacity	16.4 MBq	0.2 MBq	81.9 MBq	32.7 MBq	16.4 MBq	32.7 MBq	16.4 MBq
Radionuclide Ratios	1	0	5	2	1	2	1
Radiological Capacity	17.3 MBq		86.3 MBq	34.5 MBq	17.3 MBq	34.5 MBq	17.3 MBq
Radionuclide Ratios	100	0	10	5	0	0	0
Radiological Capacity	11 GBq		1.1 GBq	0.5 GBq			

119. The results in Table 4.4 illustrate the importance of the waste stream composition in determining capacity. For example, because of its high specific dose, the presence of Co-60 in even small proportions has a significant effect on the overall radiological capacity.

120. Even without radionuclides such as Co-60, the overall radiological capacity is generally determined by a single radionuclide. In the case of a single waste stream, this is only a potential issue if there is some uncertainty about the amount of the radionuclide in the inventory. If there is more than one waste stream, however, this may lead to one waste stream dominating disposals and limiting the amount of other waste streams that can be consigned. For example, a waste stream containing Co-60 would dominate the overall capacity and significantly limit the amount of other waste streams that could be consigned.

121. An alternative to calculating capacity for a particular mixture of waste streams is to use radiological capacity calculations to optimise site capacity in terms of the overall activity of LLW consigned. Although this approach might markedly increase the amount of some waste streams consigned, it is likely to require the identification of alternative disposal strategies for waste streams containing particular radionuclides.

122. Optimisation of capacity should not be a single activity, and it will be necessary to review the capacity calculations periodically to ensure that best use is made of the available capacity. Ongoing review of capacity calculations is necessary for several reasons:

- Initial calculations of capacity will be used to establish authorisation conditions. These may require some head-room to ensure that users have flexibility to vary disposals without being in breach of their authorisation. Actual disposals will generally be less than the authorised disposals, and periodic review of radiological capacity will allow this difference to be "recovered".

- Users characterisation of waste streams should develop with time, allowing improvements in the estimates of capacity used and future demands. Waste stream characteristics may also vary with time.

- There are other sources of radioactive material at landfill sites, including both authorised and non-authorised disposals, which may require some head-room in the radiological capacity assigned for SPB disposals. The operational history of the site, the types of wastes consigned and future plans should all be considered in determining the magnitude of this head-room. Additional information on these, which may include acknowledgement of greater levels of uncertainty, may become available at any time up to closure.

- Additional information on site characteristics, operational activities and post-closure plans may become available during the operational period and could affect the calculations of specific doses and radiological capacity. The sensitivity studies presented in Section 3 indicate that for some radionuclides, design and operations can significantly affect calculated doses. In these cases, there may be appropriate authorisation conditions that can be used to help provide assurance that radiological capacities are robust.

5 References

Scotland and Northern Ireland Forum for Environmental Research (SNIFFER) (2005). *Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: Principles Document*. SNIFFER, Edinburgh.

Scotland and Northern Ireland Forum for Environmental Research (SNIFFER) (2006a). *Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: Technical Reference Manual*. SNIFFER, Edinburgh.

Scotland and Northern Ireland Forum for Environmental Research (SNIFFER) (2006b). *Development of a Framework for Assessing the Suitability of Controlled Landfills to Accept Disposals of Solid Low-Level Radioactive Waste: User Manual*. SNIFFER, Edinburgh.

Appendix A Sensitivity Studies

The sensitivity studies described in Section 3 were conducted using a sub-set of eight radionuclides, representative of the range of radionuclides assessed in the Reference Case: H-3, C-14, Sr-90, Ru-106, Cs-137, Ra-226, U-238, and Pu-239. The doses per unit disposal (μSv yr^{-1} MBq^{-1}) for these radionuclides using the Reference Case parameter values are shown in Table A (cf. Table 2.4).

Table A: Doses per unit disposal (μSv yr^{-1} MBq^{-1}) for the Reference Case used as a Basis for the Sensitivity Studies.

	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	6.11E-08	3.37E-06	2.16E-07	1.07E-06	5.27E-08	1.29E-05	1.08E-05	1.62E-04
Gas	2.18E-08	2.49E-06	N/A	N/A	N/A	7.36E-08	N/A	N/A
Aerosol	3.12E-10	6.96E-09	1.92E-07	7.92E-08	4.68E-08	1.14E-05	9.60E-06	1.44E-04
External irradiation	0	3.98E-41	9.14E-22	0	5.65E-20	2.80E-16	9.92E-61	1.42E-17
Fire	3.90E-08	8.70E-07	2.40E-08	9.90E-07	5.85E-09	1.43E-06	1.20E-06	1.80E-05
Worker 2	1.16E-05	3.54E-04	1.48E-02	4.12E-03	6.50E-03	2.75E-01	1.40E-01	1.92E+00
Intruder	*8.71E-07*	*3.52E-04*	*4.96E-03*	*8.14E-17*	*2.25E-03*	*2.70E-01*	*1.40E-01*	*1.91E+00*
Site operations	1.16E-05	3.54E-04	1.48E-02	4.12E-03	6.50E-03	2.75E-01	1.40E-01	1.92E+00
Inadvertent excavation	*8.71E-07*	*3.52E-04*	*4.96E-03*	*8.14E-17*	*2.25E-03*	*2.70E-01*	*1.40E-01*	*1.91E+00*
Public 1	2.65E-07	1.18E-04	2.68E-04	7.36E-05	6.72E-04	2.42E-03	3.51E-04	1.98E-03
Gas	2.84E-08	3.24E-06	N/A	N/A	N/A	9.59E-08	N/A	N/A
Aerosol	2.96E-10	6.97E-09	2.17E-07	7.96E-08	6.57E-08	1.01E-05	8.03E-06	1.20E-04
Fire	3.93E-08	9.46E-07	3.07E-08	1.08E-06	9.86E-09	1.29E-06	1.01E-06	1.50E-05
Spillage	1.97E-07	1.14E-04	2.67E-04	7.25E-05	6.72E-04	2.41E-03	3.42E-04	1.85E-03
Public 2	7.59E-07	4.20E-05	1.08E-05	5.86E-06	5.18E-19	3.35E-06	5.42E-05	4.63E-07
Groundwater	2.79E-08	2.60E-05	1.24E-06	2.56E-12	6.95E-20	1.33E-06	2.98E-05	1.83E-07
Barrier failure	7.59E-07	4.20E-05	1.08E-05	5.86E-06	5.18E-19	3.35E-06	5.42E-05	4.63E-07
Public 3	1.34E-06	2.06E-05	8.42E-04	8.87E-10	2.21E-05	7.32E-04	2.83E-06	1.04E-05
Gas	3.88E-08	9.22E-06	N/A	N/A	N/A	2.71E-07	N/A	N/A
External irradiation	0	1.61E-46	2.73E-27	0	1.70E-25	1.13E-21	4.03E-66	5.77E-23
Inadvertent excavation	4.57E-07	8.84E-06	5.17E-04	4.55E-19	1.37E-05	5.67E-04	2.20E-06	8.10E-06
Bathtubbing	8.41E-07	2.55E-06	3.25E-04	8.87E-10	8.43E-06	1.65E-04	6.33E-07	2.33E-06

The sensitivity studies comprised varying individual parameter values from the Reference Case values (Table 3.1). The results are presented in Tables A1.1 to A10.2 in terms of percentage changes in the specific doses for each change in input parameter values. Specific doses arising from particular pathways may be sensitive to changes in parameter values but, because these doses contribute only a small proportion to the overall dose to an exposed group, overall doses may not show the same sensitivity. The results in Tables A1.1 to A10.2 are therefore presented both for overall doses to each exposed group (Workers 1 & 2, Public 1, 2 & 3) and for the affected pathways.

Table A1.1: Sensitivity to plume height (5 m – Table 3.1, Id 4, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	-7.74E-01	-5.63E-02	-1.20E-03	-1.05E-01	-2.22E-04	-1.32E-03	-5.31E-04	-5.23E-04
Fire	-5.22E+00	-7.02E+00	-1.04E+01	-7.14E+00	-1.52E+01	-2.48E+00	-1.85E-01	-6.88E-02
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A1.2: Sensitivity to plume height (20 m – Table 3.1, Id 4, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	1.55E+00	1.13E-01	2.39E-03	2.10E-01	4.45E-04	2.65E-03	1.06E-03	1.05E-03
Fire	1.04E+01	1.40E+01	2.09E+01	1.43E+01	3.03E+01	4.95E+00	3.71E-01	1.38E-01
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A2.1: Sensitivity to time public spends outside during aerosol releases (0.5 hr – Table 3.1, Id 5, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	-4.90E-02	-2.46E-03	-2.99E-02	-4.48E-02	-2.90E-03	-1.96E-01	-1.14E+00	-3.03E+00
Aerosol	-4.39E+01	-4.16E+01	-3.69E+01	-4.15E+01	-2.97E+01	-4.72E+01	-4.98E+01	-4.99E+01
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A2.2: Sensitivity to time public spends outside during aerosol releases (1 hr – Table 3.1, Id 5, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	9.81E-02	4.92E-03	5.98E-02	8.97E-02	5.80E-03	3.92E-01	2.28E+00	6.06E+00
Aerosol	8.78E+01	8.32E+01	7.39E+01	8.29E+01	5.94E+01	9.44E+01	9.96E+01	9.99E+01
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A3.1: Sensitivity to volume of the excavated waste in which the activity is contained (5 m^3 – Table 3.1, Id 12, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02
Inadvertent Excavation	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A3.2: Sensitivity to volume of the excavated waste in which the activity is contained (20 m^3 – Table 3.1, Id 12, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01
Inadvertent Excavation	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01	-5.00E+01
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0		0	0	0	0

Table A4.1: Sensitivity to time of excavation (10 yr – Table 3.1, Id 13, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	7.56E+01	1.25E-01	2.69E+01	9.52E+04	2.60E+01	4.38E-01	4.24E-03	3.30E-02
Inadvertent Excavation	7.56E+01	1.25E-01	2.69E+01	9.52E+04	2.60E+01	4.38E-01	4.24E-03	3.30E-02
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	2.58E+01	5.38E-02	1.65E+01	4.89E-05	1.61E+01	3.39E-01	3.29E-03	2.57E-02
Inadvertent Excavation	7.56E+01	1.25E-01	2.69E+01	9.52E+04	2.60E+01	4.38E-01	4.24E-03	3.30E-02

Table A4.2: Sensitivity to time of excavation (40 yr – Table 3.1, Id 13, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	-6.76E+01	-2.50E-01	-3.79E+01	-1.00E+02	-3.70E+01	-8.71E-01	-8.48E-03	-6.61E-02
Inadvertent Excavation	-6.76E+01	-2.50E-01	-3.79E+01	-1.00E+02	-3.70E+01	-8.71E-01	-8.48E-03	-6.61E-02
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	-2.31E+01	-1.07E-01	-2.32E+01	-5.13E-08	-2.29E+01	-6.74E-01	-6.59E-03	-5.13E-02
Inadvertent Excavation	-6.76E+01	-2.50E-01	-3.79E+01	-1.00E+02	-3.70E+01	-8.71E-01	-8.48E-03	-6.61E-02

Table A5.1: Sensitivity to time of cap failure (50 yr – Table 3.1, Id 14, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	7.11E+01	-2.36E+00	9.19E+01	-4.01E+01	2.47E+01	3.67E-01	-6.32E-04	4.29E-01
Groundwater	7.11E+01	-2.36E+00	9.19E+01	-4.01E+01	2.47E+01	3.67E-01	-6.32E-04	4.29E-01
Public 3	0	0	0	0	0	0	0	0

Table A5.2: Sensitivity to time of cap failure (200 yr – Table 3.1, Id 14, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	-4.78E+01	-2.17E+01	-3.45E+01	4.15E+01	-2.96E+01	-7.34E-01	-1.46E-03	-8.21E-01
Groundwater	-4.78E+01	-2.17E+01	-3.45E+01	4.15E+01	-2.96E+01	-7.34E-01	-1.46E-03	-8.21E-01
Public 3	0	0	0	0	0	0	0	0

Table A6.1: Sensitivity to average area of holes in liner (1.875×10^{-4} m^2 – Table 3.1, Id 31, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	*1.36E-04*	*1.36E-04*	*1.36E-04*	*1.36E-04*	*1.36E-04*	*1.36E-04*	*1.36E-04*	*1.36E-04*
Inadvertent excavation	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Groundwater	-2.15E-04	3.65E-06	-1.10E-04	-1.24E-02	-1.16E-04	-2.20E-06	-2.02E-09	-2.71E-06
Barrier failure	0	0	0		0	0	0	0
Public 3	1.32E-04	7.52E-05	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04
Gas	0	0			0	0		
Irradiation		1.36E-04	1.36E-04		1.36E-04	1.36E-04	1.36E-04	1.36E-04
Inadvertent excavation	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04
Bathtubbing	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04	1.36E-04

Table A6.2: Sensitivity to average area of holes in liner (7.5×10^{-4} m^2 – Table 3.1, Id 31, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Intruder	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04
Inadvertent excavation	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Groundwater	2.31E-04	-3.92E-06	1.18E-04	1.33E-02	1.24E-04	2.37E-06	2.17E-09	2.91E-06
Barrier failure	0	0	0		0	0	0	0
Public 3	-1.42E-04	-8.09E-05	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04
Gas	0	0				0		
Irradiation		-1.46E-04	-1.46E-04		-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04
Inadvertent excavation	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04
Bathtubbing	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04	-1.46E-04

Table A7.1: Sensitivity to river length river over which radionuclides are assumed to be discharged from the geosphere (1.0×10^3 m – Table 3.1, Id 17, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02
Groundwater	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02
Barrier failure	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02	1.00E+02
Public 3	0	0	0		0	0	0	0

Table A7.2: Sensitivity to river length river over which radionuclides are assumed to be discharged from the geosphere (3.0×10^3 m – Table 3.1, Id 17, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01
Groundwater	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01
Barrier failure	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01	-3.33E+01
Public 3	0	0	0		0	0	0	0

Table A8.1: Sensitivity to soil thickness (0.15 m – Table 3.1, Id 36, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	1.63E+01	4.18E+00	3.55E+00	2.10E-01	1.25E+00	3.36E+00	2.04E-03	1.63E+01
Groundwater	7.53E+00	3.56E+00	2.30E+00	1.01E-01	1.25E+00	3.36E+00	2.00E-03	7.53E+00
Barrier failure	1.63E+01	4.18E+00	3.55E+00	2.10E-01	1.25E+00	3.36E+00	2.04E-03	1.63E+01
Public 3	4.19E+01	8.25E+00	2.58E+01	6.67E+01	2.54E+01	1.50E+01	1.49E+01	4.19E+01
Bathtubbing	6.67E+01	6.67E+01	6.67E+01	6.67E+01	6.67E+01	6.67E+01	6.67E+01	6.67E+01

Table A8.2: Sensitivity to soil thickness (0.4 m – Table 3.1, Id 36, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	-9.16E+00	-3.08E+00	-2.80E+00	-1.18E-01	-7.54E-01	-3.02E+00	-1.58E-03	-9.16E+00
Groundwater	-4.01E+00	-2.96E+00	-2.15E+00	-5.70E-02	-7.55E-01	-3.03E+00	-1.49E-03	-4.01E+00
Barrier failure	-9.16E+00	-3.08E+00	-2.80E+00	-1.18E-01	-7.54E-01	-3.02E+00	-1.58E-03	-9.16E+00
Public 3	-2.36E+01	-4.64E+00	-1.45E+01	-3.75E+01	-1.43E+01	-8.46E+00	-8.37E+00	-2.36E+01
Bathtubbing	-3.75E+01	-3.75E+01	-3.75E+01	-3.75E+01	-3.75E+01	-3.75E+01	-3.75E+01	-3.75E+01

Table A9.1: Sensitivity to thickness of the cap (1 m – Table 3.1, Id 22, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	*6.38E+00*	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0	0	3.50E-13	4.14E-01	0	3.01E-11
Gas	0	0				1.12E+03		
Irradiation		1.38E+14	1.81E+08		4.68E+07	9.56E+06	9.30E+20	5.43E+06

Table A9.2: Sensitivity to thickness of the cap (3 m – Table 3.1, Id 22, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	0	0	0	0	0	0	0	0
Public 2	0	0	0		0	0	0	0
Public 3	0	0	0	0	0	-3.70E-02	0	0
Gas	0	0				-9.99E+01		
Irradiation		-1.00E+02	-1.00E+02		-1.00E+02	-1.00E+02	-1.00E+02	-1.00E+02

Table A10.1: Sensitivity to occupancy of house (0.65 – Table 3.1, Id 24, Run 1).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	4.28E+00	1.10E+00	3.29E-05	0	2.39E-05	1.60E-03	6.55E-06	7.79E-07
Gas	4.00E+01	4.00E+01				4.00E+01		
Aerosol	0	9.32E-03	3.83E-02	0	2.30E-01	3.10E-03	2.69E-04	1.21E-05
Fire	0	4.29E-03	1.69E-02	0	9.58E-02	1.51E-03	1.34E-04	6.04E-06
Public 2	2.45E-06	5.82E-05	1.71E-05	2.12E-06	1.09E-04	2.40E-03	5.79E-04	1.22E-02
Groundwater	1.16E-06	6.61E-05	1.56E-05	1.02E-06	1.09E-04	2.41E-03	5.79E-04	1.21E-02
Barrier failure	2.45E-06	5.82E-05	1.71E-05	2.12E-06	1.09E-04	2.40E-03	5.79E-04	1.22E-02
Public 3	-3.88E-01	-5.96E+00	1.43E-04	6.72E-04	5.17E-03	1.32E-02	8.65E-01	3.48E+00
Gas	-1.33E+01	-1.33E+01				-1.33E+01		
Irradiation		2.77E+01	2.77E+01		2.77E+01	2.77E+01	2.77E+01	2.77E+01
Inadvertent excavation	1.00E-05	5.47E-04	1.43E-04	6.72E-04	5.17E-03	1.82E-02	8.65E-01	3.48E+00
Bathtubbing	1.00E-05	5.47E-04	1.43E-04	6.72E-04	5.17E-03	1.82E-02	8.65E-01	3.48E+00

Table A10.2: Sensitivity to occupancy of house (0.85 – Table 3.1: Id 24, Run 2).

	Percentage difference with respect to original run							
	H-3	C-14	Sr-90	Ru-106	Cs-137	Ra-226	U-238	Pu-239
Worker 1	0	0	0	0	0	0	0	0
Worker 2	0	0	0	0	0	0	0	0
Public 1	-4.28E+00	-1.10E+00	-3.29E-05	0	-2.39E-05	-1.60E-03	-6.55E-06	-7.79E-07
Gas	-4.00E+01	-4.00E+01				-4.00E+01		
Aerosol	0	-9.32E-03	-3.83E-02	0	-2.30E-01	-3.10E-03	-2.69E-04	-1.21E-05
Fire	0	-4.29E-03	-1.69E-02	0	-9.58E-02	-1.51E-03	-1.34E-04	-6.04E-06
Public 2	-2.45E-06	-5.82E-05	-1.71E-05	-2.12E-06	-1.09E-04	-2.40E-03	-5.79E-04	-1.22E-02
Groundwater	-1.16E-06	-6.61E-05	-1.56E-05	-1.02E-06	-1.09E-04	-2.41E-03	-5.79E-04	-1.21E-02
Barrier failure	-2.45E-06	-5.82E-05	-1.71E-05	-2.12E-06	-1.09E-04	-2.40E-03	-5.79E-04	-1.22E-02
Public 3	3.88E-01	5.96E+00	-1.43E-04	-6.72E-04	-5.17E-03	-1.32E-02	-8.65E-01	-3.48E+00
Gas	1.33E+01	1.33E+01				1.33E+01		
Irradiation		-2.77E+01	-2.77E+01		-2.77E+01	-2.77E+01	-2.77E+01	-2.77E+01
Inadvertent excavation	-1.00E-05	-5.47E-04	-1.43E-04	-6.72E-04	-5.17E-03	-1.82E-02	-8.65E-01	-3.48E+00
Bathtubbing	-1.00E-05	-5.47E-04	-1.43E-04	-6.72E-04	-5.17E-03	-1.82E-02	-8.65E-01	-3.48E+00