COMMERCIAL

Environmental Safety Case: Disposal of Low Activity Low-level Radioactive Waste at the Port Clarence Landfill Sites – Report Appendix E

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Appendix E. Environmental Safety Case – Technical Basis {R3}

> "The environmental safety case should include quantitative environmental safety assessments for both the period of authorisation and afterwards. These assessments will need to extend into the future until the radiological risks have peaked or until the uncertainties have become so great that quantitative assessments cease to be meaningful. They should show how radionuclides might be expected to move from the wastes through the immediate physical and chemical environment of the disposal facility and through the surrounding geological formations into and through the environment. After the period of authorisation and while any significant hazard remains, the environmental safety case should explore the consequences not only of the expected evolution of the disposal system, but also of less likely evolutions and events." NS-GRA (UK Environment Agencies, 2009), para 7.2.8

534. This appendix considers the radiological aspects of an Environmental Safety Case (ESC) that supports an application for an Environment Agency Permit, for receipt and disposal of low-level radioactive waste (LLW) at the Port Clarence site, Off Huntsman Drive, Port Clarence, Middlesbrough, Cleveland, TS2 1UE.

E.1. Features, events and processes

535. Analysis of relevant Features, Events and Processes (FEPs) is used in the field of radioactive waste disposal to define relevant assessment scenarios for safety assessment studies. The term scenario is applied here as defined in the glossary, i.e. a postulated or assumed set of conditions and/or events, The set of scenarios selected for the ESC is intended to cover the range of possible situations at Port Clarence - it is not meant to infer a set of possible future conditions as used elsewhere (LLWR Ltd, 2011). For a radioactive waste disposal facility, features would include the characteristics of the system, such as the waste, groundwater and humans; events would include things that may or will occur at some time in the future, for instance intrusion into a waste cell; and processes are mechanisms which have an impact on the features described, such as erosion or groundwater flow.

536. An initial set of scenarios was based on consideration of FEPs that could lead to exposure of people from the IAEA's Improvement of Safety Assessment Methodologies for Near-Surface Disposal Facilities (ISAM) project (IAEA, 2004). This and recent Eden-NE experience with the LLWR safety case, the East Northants Resource Management Facility (ENRMF) ESC and involvement with work on Environment Agency landfill assessment methodologies has been used to supplement the initial set of scenarios.

537. Important features of the Port Clarence Site are described in the rest of this section followed by a summary of the scenarios in SectionE.2. The radiological assessments are presented in three sections dealing with the period of authorisation (Section E.3), site evolution after the period of authorisation (Section E.4) and intrusion events

(Section E.5). The heterogeneity of waste disposals is considered in Section E.6 using an assessment of large items, discrete items and particles. Biota exposure is considered in Section E.7. The scenarios that are considered in the ESC are based on identified events and the assessment models consider the appropriate processes.

538. The mathematical models used for the ESC are based mainly on approaches developed for other recent work:

- an approach for assessing special precaution burials sponsored by the Environment Agencies (SNIFFER, 2006);
- the initial radiological assessment methodology (Environment Agency, 2006a); and,
- models developed for the LLWR safety case (Hicks & Baldwin, 2011).

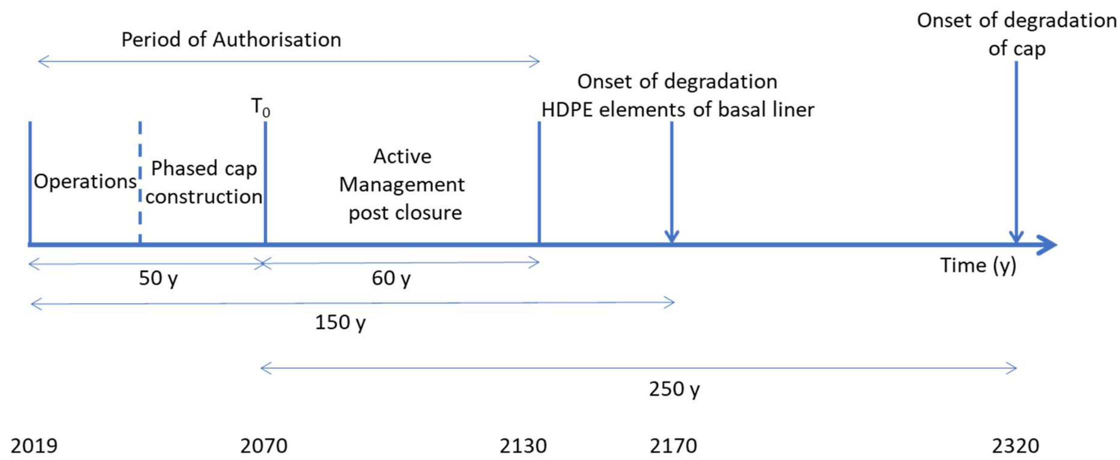
539. The impact of leaching from the landfill to groundwater is assessed using a model implemented in GoldSim (GoldSim Technology Group, 2013). The models are described in Section E.4.3.

E.1.1. Period of authorisation for the Port Clarence site

540. Figure 17 presents the timeline for the Port Clarence site. This timeline is based on dates agreed with the Environment Agency. The figure identifies the Period of Authorisation (POA, the period during which the facility holds a permit), the period of operation, the timing of cap construction and the period of active management following cap construction.

541. The starting point of the calculations presented in this report is indicated as T_0, the time when the site has been filled and the cap constructed. This is the time of closure of the site, also known as the end of the 'operational period'. Decay prior to T_0 has been disregarded as a cautious assumption. It is assumed that for the Port Clarence landfills the future operational period would be 50 years for the purpose of the ESC – noting that planning conditions do not stipulate an end date for use of the site.

Figure 17 Timeline for the Port Clarence site



542. Figure 17 also illustrates assumptions regarding the onset of degradation of the HDPE elements of the basal liner and the cap based on the HRA (MJCA, 2019b). Note that the engineered clay layer will not degrade and it is cautiously assumed that the cap will degrade although this comprises a geosynthetic clay layer .

543. During the "operational period" assumed to last until 1/1/2070 (Operations and Phased cap construction), waste would be disposed to the site and both leachate and landfill gases would be managed. The landfill will then stop receiving waste, cell capping will be completed and the site restoration plan implemented. There is an ecological management and aftercare period of 10 years following restoration (Augean, 2014), but active management controls will continue until it can be confirmed that the site no longer represents a significant risk in terms of environmental pollution or harm to human health. During the active management period, which for the purpose of the radiological risk assessment is assumed to last from 2070 to 2130, leachate and gas would continue to be managed, monitoring would continue and access to the site would be controlled. In practice the active management period is likely to be considerably longer than 60 years. The operational period and the active management period are collectively referred to as the Period of Authorisation. Passive institutional control, e.g. through the presence of land use records, would be expected to continue for some time after the end of the active management period.

544. The assessment considers times up to 100,000 years after installation of the final cap. For most radionuclides the activity concentration in groundwater will have peaked within this timescale.

E.1.2. Landfill dimensions

545. An application has been made to vary the dimensions of the currently permitted landfills. The dimensions used in the ESC are based on those new applications (MJCA, 2019a) and the approximate landfill dimensions are shown in Table 61 and a plan of

the proposed site layout is presented in Figure 3. The plan area of the total site is the sum of the plan areas of the currently constructed cells and the revised site layout.

Table 61 Dimensions of the landfills

Waste type	Phase	Area label	Plan area (m2)	Average depth when built(m)(2)	Void when built
Non-hazardous	Constructed (1)	1 & 2	74,530	11	830,000
		3A-1	10,580	22	230,000
		3B	22,280	20	444,850
	Future phase		150,090	17	2,594,910
Hazardous	Constructed (1)	3	11,470	5	60,720
		4	15,560	8	126,500
		5	18,760	12	221,500
		6 North	6,970	16	111,630
		6 South	9,770	17	166,930
	Future phases including separation structure		139,700	20	2,841,500

546. No distinctions between the disposal cells in each of the landfills are made for the radiological assessment. However, the hazardous landfill and the non-hazardous landfill are treated as separate units for the scenarios concerning landfill gas and a landfill fire.

E.1.3. Barrier engineering

547. A number of engineered barriers contribute to radiological safety:

- construction of a cap to limit infiltration;
- sorption in waste cells by soil and soil-like waste;
- installation of a HDPE liner and an engineered clay barrier below the waste cells prior to waste emplacement to limit water flow and to retard radionuclide transport; and,
- dilution of the flux of released radionuclides when it enters the alluvium and glacial till which underlies the facility.

E.1.3.1. Engineered cap

548. The engineered cap has a layered construction designed to prevent water from entering the waste cells. In accordance with the HRA (MJCA, 2019b), the radiological assessment assumes that the HDPE component of the cap gradually degrades

between 250 years and 1000 years after construction. The water inflow through the intact cap (cap design infiltration) is 31.524 mm y^{-1} (MJCA, 2019b). Until the end of the regulatory control period (period of authorisation) any damage to the cap will be detected and repaired. Gradual degradation of the cap will begin after 250 years and the water inflow will increase to grassland infiltration levels (conservatively estimated to be 202.38 mm y^{-1}) after 1,000 years (MJCA, 2019b).

E.1.3.2. Basal liner and clay barrier

549. A flexible liner is placed at the base of the waste cells in order to limit release of leachate to the underlying engineered clay barrier and hydrogeological features. The HRA assumes that the liner starts degrading after 150 years, the surface area of punctures and tears being assumed to double every 100 years (MJCA, 2019b). The same assumptions are used in the radiological assessment.

550. The efficiency of the HDPE component of the basal liner is determined by the number of defects (pinholes, holes and tears) that are present. When the HDPE component of the basal liner has degraded, outflow through the base of the landfill is controlled by the engineered clay barrier.

551. The engineered clay barrier is 1 m thick under the non-hazardous landfill, has a low hydraulic conductivity of 5.91 10^{-11} (ranging from 4.72 10^{-11} to 9.27 10^{-10} m s^{-1}) (MJCA, 2019b), effectively limiting the water flow through the base of the waste cells. Clay also has advantageous sorption properties, which will delay the migration of certain radionuclides through the barrier. The engineered clay barrier under the hazardous waste landfill is 1.5 m thick. Flows through the clay barrier are low and contaminants are assumed to be distributed between pore water and clay according to a linear equilibrium distribution model. The models cautiously use a depth of 1 m engineered clay for both landfills.

E.1.4. Landfill drainage

552. During the Period of Authorisation, the water level in the cells will be controlled so that it does not exceed 1 m above the base (MJCA, 2019b). Until the end of the period of authorisation, leachate is monitored and managed to ensure that leachate levels do not exceed this depth. Excess leachate is pumped off and either used in the WRP or transported off-site by tanker for treatment and disposal (Augean, 2006).

553. After the end of the period of authorisation, the water level may increase. With an increasing head the potential for leachate flows through the HDPE liner defects to groundwater increases. For the purposes of the groundwater assessment, it has been assumed that the landfill cells are completely saturated and therefore that all of the inventory can potentially be dissolved in pore water. Waste cells are assumed to be homogeneous, saturated and in addition to LLW filled with a mix of soil, soil-like wastes and other hazardous or non-hazardous wastes as appropriate. Soil and soil-like wastes are effective sorption substrates and soil sorption distribution coefficients (Kd) are applied. LLW is not considered an effective sorption substrate and Kd values are set to zero. It has been assumed that all contaminants are available for dissolution

and are partitioned between soil surfaces and pore water according to a linear equilibrium model.

554. The assumptions regarding the partitioning of radionuclides between waste and leachate are conservative since they disregard the sorption on wastes and not all of the radioactive contamination would be on the surface of the waste and hence available for immediate dissolution.

E.1.5. Non-radiological aspects of waste

555. As noted in paragraph 141 the types of wastes to be disposed are not known and will be subject to commercial agreements and subject to permit requirements. The radioactive waste consignments received at the ENRMF fall under the following broad groupings:

- Contaminated soil and sediments (experimental and ex-works);
- Contaminated concrete, bricks and rubble from demolition works;
- NORM in drilling mud, sediments, descaling residues or filter cake;
- Contaminated plastics;
- Contaminated non-recyclable metals;
- Other wastes (clinker, incinerator filter cake, radiochemistry residues, laboratory items, luminising material); and,
- Contaminated hazardous waste (heavy metals, asbestos).

556. It is anticipated that similar wastes will be deposited at the Port Clarence site and that future wastes may also include other lightly contaminated construction and demolition material, redundant plant and equipment and soil from the decommissioning of nuclear sites as well as operational or process waste such as disposable coveralls, plastic wrapping and paper. Similar radioactive waste is also produced by hospitals, manufacturing companies, academic institutions and by the oil and gas industry.

E.1.6. Unsaturated and saturated zones

557. An unsaturated zone underlies the landfill comprising made ground that reflects the history of the site (see Section 2.4). Flow through this zone will be subvertical. A water table exists within the alluvium and glacial till at a depth of 1 m or greater below ground level. Flow occurs within the saturated and is subhorizontal. Significant dilution occurs when radionuclides enter the saturated zone and when the flow discharges to the estuary.

E.1.7. Water abstraction points

558. There are no licensed, deregulated or private water abstraction points located within 2 km of the site (MJCA, 2019a). The compliance point for predictive modelling of

hazardous substances in the HRA is downgradient and directly adjacent to the edge of the discharge area (MJCA, 2019b).

E.2. Identifying scenarios and exposure groups

559. Throughout this report the term “scenario” is used to describe a postulated or assumed set of conditions and/or events that lead to exposure of people to radiation.

560. It is conventional, in assessments of facilities for the disposal of radioactive waste, to assume that management of the site does not persist indefinitely and that knowledge about the location of a disposal facility and the associated hazards is eventually lost. Regulatory guidance requires that an appropriate level of environmental performance should be provided without relying on any human intervention after the end of this management period. The assumption that controls would be lost is cautious as it is likely, for example, that knowledge of the landfill site would persist and that planning controls would continue to govern any redevelopment of the site for some time following closure. Nevertheless, it is assumed in the radiological assessment that management control over the site would cease in or around 2130.

561. The radiological assessment has considered a range of potential scenarios. A review of generic guidance and previous publicly available ESCs identified a set of scenarios that are discussed below, from (IAEA, 2004; SNIFFER, 2006; LLWR Ltd, 2011; Eden NE, 2015a). In cases where a scenario has not been assessed, because it will not or is very unlikely to occur at Port Clarence, the reasons for this are discussed. The scenarios discussed below consider exposure to both workers and members of the public in two separate periods, the period of authorisation and the period afterwards. These scenarios are further divided into two broad categories – those that are likely to occur and those where it is hard to quantify the likelihood of occurrence (unlikely to occur).

562. Doses and risks are assessed to a range of hypothetical exposure groups in order to identify those at greatest risk at a given time from the different scenarios. The present-day and planned land use can inform calculations of the radiological impact during the period of authorisation. For longer timescales, beyond a few decades, it is considered appropriate to use potentially exposed groups (PEGs). These will draw on present-day habit data but it is recognised that different habits could occur in the future.

563. The exposure groups considered for the period of authorisation are workers at the landfill site and members of the public (see Section E.3.3). After the end of the period of authorisation, when active management controls have stopped and only passive controls such as land use records exist, the exposure groups include workers that excavate or analyse material from the site and members of the public living on the site or utilising groundwater abstracted from wells located off-site (see Section E.4). A summary of the scenarios and human exposure groups is given below (Table 62) and in the main text (Table 8, Table 14, Table 20 and Table 24). This lists the period and expectation that the case will occur, the scenario and the exposed group.

564. The ESC presents the dose to an individual who is representative of the most exposed group (known as the representative person, and formerly known as the critical group) and considers the dose to adults, children (aged 10) and infants (aged one) in all scenarios. However, it is recognised that the developing embryo and foetus could also be considered. The LLWR safety case (LLWR Ltd, 2011) references an investigation into the magnitude of exposures to children, infants and the developing embryo and foetus (Thorne, 2006). In that study, it was found that committed effective doses to the embryo, foetus and breast-fed newborn were no larger than those estimated for one year-old infants and ten-year-old children. Similarly, the HPA (HPA, 2008) commented that 'for solid waste disposals it will be generally unnecessary to consider the embryo/foetus/breastfed infant as any increases in doses over those to other age groups will be small compared with the uncertainty in the assessed doses.'

565. Further details of the assumptions and parameters describing the exposed groups used in the radiological assessments are presented in three sections dealing with the period of authorisation (Section E.3), site evolution after the period of authorisation (Section E.4) and intrusions events (Section E.5). Exposure to heterogenous wastes is addressed in Section E.6. Biota exposure is considered in Section E.7.

Table 62 Summary of radiological assessment scenarios considered in the ESC

Scenario	Exposed group
Period of Authorisation – likely to occur	
Direct exposure	Worker
Loose tipping waste	Worker/Member of public
Leachate processing off-site	Treatment worker
	Farming family
	Angler
Release to atmosphere	Member of public
Release to groundwater*	Member of public
Cell excavation*	Worker
Period of Authorisation – unlikely to occur	
Dropped load	Worker
Wound exposure	Worker
Leachate spillage	Farming family
Exposure due to fire	Member of public
Barrier failure*	Member of public
Aircraft impact*	Member of public
After the period of Authorisation – likely to occur	
Recreational user	Member of public
Groundwater abstraction	Farming family
Wildlife exposure	Critical species
Site erosion	Member of public
Inundation from sea*	Member of public
After the period of Authorisation – exposure to heterogenous wastes	
Exposure to discrete items	Worker/ Member of public
Exposure to particles	Worker/ Member of public
Exposure to large objects	Worker/ Member of public

Scenario	Exposed group
After the period of Authorisation – unlikely to occur	
Water abstraction	Farming family
Bathtubbing	Farming family
Gas release and external exposure	Site resident
Borehole drilling	Worker
Trial pit excavation	Worker
Laboratory analyst	Worker
Excavation for housing	Worker/Resident
Excavation for smallholder	Farming family
Site re-engineering*	Worker
Other unlikely events*	

* Not explicitly assessed.

E.3. Radiological impacts during the period of authorisation {R5}

566. The active management phase is assumed to last for 60 years. In reality, the Environmental Permit for the hazardous landfill cannot be surrendered until the Environment Agency consider that the site no longer presents a potential risk to groundwater.

567. The scenarios and relevant exposure pathways considered in this ESC for the period of authorisation are summarised in Table 63. This is followed by a discussion of four other scenarios that are not considered further in the ESC; these are cell excavation, barrier failure, aircraft crash and ground water abstraction during the period of authorisation.

568. The radiological impact of each of the scenarios in Table 35 has been estimated using the approaches described in Sections E.3.3 to E.3.9.

Table 63 Summary of scenarios and exposure pathways during the period of authorisation

Event/scenario	Exposure pathway	Description
Waste receipt, storage, monitoring, transfer and placement	External irradiation	A worker is exposed to external radiation whilst accepting and disposing of waste. Including wound exposure.
Loose tipping of waste	Inhalation of contaminated dust	Contaminated dust is released in the tipping procedure.
Release to atmosphere: operational period	Gas (including radon) inhalation	Members of the public exposed to gases emanating from contaminated material in the landfill.
	Gas flaring	Members of the public exposed to gases emanating from gas burn.
	Fire in non-hazardous waste cell	Members of the public inadvertently inhales and is exposed to cloud of contaminated material released by fire.
Leachate processing off-site: treatment facility worker	External irradiation	The facility worker is exposed to external irradiation from raw sewage and sewage sludge.
	Inhalation of contaminated dust	Dust generated at the facility is inadvertently inhaled during worker activities.
	Ingestion of contaminated dust	Dust generated at the facility is inadvertently ingested during worker activities.

Event/scenario	Exposure pathway	Description
Leachate processing off-site: farming family	Ingestion of food grown on sewage sludge treated land	A farmer ingests contaminated foodstuffs as a result of growing crops on sludge conditioned soil.
	External irradiation	A farmer is exposed to external irradiation from surface layers of sludge conditioned soil.
	Inhalation of contaminated soil	Dust generated from sludge conditioned soil is inadvertently inhaled during farm activities.
	Ingestion of contaminated soil	Dust generated from sludge conditioned soil is inadvertently ingested during farm activities.
Leachate processing off-site: angler	Ingestion of food from the estuary that receives effluent discharges from the sewage treatment facility	An angler ingests fish and crustacea he catches or molluscs he collects in the estuary.
	External irradiation	Contaminated sediments on the bank of the estuary leads to external irradiation of the angler.
Dropped load: site worker and member of the public	Inhalation of contaminated dust	Contaminated dust released from a dropped container or tipper truck is inadvertently inhaled by a site worker and a member of the public.
Wound: site worker	Intake through a contaminated wound	Contaminated dust trapped in wound results in transfer of activity to blood.
Leachate spillage: farming family	Ingestion of food grown on sewage sludge treated land	A farmer ingests contaminated foodstuffs as a result of growing crops on contaminated soil or fish from a contaminated water course.
	External irradiation	A farmer is exposed to external irradiation from surface layers of contaminated soil.
	Inhalation of contaminated soil	Dust generated from contaminated soil is inadvertently inhaled during farm activities.
	Ingestion of contaminated soil	Dust generated from contaminated soil is inadvertently ingested during farm activities.
Fire	Inhalation of contaminated dust	Contaminated dust, gases and vapour released

E.3.1.1. Scenarios not explicitly assessed in the ESC

Exposure from groundwater abstraction

569. A groundwater abstraction scenario has not been included during the period of authorisation. There are no groundwater abstraction points within 2 km of the site and the groundwater beneath the site is subject to saline intrusion from the estuary making the water unsuitable for drinking or for irrigation. The direction of groundwater flow is assumed to be toward the estuary.

570. The ESC includes an assessment of groundwater abstraction after the period of authorisation but the results have not been used to determine the radiological capacity of the site. The expected rise in sea-level will maintain the saline content of water beneath the site and the existing levels of contamination from previous site activities

strongly suggest that it is unlikely that the groundwater pathway will become a credible exposure pathway.

Exposure from cell excavation

571. A scenario involving drilling into the waste during construction of new sampling or leachate wells is not considered because this action would be executed with knowledge of the presence of radioactive material, under the appropriate regulations and with appropriate precautions to minimise doses to the workers. Assessments of landfill excavation after the end of the period of authorisation have been undertaken (see human intrusion in Section E.5).

Exposure from barrier failure

572. The barrier failure scenario was included in the SNIFFER methodology (SNIFFER, 2006) to account for the possibility of damage or defects in the basal liner and a damaged or inadequate geological barrier that could lead to leachate release to groundwater. It assumes that the engineered barriers all fail at the end of operations. The engineered composite liner system at the site includes a clay component and a HDPE component. The gradual degradation and eventual disappearance of the HDPE component of the lining system is modelled. The clay component comprises a natural mineral material and therefore will not degrade other than over geological timescales.

573. It is considered unreasonable to consider this scenario for the Port Clarence landfill sites receiving LLW where the construction, operation and monitoring during the period of authorisation will all reduce the possibility of complete barrier failure in a manner that allows early release of large amounts of leachate. Even if damage did occur, the potential for non-radiological environmental damage from leachate from such a site would ensure that remediation would occur before members of the public were exposed to radiation. The complete barrier failure scenario has not therefore been assessed.

Exposure from aircraft crash

574. An aircraft crash scenario has not been included. There are no airports or military air bases in close proximity to the site. The closest airport is the Durham Tees Valley Airport located about 16.5 km to the southwest of the landfills. The frequency of civil and military aircraft crashes in the UK is very low and it is noted by the IAEA that most aircraft crashes occur within a semicircle of 7.5 km radius from the end of the runway (IAEA, 2002). The scenario is excluded for these reasons.

E.3.2. Presentation of dose assessments

575. The radiological capacity for individual radionuclides present in the LLW is obtained from the results of the ESC and depends on the radiological characteristics of the radionuclide. The radiological capacity is calculated on the basis that the LLW only contains this one radionuclide. The overall radiological capacity is the minimum of the radiological capacities calculated for each of the different assessed scenarios, for that

radionuclide. The results of the assessment are presented as effective doses per MBq disposed ($\mu Sv\ y^{-1}\ MBq^{-1}$).

576. The results of the dose assessments presented in Sections E.3.3 to E.3.8 show the maximum inventory (MBq) that could be disposed of for each radionuclide, and the dose ($\mu Sv\ y^{-1}$) from the radiological capacity. The dose calculated for each radionuclide would only be achieved if that radionuclide was the only one disposed of. Actual waste disposal will be controlled using a sum of fractions approach (see paragraph 388).

577. Estimates of radiological impact based on ‘illustrative inventories’ for waste streams that might be typical of those contributing to the total impact from disposals at the facility have been produced. These estimates are presented in Appendix D.

E.3.3. Direct exposure from waste handling and emplacement

578. It is not intended that waste is stored on-site prior to disposal. Wastes will be placed in a landfill cell as soon as practicable on receipt. If the conditions for the acceptance of low level radioactive waste by Port Clarence are not met, waste may need to be quarantined temporarily while deciding on a course of action. To allow some flexibility for waste delivery times and operational activities the ESC assumes a maximum of 24 h between receipt and disposal.

579. Wastes will be covered by at least 0.3 m thickness of suitable cover after each emplacement campaign or at the end of the working day such that there is no exposed face. Sufficient cover will be used to ensure the dose rate at 1 metre above the waste is less than 2 $\mu Sv\ h^{-1}$.

580. The exposed group considered for quarantine, waste handling and emplacement is landfill workers. Waste handling, emplacement and quarantine will not expose the public near to the site to radiation because there is no line of sight for direct radiation from the quarantine area or landfill void, and site access is controlled. The dose criterion used for this scenario is the site criterion of 1 $mSv\ y^{-1}$ for workers.

E.3.3.1. Waste handling and emplacement

Workers at landfill site

581. Radiation risk to employees from normal handling operations is from external exposure as a result of their occupancy near to a waste package prior to disposal, and external irradiation from the wastes after they have been emplaced and covered. The SNIFFER model does not include this scenario.

582. ENRMF applies a dose rate criterion of 10 $\mu Sv\ h^{-1}$ at 1 m from the LLW package on arrival, and a dose rate criterion of 2 $\mu Sv\ h^{-1}$ at 1 metre above the covered LLW waste, in order to ensure that the occupational dose is considerably less than the dose criterion of 1 $mSv\ y^{-1}$ (Eden NE, 2015a). The same approach will be used at Port Clarence.

583. The proposed authorisation condition for acceptance of LLW is that the dose at 1 m from the package face must be less than 10 µSv h^{-1}. This would be measured by the consignor prior to sending the package and would be checked upon arrival of the package at Port Clarence.

584. The proposed authorisation condition for emplacement of LLW is that wastes will be covered by at least 0.3 m thickness of suitable cover after each emplacement campaign or at the end of the working day such that there is no exposed face. Sufficient cover will be used to ensure the dose rate at 1 metre above the waste is less than 2 µSv h^{-1}. Hence, additional cover may be added if needed. This authorisation condition does not apply to NORM received under the provisions of the exemption.

585. The dose rates proposed for the authorisation conditions were used to estimate the dose to workers at the landfill site. The occupancy times used for the three work activities: receipt and monitoring; transfer and emplacement; and occupancy of the covered area used for the ENRMF ESC are reproduced in Table 64. The operations at the ENRMF are similar to those at Port Clarence, therefore the doses are anticipated to be similar.

Table 64 Estimated annual dose to landfill workers based on dose rate criteria

Work activity	Dose rate (µSv h^{-1})	Occupancy* (h y^{-1})	Estimated annual dose (mSv)
Receipt of waste consignments, including QA and monitoring, etc.	10 2	50 100	0.5 0.2
Transfer and placement of waste in landfill	2	100	0.2
Occupancy of covered waste area	2	100	0.2
Total estimated annual dose			1.1

* Occupancy times assumed for ENRMF (Eden NE, 2015a).

586. Additional ALARA precautions are that dose can be measured directly and managed actively to prevent unnecessary exposure. The dose rate drops quickly with distance from the package and hence the simple precaution of managing occupancy time and distance is practicable.

587. The approach used for the ENRMF was based on the dose from Co-60. To examine the dose from other radionuclides, the approach set out in IAEA report SR44 (IAEA, 2005) for calculating external exposure from waste packages and from waste in a landfill was applied to calculate external doses to workers from handling and occupancy of the covered waste area, respectively.

$$E_{Rn,ext} = e_{Rn,ext} t_e f_d e^{-\lambda_{Rn} t_1} \frac{1 - e^{-\lambda_{Rn} t_2}}{\lambda_{Rn} t_2}$$

where:

- $E_{Rn,ext,}$ is the committed effective dose in a year per unit activity concentration in the material for radionuclide Rn (µSv y^{-1} per Bq g^{-1});
- $e_{Rn,ext}$ is the average effective dose rate per unit activity concentration in the material for radionuclide Rn (scenario dependent) (µSv h^{-1} per Bq g^{-1});
- t_e is the exposure time (h y^{-1})
- f_d is the dilution factor (dimensionless)
- λ_{Rn} is the radioactive decay constant for radionuclide Rn (y^{-1})
- t_1 is the decay time before the start of the scenario (y); and,
- t_2 is the decay time during the scenario (y).

In both exposure cases, it is cautiously assumed that both t_1 and t_2 are zero.

588. External dose from a package was calculated using the external effective dose conversion factors ($e_{Rn,ext}$) from SR44 for transport (IAEA, 2005). External dose to a worker standing on the covered landfill was calculated using the external effective dose conversion factors ($e_{Rn,ext}$) from SR44 for a landfill (IAEA, 2005). The dose conversion factor for a semi-infinite slab was used for radionuclides where external effective dose conversion factors for a landfill were not available in SR44.

589. The waste will be covered by at least 0.3 m of material once it has been emplaced. As such the dose from the landfill was reduced to account for attenuation in the cover material.

590. Consistent with Table 64, it is assumed that a worker spends 50 h y^{-1} close to packages during receipt of waste consignments, including QA and monitoring. The remainder of their exposed working time, 300 h y^{-1} is assumed to be as a result of covered waste. Doses assuming the waste contains 200 Bq g^{-1} of each radionuclide were calculated for both cases and are shown in Table 65.

Table 65 External doses to workers from wastes containing 200 Bq g^{-1}

Radionuclide	Dose from emplacement (mSv)	Dose from handling (mSv)	Total dose (mSv)
H-3	0	0	0
C-14	0	0	0
Cl-36	0	0	0
Ca-41	0	0	0
Mn-54	$1.51\ 10^{-1}$	$3.59\ 10^{-1}$	$5.10\ 10^{-1}$
Fe-55	0	0	0
Co-60	$7.57\ 10^{-1}$	$1.13\ 10^{0}$	$1.89\ 10^{0}$
Ni-59	0	0	0
Ni-63	0	0	0
Zn-65	$1.46\ 10^{-1}$	$2.59\ 10^{-1}$	$4.05\ 10^{-1}$

Radionuclide	Dose from emplacement (mSv)	Dose from handling (mSv)	Total dose (mSv)
Se-79	0	0	0
Sr-90	0	0	0
Mo-93	$2.48\ 10^{-7}$	0	$2.48\ 10^{-7}$
Zr-93	$3.73\ 10^{-8}$	0	$3.73\ 10^{-8}$
Nb-93m	$3.76\ 10^{-8}$	0	$3.76\ 10^{-8}$
Nb-94	$2.62\ 10^{-1}$	$6.71\ 10^{-1}$	$9.33\ 10^{-1}$
Tc-99	$2.49\ 10^{-11}$	$2.13\ 10^{-9}$	$2.15\ 10^{-9}$
Ru-106	$1.63\ 10^{-2}$	$5.29\ 10^{-2}$	$6.92\ 10^{-2}$
Ag-108m	$1.85\ 10^{-1}$	$6.43\ 10^{-1}$	$8.28\ 10^{-1}$
Ag-110m	$5.49\ 10^{-1}$	$1.18\ 10^{0}$	$1.73\ 10^{0}$
Cd-109	$7.25\ 10^{-8}$	$1.22\ 10^{-5}$	$1.23\ 10^{-5}$
Sb-125	$4.05\ 10^{-2}$	$1.59\ 10^{-1}$	$2.00\ 10^{-1}$
Sn-119m	$3.23\ 10^{-48}$	nd	$3.23\ 10^{-48}$
Sn-123	$1.75\ 10^{-3}$	nd	$1.75\ 10^{-3}$
Sn-126	$2.54\ 10^{-1}$	$7.90\ 10^{-1}$	$1.04\ 10^{0}$
Te-127m	$4.88\ 10^{-6}$	$1.64\ 10^{-3}$	$1.64\ 10^{-3}$
I-129	$3.87\ 10^{-20}$	0	$3.87\ 10^{-20}$
Ba-133	$1.43\ 10^{-2}$	$1.04\ 10^{-1}$	$1.18\ 10^{-1}$
Cs-134	$2.26\ 10^{-1}$	$6.48\ 10^{-1}$	$8.74\ 10^{-1}$
Cs-135	0	0	0
Cs-137	$7.95\ 10^{-2}$	$2.45\ 10^{-1}$	$3.25\ 10^{-1}$
Ce-144	$1.08\ 10^{-4}$	$1.44\ 10^{-2}$	$1.45\ 10^{-2}$
Pm-147	$3.06\ 10^{-10}$	$1.71\ 10^{-7}$	$1.71\ 10^{-7}$
Sm-147	0	nd	0
Sm-151	$1.98\ 10^{-68}$	0	$1.98\ 10^{-68}$
Eu-152	$2.37\ 10^{-1}$	$4.67\ 10^{-1}$	$7.04\ 10^{-1}$
Eu-154	$2.73\ 10^{-1}$	$5.24\ 10^{-1}$	$7.97\ 10^{-1}$
Eu-155	$1.07\ 10^{-5}$	$4.95\ 10^{-4}$	$5.06\ 10^{-4}$
Gd-153	$7.67\ 10^{-6}$	$5.34\ 10^{-4}$	$5.42\ 10^{-4}$
Pb-210	$1.76\ 10^{-4}$	nd	$1.76\ 10^{-4}$
Po-210	$1.51\ 10^{-6}$	nd	$1.51\ 10^{-6}$
Ra-226	$5.95\ 10^{-1}$	nd	$5.95\ 10^{-1}$
Ra-228	$1.36\ 10^{0}$	nd	$1.36\ 10^{0}$
Ac-227	$3.66\ 10^{-2}$	nd	$3.66\ 10^{-2}$
Th-228	$8.56\ 10^{-1}$	nd	$8.56\ 10^{-1}$
Th-229	$6.72\ 10^{-2}$	$7.73\ 10^{-2}$	$1.44\ 10^{-1}$
Th-230	$3.42\ 10^{-7}$	nd	$3.42\ 10^{-7}$
Th-232	$5.07\ 10^{-1}$	nd	$5.07\ 10^{-1}$
Pa-231	$1.14\ 10^{-3}$	nd	$1.14\ 10^{-3}$
U-232	$6.97\ 10^{-1}$	$5.78\ 10^{-1}$	$1.27\ 10^{0}$
U-233	$2.56\ 10^{-5}$	$7.31\ 10^{-4}$	$7.57\ 10^{-4}$
U-234	$1.61\ 10^{-8}$	nd	$1.61\ 10^{-8}$
U-235	$1.27\ 10^{-3}$	nd	$1.27\ 10^{-3}$
U-236	$1.91\ 10^{-8}$	$1.22\ 10^{-8}$	$3.13\ 10^{-8}$
U-238	$3.20\ 10^{-6}$	nd	$3.20\ 10^{-6}$
Np-237	$5.62\ 10^{-3}$	$5.36\ 10^{-2}$	$5.92\ 10^{-2}$
Pu-238	$7.56\ 10^{-10}$	$4.77\ 10^{-10}$	$1.23\ 10^{-9}$
Pu-239	$4.60\ 10^{-7}$	$1.84\ 10^{-6}$	$2.30\ 10^{-6}$
Pu-240	$9.26\ 10^{-11}$	$1.31\ 10^{-12}$	$9.39\ 10^{-11}$

Radionuclide	Dose from emplacement (mSv)	Dose from handling (mSv)	Total dose (mSv)
Pu-241	$2.22\ 10^{-7}$	$7.11\ 10^{-10}$	$2.22\ 10^{-7}$
Pu-242	$2.95\ 10^{-12}$	$1.67\ 10^{-12}$	$4.62\ 10^{-12}$
Pu-244	$3.70\ 10^{-2}$	$1.02\ 10^{-1}$	$1.39\ 10^{-1}$
Am-241	$5.48\ 10^{-9}$	$2.40\ 10^{-8}$	$2.95\ 10^{-8}$
Am-242m	$2.02\ 10^{-3}$	$1.36\ 10^{-3}$	$3.38\ 10^{-3}$
Am-243	$1.42\ 10^{-3}$	$2.38\ 10^{-2}$	$2.52\ 10^{-2}$
Cm-242	$4.32\ 10^{-9}$	$2.07\ 10^{-12}$	$4.32\ 10^{-9}$
Cm-243	$9.31\ 10^{-4}$	$1.92\ 10^{-2}$	$2.01\ 10^{-2}$
Cm-244	$5.43\ 10^{-51}$	$1.14\ 10^{-12}$	$1.14\ 10^{-12}$
Cm-245	$9.53\ 10^{-5}$	$3.05\ 10^{-3}$	$3.15\ 10^{-3}$
Cm-246	$2.28\ 10^{-22}$	$3.65\ 10^{-16}$	$3.65\ 10^{-16}$
Cm-248	$1.01\ 10^{-47}$	$9.42\ 10^{-13}$	$9.42\ 10^{-13}$

“nd” indicates that no external effective dose conversion factor was available.

591. For the majority of radionuclides, the total dose is less than 1 mSv y^{-1} for waste at an activity concentration of 200 Bq g^{-1}. The exceptions are Co-60, Ag-110m, Sn-126, Ra-228 and U-232 where the total dose is between 1 and 2 mSv y^{-1}.

592. As discussed in Subsection 7.4.2.2 of the main report, the proposed activity concentration limits vary with radionuclide. The dose to a worker from handling and emplacement of the waste is one of the scenarios used to determine the activity concentration limits.

593. The proposed activity concentration limit for Ra-228 is 100 Bq g^{-1}, therefore the dose to the worker would be lower than 1 mSv. The proposed limits for Co-60, Ag-110m, Sn-126 and U-232 are 200 Bq g^{-1}, therefore the calculated doses given in Table 65 for these radionuclides apply. However, the operational constraints of a 10 µSv h^{-1} dose rate from packages on receipt, and of a 2 µSv h^{-1} dose rate post emplacement, should ensure that the doses to workers do not exceed 1 mSv y^{-1}.

594. The dose to workers during the operational phase can be managed through occupational radiation dose protection practices. Hence the external dose assessment for waste emplacement has not been used to constrain the overall radiological capacity. However, this scenario has been used to inform proposed radionuclide specific activity concentration limits for the wastes (see Section 7.4.2.3 for further details).

595. Additional ALARA precautions are that all wastes are handled by machines and operatives generally do not enter the operational area on foot. On most days the only reason to enter the operational area on foot is for final inspection at the end of the day and health physics monitoring. Workplace monitoring will confirm actual doses and enable dose limitation to be managed.

Members of the public

596. This scenario considers a member of the public who stands at a distance in direct line of sight of a waste package/shipment and hence receives direct radiation exposure. This can be estimated by considering the waste as a single point source with a 10

µSv h^{-1} dose rate at 1 m, and assuming that the member of the public is located 50 m from the waste. The dose rate at 50 m can be estimated from:

$$D_1 = D_2 \cdot \frac{{X_2}^2}{{X_1}^2}$$

where:

- D_1 and D_2 are dose rate at positions 1 and 2 (µSv h^{-1}); and,
- X_1 and X_2 are dose rate at positions 1 and 2 (µSv h^{-1}).

597. This gives an estimated maximum dose rate at 50 m of 4 10^{-3} µSv h^{-1}. If the person stands in that location for 8 hours per day and there is waste at the maximum activity in that location every day then the person would receive 12 µSv y^{-1}. Under the same assumptions but with a 100 m distance to the person, the maximum estimated dose would be 3 µSv y^{-1}.

E.3.4. Exposure from loose tipping of waste

598. The assessment of doses from waste released to atmosphere following the loose tipping of contaminated waste during the operational phase is based on the methodology developed by the IAEA (IAEA, 2005). Members of the exposed group are assumed to be adult, a child or an infant and to be exposed as a result of inhalation of contaminated dust.

599. Exposure of the public to dust has been calculated under the following assumptions:

- a waste tipper tips 15 tonnes of solid waste;
- the tipper is filled with a dry solid;
- the distance to the nearest public is 50 m and the wastes are exposed for 30 minutes after each consignment;
- there are 80 consignments of tipped waste each year;
- the waste is immediately covered after each tip; and,
- the ambient dust loading at the location of the public is comprised of dust from the waste mixed with other clean dust, and the mixing is represented by a dilution factor.

600. The inhalation dose was determined as follows:

$$Dose_{Rn,inh} = D_{Rn,inh} \cdot \mathrm{T} \cdot A \cdot Dil \cdot CF \cdot B \cdot M_{inh}$$

where:

- $\mathrm{Dose}_{\mathrm{Rn},inh}$ is the dose from inhalation of dust (Sv y^{-1});
- $D_{Rn,\mathrm{inh}}$ is the radionuclide specific dose coefficient for inhalation (Sv Bq^{-1});

- T is the exposure time (h y^{-1});
- A is the activity concentration in contaminated waste (Bq g^{-1});
- Dil is the total dilution factor for the fraction of dust inhaled;
- CF is the concentration factor in fine dust
- B is the inhalation rate (m^3 h^{-1}) see Table 67; and,
- M_{inh} is the dust loading (g m^{-3}).

Table 66 Parameters for inhalation of dust used by the IAEA SR44 model

Parameter	Units	Value	Description
CF		4	Concentration factor in fine dust
M_{inh}	g m^{-3}	1 10^{-4}	Dust loading
Dil		0.01	Total dilution factor

Table 67 Worker and public habit data for exposure to dust and gas: applicable during the Period of Authorisation

Parameter	Units	Value	Comment
Inhalation rate – worker	m^3 h^{-1}	1.2	
Inhalation rate – adult	m^3 h^{-1}	1.0	
Inhalation rate – child	m^3 h^{-1}	0.64	
Inhalation rate – infant	m^3 h^{-1}	0.22	
Time in plume – worker	h y^{-1}	880	4 hours per day, 220 working days
Time in plume – public	h y^{-1}	1753.2	4.8 hours per day, 365.25 days

E.3.4.1. Doses from dust released by loose tipping

601. Calculations were performed initially using an activity concentration of 200 Bq g^{-1} in disposed waste. The estimated dose to workers (mSv y^{-1}) is always much higher than that to members of the public (µSv y^{-1}, see Table 68. Applying a dose constraint of 1 mSv y^{-1} to workers and 0.3 mSv y^{-1} to members of the public, Table 68 shows that activity concentrations in loose tipped waste will need to be less than 200 Bq g^{-1} for some radionuclides. The dose to workers is the limiting dose.

602. Applying the site criterion of 1 mSv y^{-1} for workers, a maximum activity concentration for loose tipped waste was derived (last column of Table 68). Note that compliance with the Paris Convention will also limit activity concentrations in disposed waste (NEA, 2017). The Paris Convention includes lower activity concentrations than those presented in Table 68 for H-3, C-14, Co-60, Sr-90, Tc-99 and Cs-137 (see Table 4).

603. The dose to a worker from loose tipped waste is one of the scenarios used to determine the radionuclide activity concentration limits for loose tipped waste. These radionuclide activity concentrations will be applied using the sum of fractions approach (see paragraph 120).

Table 68 Doses estimated from loose tipping waste at 200 Bq g^{-1} and the maximum activity concentration for loose tipped waste that meets 1 mSv y^{-1} to a worker

Radionuclide	Worker (mSv y^{-1})	Public (µSv y^{-1})			Maximum activity concentration (Bq g^{-1})
		Adult	Child	Infant	
H-3$	$9.98\ 10^{-6}$	$8.32\ 10^{-6}$	$7.78\ 10^{-6}$	$7.04\ 10^{-6}$	$4.54\ 10^{6}$
C-14$	$2.23\ 10^{-4}$	$1.86\ 10^{-4}$	$1.52\ 10^{-4}$	$1.20\ 10^{-4}$	$2.03\ 10^{5}$
Cl-36	$2.80\ 10^{-4}$	$2.34\ 10^{-4}$	$2.05\ 10^{-4}$	$1.83\ 10^{-4}$	$1.62\ 10^{5}$
Ca-41	$6.91\ 10^{-6}$	$5.76\ 10^{-6}$	$6.76\ 10^{-6}$	$4.22\ 10^{-6}$	$6.55\ 10^{6}$
Mn-54	$5.76\ 10^{-5}$	$4.80\ 10^{-5}$	$4.92\ 10^{-5}$	$4.36\ 10^{-5}$	$7.86\ 10^{5}$
Fe-55	$2.96\ 10^{-5}$	$2.46\ 10^{-5}$	$2.87\ 10^{-5}$	$2.25\ 10^{-5}$	$1.53\ 10^{6}$
Co-60$	$1.19\ 10^{-3}$	$9.92\ 10^{-4}$	$8.19\ 10^{-4}$	$6.05\ 10^{-4}$	$3.80\ 10^{4}$
Ni-59	$1.69\ 10^{-5}$	$1.41\ 10^{-5}$	$1.21\ 10^{-5}$	$1.06\ 10^{-5}$	$2.68\ 10^{6}$
Ni-63	$4.99\ 10^{-5}$	$4.16\ 10^{-5}$	$3.48\ 10^{-5}$	$3.03\ 10^{-5}$	$9.07\ 10^{5}$
Zn-65	$8.45\ 10^{-5}$	$7.04\ 10^{-5}$	$7.78\ 10^{-5}$	$7.04\ 10^{-5}$	$5.36\ 10^{5}$
Se-79	$2.61\ 10^{-4}$	$2.18\ 10^{-4}$	$1.78\ 10^{-4}$	$1.41\ 10^{-4}$	$1.73\ 10^{5}$
Sr-90$	$6.20\ 10^{-3}$	$5.17\ 10^{-3}$	$3.74\ 10^{-3}$	$2.88\ 10^{-3}$	$7.30\ 10^{3}$
Mo-93	$8.83\ 10^{-5}$	$7.36\ 10^{-5}$	$5.73\ 10^{-5}$	$4.08\ 10^{-5}$	$5.13\ 10^{5}$
Zr-93	$9.60\ 10^{-4}$	$8.00\ 10^{-4}$	$1.99\ 10^{-4}$	$4.51\ 10^{-5}$	$4.72\ 10^{4}$
Nb-93m	$6.91\ 10^{-5}$	$5.76\ 10^{-5}$	$5.12\ 10^{-5}$	$4.58\ 10^{-5}$	$6.55\ 10^{5}$
Nb-94	$1.88\ 10^{-3}$	$1.57\ 10^{-3}$	$1.19\ 10^{-3}$	$8.45\ 10^{-4}$	$2.41\ 10^{4}$
Tc-99$	$4.99\ 10^{-4}$	$4.16\ 10^{-4}$	$3.48\ 10^{-4}$	$2.60\ 10^{-4}$	$9.07\ 10^{4}$
Ru-106	$2.53\ 10^{-3}$	$2.11\ 10^{-3}$	$1.86\ 10^{-3}$	$1.62\ 10^{-3}$	$1.79\ 10^{4}$
Ag-108m	$1.42\ 10^{-3}$	$1.18\ 10^{-3}$	$9.01\ 10^{-4}$	$6.12\ 10^{-4}$	$3.19\ 10^{4}$
Ag-110m	$4.61\ 10^{-4}$	$3.84\ 10^{-4}$	$3.69\ 10^{-4}$	$2.89\ 10^{-4}$	$9.83\ 10^{4}$
Cd-109	$3.11\ 10^{-4}$	$2.59\ 10^{-4}$	$2.87\ 10^{-4}$	$2.60\ 10^{-4}$	$1.46\ 10^{5}$
Sb-125	$4.98\ 10^{-4}$	$4.15\ 10^{-4}$	$3.55\ 10^{-4}$	$2.88\ 10^{-4}$	$9.10\ 10^{4}$
Sn-119m	$8.45\ 10^{-5}$	$7.04\ 10^{-5}$	$6.35\ 10^{-5}$	$5.56\ 10^{-5}$	$5.36\ 10^{5}$
Sn-123	$3.11\ 10^{-4}$	$2.59\ 10^{-4}$	$2.46\ 10^{-4}$	$2.18\ 10^{-4}$	$1.46\ 10^{5}$
Sn-126	$1.09\ 10^{-3}$	$9.11\ 10^{-4}$	$8.55\ 10^{-4}$	$7.20\ 10^{-4}$	$4.14\ 10^{4}$
Te-127m	$3.76\ 10^{-4}$	$3.14\ 10^{-4}$	$2.87\ 10^{-4}$	$2.32\ 10^{-4}$	$1.20\ 10^{5}$
I-129	$1.38\ 10^{-3}$	$1.15\ 10^{-3}$	$1.37\ 10^{-3}$	$6.05\ 10^{-4}$	$3.28\ 10^{4}$
Ba-133	$3.84\ 10^{-4}$	$3.20\ 10^{-4}$	$2.66\ 10^{-4}$	$2.04\ 10^{-4}$	$1.18\ 10^{5}$
Cs-134	$7.68\ 10^{-4}$	$6.40\ 10^{-4}$	$5.73\ 10^{-4}$	$4.44\ 10^{-4}$	$5.90\ 10^{4}$
Cs-135	$3.30\ 10^{-4}$	$2.75\ 10^{-4}$	$2.25\ 10^{-4}$	$1.69\ 10^{-4}$	$1.37\ 10^{5}$
Cs-137$	$1.50\ 10^{-3}$	$1.25\ 10^{-3}$	$9.83\ 10^{-4}$	$7.04\ 10^{-4}$	$3.02\ 10^{4}$
Ce-144	$2.04\ 10^{-3}$	$1.70\ 10^{-3}$	$1.60\ 10^{-3}$	$1.90\ 10^{-3}$	$2.22\ 10^{4}$
Pm-147	$1.92\ 10^{-4}$	$1.60\ 10^{-4}$	$1.43\ 10^{-4}$	$1.27\ 10^{-4}$	$2.36\ 10^{5}$
Sm-147	$3.69\ 10^{-1}$	$3.07\ 10^{-1}$	$2.25\ 10^{-1}$	$1.62\ 10^{-1}$	$1.23\ 10^{2}$
Sm-151	$1.54\ 10^{-4}$	$1.28\ 10^{-4}$	$9.22\ 10^{-5}$	$7.04\ 10^{-5}$	$2.95\ 10^{5}$
Eu-152	$1.61\ 10^{-3}$	$1.34\ 10^{-3}$	$1.00\ 10^{-3}$	$7.04\ 10^{-4}$	$2.81\ 10^{4}$
Eu-154	$2.04\ 10^{-3}$	$1.70\ 10^{-3}$	$1.33\ 10^{-3}$	$1.06\ 10^{-3}$	$2.22\ 10^{4}$
Eu-155	$2.65\ 10^{-4}$	$2.21\ 10^{-4}$	$1.88\ 10^{-4}$	$1.62\ 10^{-4}$	$1.71\ 10^{5}$
Gd-153	$8.06\ 10^{-5}$	$6.72\ 10^{-5}$	$7.99\ 10^{-5}$	$8.45\ 10^{-5}$	$5.61\ 10^{5}$
Pb-210	$3.84\ 10^{-1}$	$3.20\ 10^{-1}$	$2.71\ 10^{-1}$	$2.27\ 10^{-1}$	$1.18\ 10^{2}$
Po-210	$1.65\ 10^{-1}$	$1.38\ 10^{-1}$	$1.21\ 10^{-1}$	$9.86\ 10^{-2}$	$2.74\ 10^{2}$
Ra-226	$7.50\ 10^{-1}$	$6.25\ 10^{-1}$	$5.18\ 10^{-1}$	$4.32\ 10^{-1}$	$6.04\ 10^{1}$
Ra-228	2.29	1.91	1.63	1.46	$1.98\ 10^{1}$

Radionuclide	Worker (mSv y^{-1})	Public (µSv y^{-1})			Maximum activity concentration (Bq g^{-1})
		Adult	Child	Infant	
Ac-227	$2.18\ 10^{1}$	$1.82\ 10^{1}$	$1.53\ 10^{1}$	$1.16\ 10^{1}$	2.07
Th-228	1.68	1.40	1.22	1.13	$2.70\ 10^{1}$
Th-229	9.84	8.20	6.37	3.91	4.60
Th-230	3.84	3.20	2.25	1.41	$1.18\ 10^{1}$
Th-232	6.51	5.43	4.30	3.01	6.95
Pa-231	5.38	4.48	3.07	1.62	8.42
U-232	3.10	2.58	2.10	1.81	$1.46\ 10^{1}$
U-233	$3.69\ 10^{-1}$	$3.07\ 10^{-1}$	$2.46\ 10^{-1}$	$2.11\ 10^{-1}$	$1.23\ 10^{2}$
U-234	$3.61\ 10^{-1}$	$3.01\ 10^{-1}$	$2.46\ 10^{-1}$	$2.04\ 10^{-1}$	$1.25\ 10^{2}$
U-235	$3.26\ 10^{-1}$	$2.72\ 10^{-1}$	$2.25\ 10^{-1}$	$1.83\ 10^{-1}$	$1.39\ 10^{2}$
U-236	$3.34\ 10^{-1}$	$2.78\ 10^{-1}$	$2.25\ 10^{-1}$	$1.90\ 10^{-1}$	$1.36\ 10^{2}$
U-238$	$3.07\ 10^{-1}$	$2.56\ 10^{-1}$	$2.05\ 10^{-1}$	$1.76\ 10^{-1}$	$1.47\ 10^{2}$
Np-237	1.92	1.60	1.02	$6.55\ 10^{-1}$	$2.36\ 10^{1}$
Pu-238	4.22	3.52	2.25	1.34	$1.07\ 10^{1}$
Pu-239$	4.61	3.84	2.46	1.41	9.83
Pu-240	4.61	3.84	2.46	1.41	9.83
Pu-241	$8.83\ 10^{-2}$	$7.36\ 10^{-2}$	$4.92\ 10^{-2}$	$2.04\ 10^{-2}$	$5.13\ 10^{2}$
Pu-242	4.22	3.52	2.46	1.34	$1.07\ 10^{1}$
Pu-244	4.22	3.52	2.46	1.34	$1.07\ 10^{1}$
Am-241$	3.69	3.07	2.05	1.27	$1.23\ 10^{1}$
Am-242m	4.45	3.71	2.49	1.41	$1.02\ 10^{1}$
Am-243	3.69	3.07	2.05	1.20	$1.23\ 10^{1}$
Cm-242	$2.27\ 10^{-1}$	$1.89\ 10^{-1}$	$1.68\ 10^{-1}$	$1.48\ 10^{-1}$	$2.00\ 10^{2}$
Cm-243	2.66	2.22	1.50	1.06	$1.70\ 10^{1}$
Cm-244	2.19	1.82	1.25	$9.15\ 10^{-1}$	$2.07\ 10^{1}$
Cm-245	3.80	3.17	2.05	1.27	$1.19\ 10^{1}$
Cm-246	3.76	3.14	2.05	1.27	$1.20\ 10^{1}$
Cm-248	$1.38\ 10^{1}$	$1.15\ 10^{1}$	7.58	4.58	3.28

$ radionuclides included in the Paris convention (NEA, 2017)

E.3.5. Exposure to gas during site operations

604. The permit application involves no specific authorised gaseous discharge routes for the RSR permit (gas flaring occurs under an existing non-RSR Permit). During operations, landfill workers on the site would be exposed to gas emanating from disposed waste. Public exposure to gas emanating from the waste would only occur at some distance from the source. These impacts are assessed.

605. Emission of radioactive gases as a result of combustion for power generation or flaring has also been assessed assuming that gas predominantly arises from the non-hazardous landfill. Gas collection and combustion is included in earlier capped cells but these do not contain radioactive waste. Hazardous waste landfill cells containing radioactive waste will contain insufficient material to require flaring.

606. An aerosol pathway does not arise as leachate is not sprayed on to the landfill.

607. Resuspension of dust has only been assessed for loose tipped waste (see sub-section E.3.4), all other waste is packaged, covered with suitable material before packaging can degrade and a condition for accepting wastes will require low surface contamination of packages and will be monitored.

608. The dose criteria applied in the assessment are the site criterion of 1 mSv y^{-1} for workers and the dose constraint for the public of 0.3 mSv y^{-1}.

E.3.5.1. Estimating activity concentrations of gas release from disposed waste

609. The assessment of doses from gases released from disposed waste to atmosphere is based on the SNIFFER assessment methodology (SNIFFER, 2006). Members of the exposed groups are assumed to be adults and to be exposed as a result of inhalation.

610. Radioactive gas, i.e., $^{14}CO_2$, $^{14}CH_4$, ^3H, and radon can be released to atmosphere from the waste. The first three may be generated through microbial degradation or corrosion of the radioactive waste. However, there will be a limit on the biodegradable content of LLW wastes to reduce this (Augean, 2019). Radon is generated through the decay of Ra-226, which in turn is a decay product of Th-230. The gas pathway has therefore considered radioactive carbon, tritium and radon.

611. Radioactive gases could be inhaled by workers on-site or by members of the public spending time immediately downwind of the site during the operational period and active management period. It could also be inhaled by members of the public living in a house built on the site sometime after the end of the period of authorisation and this is addressed later (see Section E.5.5 and E.5.6). Table 67 details the habit data assumed for the exposed groups during the period of authorisation.

612. During operations, landfill workers on the site would be exposed to gas emanating from disposed waste, public exposure to gas would only occur at some distance from the source. Exposure to gas has been considered for C-14, H-3 and radon.

Gas generation – H-3 and C-14

613. The release rate of radioactive gas for H-3 (in hydrogen, water, or methane) and C-14 (in carbon dioxide or methane), $R_{Rn,gas}$ (Bq y^{-1}), at time t is given by (SNIFFER, 2006):

$$R_{Rn,gas}(t) = \frac{A_{Rn,waste} \cdot e^{-\lambda_{Rn} t} \cdot f_{gas}}{\tau_{gas}}$$

where:

- $A_{Rn,waste}$ is the initial activity of radionuclide *Rn* in the waste (Bq);
- λ_{Rn} is the decay constant of radionuclide *Rn* (y^{-1});
- f_{gas} is the fraction of the activity associated with gas; and,

- τ_{gas} is the average timescale of gas generation (y).

614. The parameters used in this study are summarised in Table 69 and are from (Augean, 2010). The hazardous waste acceptance criteria at Port Clarence include a restriction on the amount of organic carbon that is disposed (6%). It is this organic carbon that would be subject to microbial action and be released as gas and this limit effectively caps the proportion of C-14 that could be released in a gaseous form. The proposed CFA for radioactive waste permits LLW to contain a greater amount of organic carbon subject to the overall site limit. Disposal of high carbon content waste to the non-hazardous waste landfill is not BAT and is therefore unlikely to occur, high carbon content waste is more likely to be disposed of through incineration. The same gas generation rates have therefore been used for both waste types.

615. The release rate is expected to vary with time. Gas generation within the landfill has been simulated using the GasSim model (Augean, 2010) which shows a rapid build-up in the rate of release after capping followed by an exponential decline. The peak annual gas yield for carbon is less than 10% of the total quantity of gas. The average timescale of gas generation has therefore been set at 10 years during operations.

Table 69 Gas generation parameters

Parameter	Units	Value	Description
$A_{Rn,waste}$	Bq	$1\ 10^{6}$	Initial activity of radionuclide *Rn*
f_{gas}		H-3: $3.9\ 10^{-2}$	Fraction of activity associated with gas
		C-14: $6.0\ 10^{-2}$	
τ_{gas}	y	10	Average timescale of gas generation

From (Augean, 2010)

616. The activity concentration of a radionuclide in air, $C_{Rn,gas,outdoors}$ (Bq m^{-3}), can be approximated by dividing the release rate by the air volume into which the activity released per year is diluted (SNIFFER, 2006):

$$C_{Rn,gas,outdoors} = \frac{R_{Rn,gas}}{(\mathrm{W} \cdot \mathrm{u} \cdot \mathrm{h} \cdot \mathrm{s}_y)}$$

where:

- $R_{Rn,gas}$ is the release rate of radionuclide Rn in gas (Bq y^{-1}) at the time of interest;
- W is the width of the source perpendicular to the wind direction (m);
- u is the mean wind speed (m s^{-1});
- h is the height for vertical mixing (m); and,
- s_y is the number of seconds in a year, $3.16\ 10^{7}$ (s y^{-1}).

617. The dose from gases other than radon is given by (SNIFFER, 2006):

$$Dose_{Rn,gas,outdoors} = C_{Rn,gas,outdoors} \cdot B \cdot O_{out} \cdot D_{Rn,inh}$$

where:

- O_{out} is the time spent in the gas plume (h y^{-1});
- B is the breathing rate (m^3 h^{-1}); and,
- $D_{Rni,nh}$ is the dose coefficient for inhalation (Sv Bq^{-1}).

618. The dispersion parameter values used in the ESC are given in Table 70, the dose coefficients in Table 200 and the habit data in Table 67.

Table 70 Parameter values used in calculations of doses through the gas pathway during site operations

Parameter	Units	Value	Description
W	m	313	Width of source perpendicular to the wind direction
u	m s^{-1}	4.63	Mean wind speed
h	m	2.0	Height for vertical mixing
s_y	s	3.16 10^7	Seconds in a year

619. The meterological data for Teeside indicates wind direction and speed (Table 71). This is used to calculate the direction in which the highest impact would occur over the range of recorded wind speeds. These calculations indicate that the highest dose occurs to a group exposed to the south of the site. It assumes that mixing is limited to a height of 2 m and that the width of the source is limited to the narrowest width based on current information for the site. These assumptions are conservative. Wind data for the meteorological station closest to Port Clarence show that the peak dose, using a combination of wind speed and the prevailing sector, to a member of the public is about 13% of the value calculated assuming that the exposed group is always downwind of the release point.

Table 71 Wind data from Teeside for 2007 to 2011

Wind direction	Wind speed (m s^{-1}): fraction of year in each direction						
	0.5 – 2	2 - 3	3 – 4	4 – 6	6 - 8	8 – 10	>= 10
N	0.0064	0.0077	0.0073	0.0094	0.0051	0.0022	0.0008
NNE	0.0053	0.0071	0.0068	0.0149	0.0087	0.0028	0.0006
NE	0.0045	0.0062	0.0072	0.0148	0.0082	0.0031	0.0006
ENE	0.0044	0.0044	0.0046	0.0065	0.0042	0.0016	0.0003
E	0.0038	0.0036	0.0027	0.0030	0.0012	0.0004	0.0001
ESE	0.0047	0.0042	0.0026	0.0032	0.0019	0.0006	0.0001
SE	0.0071	0.0073	0.0076	0.0094	0.0031	0.0010	0.0003

Wind direction	Wind speed (m s^{-1}): fraction of year in each direction						
	0.5 – 2	2 - 3	3 – 4	4 – 6	6 - 8	8 – 10	>= 10
SSE	0.0093	0.0138	0.0127	0.0209	0.0126	0.0062	0.0033
S	0.0104	0.0172	0.0198	0.0406	0.0274	0.0100	0.0039
SSW	0.0134	0.0221	0.0212	0.0304	0.0178	0.0084	0.0070
SW	0.0125	0.0183	0.0140	0.0212	0.0186	0.0131	0.0118
WSW	0.0113	0.0134	0.0123	0.0230	0.0202	0.0115	0.0090
W	0.0089	0.0098	0.0116	0.0212	0.0126	0.0068	0.0037
WNW	0.0078	0.0085	0.0085	0.0137	0.0074	0.0028	0.0012
NW	0.0072	0.0076	0.0065	0.0144	0.0080	0.0021	0.0005
NNW	0.0075	0.0089	0.0079	0.0139	0.0071	0.0030	0.0008

Gas generation – Radon

620. Radon (i.e. Rn-222) gas is a short-lived (half-life of 3.82 days) radionuclide that is released as a consequence of the decay of Ra-226. Over long timescales, the ingrowth of Ra-226 through the Cm-246 decay chain (see Figure 13) will also result in radon gas release.

621. Radon decays to a number of very short-lived radioactive decay products, and it is these progeny, rather than radon itself, that present the greater risk. However, conventionally, ‘radon’ is used as convenient shorthand to include both radon and its progeny (Quintessa Ltd, 2011).

622. The flux of radon, $F_{radon}(t)$ (Bq y^{-1}), through an intact (or partially damaged cap) is calculated according to (SNIFFER, 2006):

$$F_{radon}(t) = \lambda_{Rn-222} \cdot AREA \cdot C_{Ra-226} \cdot e^{-\lambda_{Ra-226} t} \cdot \rho_{waste} \cdot \tau \cdot H_1 \cdot e^{\frac{-h_2}{H_2}}$$

where :

- $AREA$ is the surface area containing radioactive waste;
- C_{Ra-226} is the initial ^{226}Ra concentration in the waste (Bq kg^{-1});
- t is the time at which the flux is evaluated;
- ρ_{waste} is the bulk density of the waste (kg m^{-3});
- τ is the emanation factor, the fraction of the radon atoms produced which escape from the solid phase of the waste into the pore spaces;
- H_1 is the effective diffusion relaxation length for the waste (m);
- h_2 is the thickness of the cover (m); and,
- H_2 is the effective relaxation length of the cover (m).

623. The activity concentration of radon in outdoor air is calculated using the equation given in paragraph 616 and the parameters in Table 70. The radon calculations for members of the public are adjusted for the wind direction and speed (see paragraph 619).

624. The release of radon gas is sensitive to the cover depth and the assumption that the complete inventory is only covered with the daily cover depth (0.3 m of material) is not realistic over the operational period. The landfill comprises a series of cells and the average period until a further layer of waste is applied at any location is about two months. It has therefore been assumed that any waste is covered with at least a further 0.7 m of material within 2 months. Thus the dose is a combination of 2 months with 0.3 m cover and 10 months with ≥1 m cover. A cover depth of 1 m or more reduces radon emissions significantly (more than a 97% reduction) so the annual radon dose from each layer is essentially that from the first 2 months.

Table 72 Radon parameters

Parameter	Units	Value	Description	Comment
ρ_{waste}	kg m^{-3}	1,530	Waste density	
τ		0.1	emanation factor	From (HPA, 2007)
H_1	m	0.2	effective diffusion relaxation length for the waste	From (HPA, 2007)
H_2	m	0.2	effective relaxation length of the cover	From (HPA, 2007)
h_2	m	0.3	thickness of cover for first two months	daily cover depth
		1.0	thickness of cover for remaining ten months	
AREA	m^2	130,874	hazardous landfill	
		142,594	Non-hazardous landfill	

E.3.5.2. Assessment calculation for gas releases

625. The dose from gases is given by (SNIFFER, 2006):

$$Dose_{Rn,gas,outdoors} = C_{Rn,gas,outdoors} \cdot B \cdot O_{out} \cdot D_{Rn,inh}$$

where:

- O_{out} is the time spent in the gas plume (h y^{-1});
- B is the breathing rate (m^3 h^{-1}); and,
- $D_{Rni,nh}$ is the dose coefficient for inhalation (Sv Bq^{-1}).

626. The dose coefficients for C-14 and H-3 are in Table 200 and the habit data in Table 67.

627. The dose coefficient for radon (Table 73) applied in this ESC accounts for the effect of the daughters of Rn-222 in the body and is taken from the radiological assessment methodology developed by the Environment Agency IRAM, see Section 4 in Appendix B (Environment Agency, 2006b). Habit data for workers and members of the public are presented in Table 67.

Table 73 Inhalation dose coefficient for use in calculation of radon doses

Parameter	Units	Value	Description
D_{inh}	Sv Bq^{-1}	6.0 10^{-9}	Adult
		1.2 10^{-8}	Child
		4.5 10^{-8}	Infant

E.3.5.3. Doses from gas releases during operations

628. The release of gases during operations will expose landfill workers on the site. Public exposure to gas would occur at some distance from the source. The calculations assume that there is no radioactive decay (or daughter ingrowth) and that members of the public are always present in the wind direction resulting in the highest dose, for Ra-226 it is also assumed that waste is covered on a daily basis to a depth of 0.3 m, and covered again within 2 months.

Table 74 Dose estimated for exposure from gas released during operations

Radionuclide	Dose (µSv y^{-1} MBq^{-1})	
	Worker	Public
H-3	2.51 10^{-8}	1.11 10^{-8}
C-14	8.62 10^{-7}	3.80 10^{-7}
Ra-226	3.51 10^{-6}	2.58 10^{-7}

629. The dose estimates indicate that the worst case is for Ra-226 disposal in the case of the worker and C-14 in the case of the public. This gas release during operations scenario has the potential to constrain the radiological capacity of Port Clarence. The results are independent of the eventual Ra-226 placement depth in the site since this scenario assesses the dose immediately after emplacement.

630. The doses calculated using illustrative inventories are considered further in Appendix D.

E.3.5.4. Exposure to gases collected from capped cells

631. The site operates a system for management of landfill gas. The landfill gas generation and utilisation for the year 2018 at Port Clarence is summarised in Table 75 [Pers. Comm. Peter Oldfield to Nick Mitchell, “Port Clarence”, 04/04/2019].

Table 75 Landfill gas generation and utilisation for the year 2018 at Port Clarence

Area	Landfill gas generated (t)	Proportion used for power generation (%)	Proportion vented passively (%)
Port Clarence Hazardous landfill	146	0	100
Port Clarence Non-hazardous landfill	4807	65	35

632. A facility has been installed at Port Clarence that generates electricity from the landfill gas. The radiological assessment of that facility assumes conversion of CH_4 into CO_2 and H_2O.

633. The LLW wastes that will be disposed of at the site have a generally low level of organic matter and are only slowly degradable. The levels of radioactivity in LLW are too low to give rise to a risk from radiolytic hydrogen gas evolution.

634. PC-CREAM 08 provides the assessment of gaseous tritium and C-14 releases, generally keeping the default parameters. PC-CREAM 08 implements a Gaussian Plume model for atmospheric dispersion of gases and a combination of assessment models, including Plume, Farmland, Granis and Resus (Smith & Simmonds, 2015).

635. The methodology uses the disposed inventory and a gas release rate to give the activity released in a year. Hence, the gas generation values in Table 75 are not used directly in the radiological assessment, but can subsequently be used to derive an activity concentration in the generated gas. The tritium release rate was taken from the LLWR assessment for the period of authorization (Penfold & Paulley, 2011) as 0.01 y^{-1}. The C-14 release rate was taken from the C-14 assessment undertaken for LLWR (LLWR Ltd, 2013) as 0.0316 y^{-1}, which is appropriately cautious for mixed waste containing various materials. Decreases of activity through release and decay were not taken into account, the assessment is based on the maximum release rate, which occurs when the site is just filled completely.

636. The weather conditions were based on annual data from 2007 to 2011 published by the Meteorological Office (Table 71). The average wind speed is 4.6 m s^{-1}. It should be noted that the weather data is presented as wind blowing from directions, while PC-CREAM implements wind blowing to directions in the meteorological data files.

637. PC-CREAM 08 uses a Pasquill weather classification system. The UK generally has a mix of Categories C and D. The Radioactivity in Food and the Environment (RIFE) report for 2017 (RIFE-23, 2018) advises 70% as the frequency of Pasquill stability category D for Hartlepool, which is used as surrogate for Middlesbrough and PC-CREAM uses 10% rain as a default. This was implemented as 27% Category C, 63% Category D, 3% Category C Rain and 7% Category D Rain.

638. Three receptor locations were chosen to represent:

- the worst-case scenario (site boundary);

- the nearest downwind residential site (Greetham); and,
- the nearest residential site (Middlesbrough Tees bank).

639. Table 76 summarises the results of the gaseous dispersion assessment for tritium and C-14, where the disposed radionuclide inventory has been assumed to be 1 MBq for both tritium and C-14.

Table 76 Calculated annual dose for gaseous dispersion following disposal of 1 MBq of H-3 and C-14 and dose at radiological capacity

Receptor		Calculated annual dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)		Dose at radiological capacity ($\mu Sv\ y^{-1}$)	
Location	Age group	H-3	C-14	H-3	C-14
Site boundary (500 m, 20 degrees)	Adult	$1.80\ 10^{-9}$	$1.50\ 10^{-6}$	$1.16\ 10^{1}$	$7.04\ 10^{1}$
	Child	$1.76\ 10^{-9}$	$1.50\ 10^{-6}$	$1.13\ 10^{1}$	$7.04\ 10^{1}$
	Infant	$3.17\ 10^{-9}$	$1.92\ 10^{-6}$	$2.04\ 10^{1}$	$9.01\ 10^{1}$
Greetham (5000 m, 345 degrees)	Adult	$3.61\ 10^{-11}$	$3.01\ 10^{-8}$	$2.32\ 10^{-1}$	$1.41\ 10^{0}$
	Child	$3.52\ 10^{-11}$	$3.00\ 10^{-8}$	$2.26\ 10^{-1}$	$1.41\ 10^{0}$
	Infant	$6.35\ 10^{-11}$	$3.85\ 10^{-8}$	$4.08\ 10^{-1}$	$1.81\ 10^{0}$
Middlesbrough Tees bank (800 m, 135 degrees)	Adult	$4.05\ 10^{-10}$	$3.37\ 10^{-7}$	$2.60\ 10^{0}$	$1.58\ 10^{1}$
	Child	$3.95\ 10^{-10}$	$3.36\ 10^{-7}$	$2.54\ 10^{0}$	$1.58\ 10^{1}$
	Infant	$7.11\ 10^{-10}$	$4.32\ 10^{-7}$	$4.57\ 10^{0}$	$2.03\ 10^{1}$
Radiological capacity (MBq)				$6.43\ 10^{9}$	$1.87\ 10^{8}$

640. These results for C-14 are comparable to the operational assessment (see Table 74) and an order of magnitude lower for H-3. This scenario would only apply to the non-hazardous waste cells and will not therefore constrain the radiological capacity of Port Clarence.

E.3.6. Exposures from non-hazardous waste landfill fire

641. Fires at landfill sites can disperse material from the waste and hence a fire at Port Clarence would create the potential for exposure to radioactive material dispersed in the air. The disposal of organic carbon to the hazardous waste site at Port Clarence is limited to 6% total organic carbon and a fire is very unlikely in hazardous waste cells. Hence, no fire assessment has been made for the hazardous landfill at Port Clarence. However, there is no limit on total organic carbon for the non-hazardous waste landfill at Port Clarence and hence a cautious assessment has been undertaken. Note that the disposal of wastes that are high in organic carbon, and which could therefore ignite, is unlikely because BAT disposal of these wastes would be incineration.

642. The assessment uses the same approach adopted by PHE for the NORM waste assessments at Port Clarence (Jones, et al., 2014). The approach taken by PHE simplifies the processes involved and gives an illustrative value for the resulting dose; doses from a real fire depend heavily on environmental and meteorological factors at the time.

643. Doses are calculated for members of the public, who are assumed to be working 250 m downwind of the fire. It is assumed that the fire burns for one hour at ground level, and consumes 100 m^3 of the waste (SNIFFER, 2006).

644. The dose criteria used in the assessment are 1 mSv y^{-1} for workers at the site and the dose constraint for the public of 0.3 mSv y^{-1}.

E.3.6.1. Assessment calculation for releases to air from a fire

645. An element-specific fraction of the burnt material is lifted with the plume (Asselineau et al., 1995), which is neutrally buoyant and non-depleting.

646. The exposure pathways are inhalation of and external exposure to radionuclides in the smoke plume for the duration of the fire. The dose from one fire is given by:

$$Dose_{fire} = D_{Inh} + D_{ext,cloud}$$

where:

- D_{fire} is total dose from a fire (mSv);
- D_{Inh} is the dose from inhalation (mSv); and,
- $D_{ext,cloud}$ is the dose from external exposure from the passing cloud.

647. The dose from inhalation is given by:

$$Dose_{Rn,Inh} = C_{Rn,smoke} \cdot T_{fire} \cdot R_{Inh} \cdot D_{Rn,Inh}$$

where:

- $D_{Rn,Inh}$ is the dose from inhalation for radionuclide *Rn* (Sv);
- $C_{Rn,smoke}$ is the activity concentration in the smoke plume for radionuclide *Rn* (Bq m^{-3});
- T_{fire} is the duration of the fire (h) and exposure time;
- R_{Inh} is the inhalation rate (m^3 h^{-1}); and,
- $D_{Rn,Inh}$ is the inhalation dose coefficient for radionuclide *Rn* (Sv Bq^{-1}).

648. The dose from external exposure is given by:

$$Dose_{Rn,ext,cloud} = C_{Rn,smoke} \cdot T_{fire} \cdot D_{Rn,ext,cloud}$$

where:

- $D_{Rn,ext,cloud}$ is the external dose from the plume for radionuclide *Rn* (Sv);
- $C_{Rn,smoke}$ is the activity concentration in the smoke plume for radionuclide *Rn* (Bq m^{-3});

- T_{fire} is the duration of the fire (h) and exposure time; and,
- $D_{Rn,ext,cloud}$ is the external dose coefficient for radionuclide *Rn* (Sv Bq^{-1} h^{-1} m^3).

649. The activity concentration in the smoke (SNIFFER, 2006) is given by:

$$C_{Rn,smoke} = \frac{V_{fire}}{V_{landfill}} \cdot RF \cdot A_{landfill} \cdot \frac{C_{TIAC} \cdot CF}{T_{fire}}$$

where:

- $C_{Rn,smoke}$ is the activity concentration in the smoke plume for radionuclide *Rn* (Bq m^{-3});
- V_{fire} is the volume of waste consumed by the fire (m^3);
- $V_{landfill}$ is the volume of the landfill (m^3);
- *RF* is the fraction of the radionuclide consumed in the fire that reaches the plume (dimensionless, see Table 200);
- $A_{landfill}$ is the activity disposed of into the landfill (Bq);
- C_{TIAC} is the time-integrated activity concentration in air at the receptor point from a 30 minute release (Bq h m^{-3} Bq^{-1});
- *CF* correction factor for a 60 minute release (dimensionless);
- T_{fire} is the duration of the fire (h).

650. It is assumed that there are two fires per year and hence the dose is multiplied by two.

651. Parameter values and dose coefficients used in the calculations are summarised in Table 77.

Table 77 Parameters for use in calculation of doses from fires

Parameter	Units	Value	Reference
Tf_{ire}	H	1	(SNIFFER, 2006)
V_{fire}	m^3	100	(SNIFFER, 2006)
C_{TIAC}	Bq h m^{-3} Bq^{-1}	2.8 10^{-7}	(Clarke, 1979)
CF		0.7	(Clarke, 1979)

E.3.6.2. Doses from a fire in the non-hazardous waste landfill

652. The calculated doses shown below for each of the assessed groups are per MBq input to Port Clarence (Table 78). The radiological capacity (MBq) associated with the limiting age group for each radionuclide is also shown in the right-hand column. This scenario has the potential to constrain the radiological capacity of the non-hazardous landfill at Port Clarence.

Table 78 Doses estimated for two fires in a non-hazardous waste cell in one year

Radionuclide	Public dose (mSv y^{-1}) per MBq disposed			Radiological capacity (MBq)
	Adult	Child	Infant	
H-3$	$5.00\ 10^{-12}$	$4.68\ 10^{-12}$	$4.23\ 10^{-12}$	$6.00\ 10^{10}$
C-14$	$1.12\ 10^{-10}$	$9.11\ 10^{-11}$	$7.19\ 10^{-11}$	$2.69\ 10^{9}$
Cl-36	$1.40\ 10^{-10}$	$1.23\ 10^{-10}$	$1.10\ 10^{-10}$	$2.14\ 10^{9}$
Ca-41	$3.46\ 10^{-15}$	$4.06\ 10^{-15}$	$2.54\ 10^{-15}$	$7.38\ 10^{13}$
Mn-54	$3.17\ 10^{-14}$	$3.24\ 10^{-14}$	$2.91\ 10^{-14}$	$9.26\ 10^{12}$
Fe-55	$1.48\ 10^{-14}$	$1.72\ 10^{-14}$	$1.35\ 10^{-14}$	$1.74\ 10^{13}$
Co-60$	$6.05\ 10^{-13}$	$5.01\ 10^{-13}$	$3.73\ 10^{-13}$	$4.96\ 10^{11}$
Ni-59	$8.46\ 10^{-15}$	$7.26\ 10^{-15}$	$6.35\ 10^{-15}$	$3.54\ 10^{13}$
Ni-63	$2.50\ 10^{-14}$	$2.09\ 10^{-14}$	$1.82\ 10^{-14}$	$1.20\ 10^{13}$
Zn-65	$4.43\ 10^{-12}$	$4.88\ 10^{-12}$	$4.43\ 10^{-12}$	$6.15\ 10^{10}$
Se-79	$1.31\ 10^{-10}$	$1.07\ 10^{-10}$	$8.46\ 10^{-11}$	$2.29\ 10^{9}$
Sr-90$	$3.11\ 10^{-12}$	$2.25\ 10^{-12}$	$1.73\ 10^{-12}$	$9.66\ 10^{10}$
Mo-93	$4.42\ 10^{-14}$	$3.45\ 10^{-14}$	$2.45\ 10^{-14}$	$6.78\ 10^{12}$
Zr-93	$4.81\ 10^{-13}$	$1.19\ 10^{-13}$	$2.71\ 10^{-14}$	$6.24\ 10^{11}$
Nb-93m	$3.46\ 10^{-14}$	$3.08\ 10^{-14}$	$2.75\ 10^{-14}$	$8.66\ 10^{12}$
Nb-94	$9.48\ 10^{-13}$	$7.19\ 10^{-13}$	$5.13\ 10^{-13}$	$3.16\ 10^{11}$
Tc-99$	$2.50\ 10^{-13}$	$2.09\ 10^{-13}$	$1.57\ 10^{-13}$	$1.20\ 10^{12}$
Ru-106	$1.27\ 10^{-10}$	$1.12\ 10^{-10}$	$9.74\ 10^{-11}$	$2.36\ 10^{9}$
Ag-108m	$7.17\ 10^{-12}$	$5.47\ 10^{-12}$	$3.74\ 10^{-12}$	$4.18\ 10^{10}$
Ag-110m	$2.40\ 10^{-12}$	$2.31\ 10^{-12}$	$1.83\ 10^{-12}$	$1.25\ 10^{11}$
Cd-109	$1.56\ 10^{-13}$	$1.72\ 10^{-13}$	$1.57\ 10^{-13}$	$1.74\ 10^{12}$
Sb-125	$2.51\ 10^{-11}$	$2.15\ 10^{-11}$	$1.75\ 10^{-11}$	$1.20\ 10^{10}$
Sn-119m	$4.23\ 10^{-14}$	$3.82\ 10^{-14}$	$3.34\ 10^{-14}$	$7.09\ 10^{12}$
Sn-123	$1.56\ 10^{-13}$	$1.48\ 10^{-13}$	$1.31\ 10^{-13}$	$1.92\ 10^{12}$
Sn-126	$5.54\ 10^{-13}$	$5.21\ 10^{-13}$	$4.39\ 10^{-13}$	$5.41\ 10^{11}$
Te-127m	$1.89\ 10^{-11}$	$1.72\ 10^{-11}$	$1.40\ 10^{-11}$	$1.59\ 10^{10}$
I-129	$6.93\ 10^{-10}$	$8.25\ 10^{-10}$	$3.64\ 10^{-10}$	$3.64\ 10^{8}$
Ba-133	$1.94\ 10^{-13}$	$1.61\ 10^{-13}$	$1.24\ 10^{-13}$	$1.55\ 10^{12}$
Cs-134	$3.90\ 10^{-11}$	$3.50\ 10^{-11}$	$2.72\ 10^{-11}$	$7.69\ 10^{9}$
Cs-135	$1.65\ 10^{-11}$	$1.35\ 10^{-11}$	$1.02\ 10^{-11}$	$1.81\ 10^{10}$
Cs-137$	$7.52\ 10^{-11}$	$5.93\ 10^{-11}$	$4.25\ 10^{-11}$	$3.99\ 10^{9}$
Ce-144	$1.02\ 10^{-12}$	$9.60\ 10^{-13}$	$1.14\ 10^{-12}$	$2.63\ 10^{11}$
Pm-147	$9.62\ 10^{-14}$	$8.62\ 10^{-14}$	$7.62\ 10^{-14}$	$3.12\ 10^{12}$
Sm-147	$1.85\ 10^{-10}$	$1.35\ 10^{-10}$	$9.73\ 10^{-11}$	$1.62\ 10^{9}$
Sm-151	$7.70\ 10^{-14}$	$5.54\ 10^{-14}$	$4.23\ 10^{-14}$	$3.90\ 10^{12}$
Eu-152	$8.12\ 10^{-13}$	$6.07\ 10^{-13}$	$4.27\ 10^{-13}$	$3.70\ 10^{11}$
Eu-154	$1.02\ 10^{-12}$	$8.05\ 10^{-13}$	$6.39\ 10^{-13}$	$2.93\ 10^{11}$
Eu-155	$1.33\ 10^{-13}$	$1.13\ 10^{-13}$	$9.75\ 10^{-14}$	$2.26\ 10^{12}$
Gd-153	$4.07\ 10^{-14}$	$4.83\ 10^{-14}$	$5.10\ 10^{-14}$	$5.88\ 10^{12}$
Pb-210	$9.61\ 10^{-8}$	$8.14\ 10^{-8}$	$6.84\ 10^{-8}$	$3.12\ 10^{6}$
Po-210	$8.27\ 10^{-11}$	$7.26\ 10^{-11}$	$5.93\ 10^{-11}$	$3.63\ 10^{9}$
Ra-226	$3.76\ 10^{-10}$	$3.11\ 10^{-10}$	$2.60\ 10^{-10}$	$7.99\ 10^{8}$
Ra-228	$1.15\ 10^{-9}$	$9.82\ 10^{-10}$	$8.80\ 10^{-10}$	$2.61\ 10^{8}$
Ac-227	$1.09\ 10^{-8}$	$9.17\ 10^{-9}$	$7.00\ 10^{-9}$	$2.74\ 10^{7}$
Th-228	$8.39\ 10^{-10}$	$7.35\ 10^{-10}$	$6.76\ 10^{-10}$	$3.57\ 10^{8}$

Radionuclide	Public dose (mSv y^{-1}) per MBq disposed			Radiological capacity (MBq)
	Adult	Child	Infant	
Th-229	$4.93\ 10^{-9}$	$3.83\ 10^{-9}$	$2.35\ 10^{-9}$	$6.09\ 10^{7}$
Th-230	$1.92\ 10^{-9}$	$1.35\ 10^{-9}$	$8.46\ 10^{-10}$	$1.56\ 10^{8}$
Th-232	$3.26\ 10^{-9}$	$2.58\ 10^{-9}$	$1.81\ 10^{-9}$	$9.19\ 10^{7}$
Pa-231	$2.69\ 10^{-9}$	$1.85\ 10^{-9}$	$9.73\ 10^{-10}$	$1.11\ 10^{8}$
U-232	$1.55\ 10^{-9}$	$1.26\ 10^{-9}$	$1.09\ 10^{-9}$	$1.93\ 10^{8}$
U-233	$1.85\ 10^{-10}$	$1.48\ 10^{-10}$	$1.27\ 10^{-10}$	$1.62\ 10^{9}$
U-234	$1.81\ 10^{-10}$	$1.48\ 10^{-10}$	$1.23\ 10^{-10}$	$1.66\ 10^{9}$
U-235	$1.64\ 10^{-10}$	$1.35\ 10^{-10}$	$1.10\ 10^{-10}$	$1.83\ 10^{9}$
U-236	$1.67\ 10^{-10}$	$1.35\ 10^{-10}$	$1.14\ 10^{-10}$	$1.79\ 10^{9}$
U-238$	$1.54\ 10^{-10}$	$1.23\ 10^{-10}$	$1.06\ 10^{-10}$	$1.95\ 10^{9}$
Np-237	$9.62\ 10^{-10}$	$6.16\ 10^{-10}$	$3.94\ 10^{-10}$	$3.12\ 10^{8}$
Pu-238	$2.12\ 10^{-9}$	$1.35\ 10^{-9}$	$8.04\ 10^{-10}$	$1.42\ 10^{8}$
Pu-239$	$2.31\ 10^{-9}$	$1.48\ 10^{-9}$	$8.46\ 10^{-10}$	$1.30\ 10^{8}$
Pu-240	$2.31\ 10^{-9}$	$1.48\ 10^{-9}$	$8.46\ 10^{-10}$	$1.30\ 10^{8}$
Pu-241	$4.42\ 10^{-11}$	$2.95\ 10^{-11}$	$1.23\ 10^{-11}$	$6.78\ 10^{9}$
Pu-242	$2.12\ 10^{-9}$	$1.48\ 10^{-9}$	$8.04\ 10^{-10}$	$1.42\ 10^{8}$
Pu-244	$2.12\ 10^{-9}$	$1.48\ 10^{-9}$	$8.04\ 10^{-10}$	$1.42\ 10^{8}$
Am-241$	$1.85\ 10^{-9}$	$1.23\ 10^{-9}$	$7.62\ 10^{-10}$	$1.62\ 10^{8}$
Am-242m	$2.23\ 10^{-9}$	$1.50\ 10^{-9}$	$8.47\ 10^{-10}$	$1.35\ 10^{8}$
Am-243	$1.85\ 10^{-9}$	$1.23\ 10^{-9}$	$7.20\ 10^{-10}$	$1.62\ 10^{8}$
Cm-242	$1.14\ 10^{-10}$	$1.01\ 10^{-10}$	$8.89\ 10^{-11}$	$2.64\ 10^{9}$
Cm-243	$1.33\ 10^{-9}$	$9.02\ 10^{-10}$	$6.37\ 10^{-10}$	$2.25\ 10^{8}$
Cm-244	$1.10\ 10^{-9}$	$7.51\ 10^{-10}$	$5.50\ 10^{-10}$	$2.74\ 10^{8}$
Cm-245	$1.90\ 10^{-9}$	$1.23\ 10^{-9}$	$7.62\ 10^{-10}$	$1.58\ 10^{8}$
Cm-246	$1.89\ 10^{-9}$	$1.23\ 10^{-9}$	$7.62\ 10^{-10}$	$1.59\ 10^{8}$
Cm-248	$6.93\ 10^{-9}$	$4.56\ 10^{-9}$	$2.75\ 10^{-9}$	$4.33\ 10^{7}$

E.3.7. Exposures from leachate processing

653. The permit application involves no specific authorised radioactive liquid discharge routes. The leachate collected at the landfill is currently used at the WRP treatment facilities or sent off-site during maintenance periods. An assessment has been made of the radiological impact arising from off-site treatment of contaminated leachate at an industrial liquid waste treatment plant and a reed-bed facility. An EPR radioactive substances Permit would be required for offsite disposal of leachate containing radionuclides.

654. A GoldSim groundwater model for the whole site provides an estimate of the annual leachate from the facility and an estimate of the maximum activity concentration in the leachate; the activity concentrations are used to assess the impact of leachate treatment. The radiological assessment considers:

- the treatment of contaminated leachate at an off-site hazardous waste treatment facility followed by discharge of liquid effluent to an estuary and use of sludge from the treatment works digestors as a soil improver; and,
- the treatment of contaminated leachate at an off-site reed bed facility.

655. The assessment considers the radiation exposure of workers at the treatment facility, anglers fishing in the estuary into which the treatment works discharge and a farming family assumed to grow crops on land fertilised with sludge from the aerobic digestors.

656. The assessment is based on the Environment Agency initial radiological assessment methodology (Environment Agency, 2006b) that has been implemented in Excel to consider all the radionuclides listed in Table 6. The initial radiological assessment methodology for a sewage treatment works is used here as a proxy for a hazardous waste processing facility taking into account an appropriate total input flow rate. It is assumed that worker doses at the hazardous waste treatment facility would be similar to worker doses at a sewage treatment facility. The methodology accounts for radionuclide-specific partitioning of activity between treated sewage effluent and sewage solids.

657. The Reed Bed assessment considers contamination of the total area of the Reed Beds (49,000 m^2) and accumulation over 7 years which is the anticipated operating life of the beds. This dose assessment considers external exposure to this contaminated area as irradiation from a semi-infinite slab. The treated leachate is then discharged to the estuary via Billingham Beck. Disposal of the reed bed is not considered in this assessment but would be required to support an application for authorised discharges of leachate containing radionuclides from LLW disposals to the reed bed facility.

658. The dose criteria used in the assessment are 1 mSv y^{-1} for workers in the off-site facility and the dose constraint for the public of 0.3 mSv y^{-1}.

E.3.7.1. Estimating activity concentrations in leachate

659. The flux of radionuclides to the treatment works (Bq y^{-1}) uses the peak leachate activity concentrations (per MBq input to the landfill) at 60 years and the leachate export rate (2647 m^3 y^{-1}) from the site. The ingrowth of daughters is modelled using GoldSim and the activity concentrations of the daughters are propagated through the model and the dose contributions summed.

Table 79 Projected leachate activity concentration and input to treatment works

Radionuclide	Leachate activity concentration (Bq m^{-3}/MBq)	Flux to treatment works (Bq y^{-1}/MBq)
H-3	$6.78\ 10^{-1}$	$1.79\ 10^{3}$
C-14	$2.23\ 10^{-3}$	$5.90\ 10^{0}$
Cl-36	$4.36\ 10^{-1}$	$1.16\ 10^{3}$
Ca-41	$2.72\ 10^{-2}$	$7.20\ 10^{1}$
Mn-54	$8.90\ 10^{-5}$	$2.35\ 10^{-1}$
Fe-55	$1.98\ 10^{-4}$	$5.25\ 10^{-1}$
Co-60	$4.09\ 10^{-4}$	$1.08\ 10^{0}$

Radionuclide	Leachate activity concentration ($Bq\ m^{-3}/MBq$)	Flux to treatment works ($Bq\ y^{-1}/MBq$)
Ni-59	$7.97\ 10^{-4}$	$2.11\ 10^{0}$
Ni-63	$7.91\ 10^{-4}$	$2.09\ 10^{0}$
Zn-65	$9.34\ 10^{-5}$	$2.47\ 10^{-1}$
Se-79	$1.12\ 10^{-3}$	$2.95\ 10^{0}$
Sr-90	$4.17\ 10^{-3}$	$1.10\ 10^{1}$
Mo-93	$5.55\ 10^{-3}$	$1.47\ 10^{1}$
Zr-93	$5.44\ 10^{-4}$	$1.44\ 10^{0}$
Nb-93m	$1.43\ 10^{-4}$	$3.77\ 10^{-1}$
Nb-94	$1.49\ 10^{-4}$	$3.94\ 10^{-1}$
Tc-99	$5.06\ 10^{-1}$	$1.34\ 10^{3}$
Ru-106	$4.42\ 10^{-4}$	$1.17\ 10^{0}$
Ag-108m	$5.86\ 10^{-4}$	$1.55\ 10^{0}$
Ag-110m	$2.38\ 10^{-4}$	$6.30\ 10^{-1}$
Cd-109	$8.91\ 10^{-4}$	$2.36\ 10^{0}$
Sb-125	$2.81\ 10^{-3}$	$7.45\ 10^{0}$
Sn-119m	$6.38\ 10^{-5}$	$1.69\ 10^{-1}$
Sn-123	$2.83\ 10^{-5}$	$7.49\ 10^{-2}$
Sn-126	$1.40\ 10^{-4}$	$3.69\ 10^{-1}$
Te-127m	$7.45\ 10^{-5}$	$1.97\ 10^{-1}$
I-129	$3.14\ 10^{-2}$	$8.31\ 10^{1}$
Ba-133	$3.42\ 10^{-1}$	$9.05\ 10^{2}$
Cs-134	$1.35\ 10^{-4}$	$3.57\ 10^{-1}$
Cs-135	$1.86\ 10^{-4}$	$4.92\ 10^{-1}$
Cs-137	$1.82\ 10^{-4}$	$4.81\ 10^{-1}$
Ce-144	$8.32\ 10^{-5}$	$2.20\ 10^{-1}$
Pm-147	$3.84\ 10^{-4}$	$1.02\ 10^{0}$
Sm-147	$2.40\ 10^{-4}$	$6.35\ 10^{-1}$
Sm-151	$2.38\ 10^{-4}$	$6.30\ 10^{-1}$
Eu-152	$8.83\ 10^{-4}$	$2.34\ 10^{0}$
Eu-154	$8.58\ 10^{-4}$	$2.27\ 10^{0}$
Eu-155	$8.06\ 10^{-4}$	$2.13\ 10^{0}$
Gd-153	$1.80\ 10^{-4}$	$4.76\ 10^{-1}$
Pb-210	$1.08\ 10^{-4}$	$2.86\ 10^{-1}$
Po-210	$2.35\ 10^{-4}$	$6.23\ 10^{-1}$
Ra-226	$8.93\ 10^{-5}$	$2.36\ 10^{-1}$
Ra-228	$7.93\ 10^{-5}$	$2.10\ 10^{-1}$

Radionuclide	Leachate activity concentration (Bq m^{-3}/MBq)	Flux to treatment works (Bq y^{-1}/MBq)
Ac-227	1.27 10^{-4}	3.37 10^{-1}
Th-228	8.31 10^{-5}	2.20 10^{-1}
Th-229	1.18 10^{-4}	3.11 10^{-1}
Th-230	1.18 10^{-4}	3.11 10^{-1}
Th-232	1.18 10^{-4}	3.11 10^{-1}
Pa-231	1.12 10^{-4}	2.95 10^{-1}
U-232	1.10 10^{-3}	2.92 10^{0}
U-233	1.12 10^{-3}	2.95 10^{0}
U-234	1.12 10^{-3}	2.95 10^{0}
U-235	1.12 10^{-3}	2.95 10^{0}
U-236	1.12 10^{-3}	2.95 10^{0}
U-238	1.12 10^{-3}	2.95 10^{0}
Np-237	6.34 10^{-3}	1.68 10^{1}
Pu-238	2.99 10^{-4}	7.92 10^{-1}
Pu-239	3.02 10^{-4}	7.98 10^{-1}
Pu-240	3.02 10^{-4}	7.98 10^{-1}
Pu-241	2.88 10^{-4}	7.61 10^{-1}
Pu-242	3.02 10^{-4}	7.98 10^{-1}
Pu-244	3.02 10^{-4}	7.98 10^{-1}
Am-241	8.57 10^{-5}	2.27 10^{-1}
Am-242m	8.55 10^{-5}	2.26 10^{-1}
Am-243	8.59 10^{-5}	2.27 10^{-1}
Cm-242	6.45 10^{-6}	1.71 10^{-2}
Cm-243	2.34 10^{-5}	6.20 10^{-2}
Cm-244	2.31 10^{-5}	6.12 10^{-2}
Cm-245	2.40 10^{-5}	6.35 10^{-2}
Cm-246	2.40 10^{-5}	6.35 10^{-2}
Cm-248	2.40 10^{-5}	6.35 10^{-2}

E.3.7.2. Assessment calculations for off-site leachate treatment

660. The pathways for exposure to radiation of the hazardous waste treatment facility worker and the sewage treatment plant worker are assumed to be similar and the dose assessment is based on the EA Initial Radiological Assessment Methodology (Environment Agency, 2006a) and (Environment Agency, 2006b) for discharge to a sewage treatment plant. The default IRAM calculations are based on generic data and provide a cautious estimate of the radiation dose arising to various exposed groups. The Environment Agency IRAM model assumes a default volume throughput at the

sewage works of 60 m^3 day^{-1}. This is based on a small sewage treatment works serving about 500 people. In contrast, the Bran Sands treatment facility has a throughput of about 3 10^5 m^3 day^{-1}. This means that the radionuclide activity concentrations in the discharges and sewage sludge would be substantially lower than those assumed in the default case. This has been represented in the calculations (see equations below).

Treatment Facility Worker

661. Members of the exposed group are assumed to be adults working at a treatment plant and to be exposed as a result of:

- external radiation from radionuclides in raw effluent and sludge;
- inadvertent inhalation of raw effluent and sludge; and,
- inadvertent ingestion of raw effluent and sludge.

662. The EA methodology was used to produce tables of Dose Per Unit Release (DPUR; μSv y^{-1} per Bq y^{-1} discharge) that are then used to obtain doses from discharges. The assessment model is described below. It uses leachate contamination levels derived from the GoldSim groundwater model (see Section E.4.3) and a realistic throughput for the treatment works.

Table 80 Treatment plant worker characteristics

Parameter	Value	Comment
Time at plant (h y^{-1})	2000	Standard assumption in [(Environment Agency, 2006a), (Environment Agency, 2006b)].
Proportion near treatment tanks	0.25	
Dust in air from sewage/sludge (kg m^{-3})	1 10^{-7}	
Inhalation rate (m^3 h^{-1})	1.2	
Inadvertent sludge ingestion (mg h^{-1})	5	

663. The radiation dose incurred by an adult treatment plant worker for each radionuclide ($Dose_{Rn,worker}$) is given by:

$$Dose_{Rn,worker} = F_{Rn} \cdot DF_{Rn,worker} \cdot Dil$$

where:

- F_{Rn} is the flux of the radionuclide to the treatment works (Bq y^{-1});
- $DF_{Rn,\ worker}$ is the dose per unit flux to the given exposed group (Sv y^{-1} per Bq y^{-1}) based on default assumptions – Total DPUR calculated using EA methodology and reproduced in Table 81 (adult sewage worker) ; and,
- Dil is a dilution factor that is given by the ratio of the default and actual treatment throughputs.

Table 81 Dose per unit release factors for effluent treatment workers – leachate to treatment facility scenario (μSv/y per Bq/y of discharge to leachate) calculated using the EA IRAM methodology

Radionuclide	External irradiation DPUR	Inadvertent ingestion and inhalation DPUR	Total DPUR
H-3	0	$1.45\ 10^{-13}$	$1.45\ 10^{-13}$
C-14	$1.87\ 10^{-13}$	$4.32\ 10^{-12}$	$4.50\ 10^{-12}$
Cl-36	$2.40\ 10^{-11}$	$4.56\ 10^{-12}$	$2.85\ 10^{-11}$
Ca-41	0	$5.90\ 10^{-12}$	$5.90\ 10^{-12}$
Mn-54	$2.05\ 10^{-7}$	$1.39\ 10^{-11}$	$2.05\ 10^{-7}$
Fe-55	0	$1.18\ 10^{-11}$	$1.18\ 10^{-11}$
Co-60	$1.03\ 10^{-6}$	$1.25\ 10^{-10}$	$1.03\ 10^{-6}$
Ni-59	0	$1.41\ 10^{-12}$	$1.41\ 10^{-12}$
Ni-63	0	$3.46\ 10^{-12}$	$3.46\ 10^{-12}$
Zn-65	$1.46\ 10^{-7}$	$7.28\ 10^{-11}$	$1.46\ 10^{-7}$
Se-79	$7.60\ 10^{-13}$	$5.86\ 10^{-11}$	$5.93\ 10^{-11}$
Sr-90	$2.47\ 10^{-10}$	$1.43\ 10^{-10}$	$3.89\ 10^{-10}$
Mo-93	$5.92\ 10^{-12}$	$1.30\ 10^{-11}$	$1.89\ 10^{-11}$
Zr-93	0	$5.80\ 10^{-11}$	$5.80\ 10^{-11}$
Nb-93m	$4.24\ 10^{-12}$	$3.12\ 10^{-12}$	$7.36\ 10^{-12}$
Nb-94	$3.95\ 10^{-7}$	$5.50\ 10^{-11}$	$3.95\ 10^{-7}$
Tc-99	$1.26\ 10^{-12}$	$3.93\ 10^{-12}$	$5.18\ 10^{-12}$
Ru-106	$1.27\ 10^{-8}$	$3.46\ 10^{-11}$	$1.27\ 10^{-8}$
Ag-108m	$6.91\ 10^{-7}$	$1.09\ 10^{-10}$	$6.91\ 10^{-7}$
Ag-110m	$1.19\ 10^{-6}$	$1.02\ 10^{-10}$	$1.19\ 10^{-6}$
Cd-109	$1.45\ 10^{-10}$	$8.88\ 10^{-12}$	$1.54\ 10^{-10}$
Sb-125	$1.55\ 10^{-7}$	$4.84\ 10^{-11}$	$1.55\ 10^{-7}$
Sn-119m	$1.19\ 10^{-10}$	$7.28\ 10^{-12}$	$1.26\ 10^{-10}$
Sn-123	$1.93\ 10^{-9}$	$4.09\ 10^{-11}$	$1.97\ 10^{-9}$
Sn-126	$4.84\ 10^{-7}$	$1.10\ 10^{-10}$	$4.84\ 10^{-7}$
Te-127m	$2.05\ 10^{-10}$	$4.45\ 10^{-11}$	$2.50\ 10^{-10}$
I-129	$2.30\ 10^{-10}$	$8.73\ 10^{-10}$	$1.10\ 10^{-9}$
Ba-133	$1.98\ 10^{-8}$	$7.16\ 10^{-12}$	$1.98\ 10^{-8}$
Cs-134	$2.38\ 10^{-7}$	$2.24\ 10^{-10}$	$2.38\ 10^{-7}$
Cs-135	$9.74\ 10^{-13}$	$2.56\ 10^{-11}$	$2.66\ 10^{-11}$
Cs-137	$8.65\ 10^{-8}$	$1.62\ 10^{-10}$	$8.67\ 10^{-8}$
Ce-144	$2.84\ 10^{-9}$	$1.20\ 10^{-10}$	$2.96\ 10^{-9}$
Pm-147	$2.03\ 10^{-12}$	$7.20\ 10^{-12}$	$9.22\ 10^{-12}$

Radionuclide	External irradiation DPUR	Inadvertent ingestion and inhalation DPUR	Total DPUR
Sm-147	0	$5.34\ 10^{-9}$	$5.34\ 10^{-9}$
Sm-151	$4.02\ 10^{-14}$	$3.71\ 10^{-12}$	$3.75\ 10^{-12}$
Eu-152	$2.86\ 10^{-7}$	$4.60\ 10^{-11}$	$2.86\ 10^{-7}$
Eu-154	$3.13\ 10^{-7}$	$6.24\ 10^{-11}$	$3.13\ 10^{-7}$
Eu-155	$7.40\ 10^{-9}$	$9.24\ 10^{-12}$	$7.41\ 10^{-9}$
Gd-153	$9.64\ 10^{-9}$	$5.90\ 10^{-12}$	$9.64\ 10^{-9}$
Pb-210	$4.37\ 10^{-10}$	$7.26\ 10^{-8}$	$7.31\ 10^{-8}$
Po-210	$3.51\ 10^{-12}$	$4.16\ 10^{-8}$	$4.16\ 10^{-8}$
Ra-226	$4.57\ 10^{-7}$	$5.05\ 10^{-8}$	$5.08\ 10^{-7}$
Ra-228	$1.62\ 10^{-7}$	$9.31\ 10^{-9}$	$1.71\ 10^{-7}$
Ac-227	$1.18\ 10^{-7}$	$5.06\ 10^{-7}$	$6.24\ 10^{-7}$
Th-228	$7.21\ 10^{-7}$	$4.01\ 10^{-8}$	$7.61\ 10^{-7}$
Th-229	$1.14\ 10^{-7}$	$2.31\ 10^{-7}$	$3.45\ 10^{-7}$
Th-230	$8.66\ 10^{-11}$	$8.91\ 10^{-8}$	$8.92\ 10^{-8}$
Th-232	$4.33\ 10^{-7}$	$1.75\ 10^{-7}$	$6.08\ 10^{-7}$
Pa-231	$1.37\ 10^{-8}$	$1.39\ 10^{-7}$	$1.53\ 10^{-7}$
U-232	$9.04\ 10^{-12}$	$9.93\ 10^{-9}$	$9.94\ 10^{-9}$
U-233	$1.40\ 10^{-11}$	$1.16\ 10^{-9}$	$1.17\ 10^{-9}$
U-234	$4.02\ 10^{-12}$	$1.13\ 10^{-9}$	$1.14\ 10^{-9}$
U-235	$7.23\ 10^{-9}$	$1.04\ 10^{-9}$	$8.26\ 10^{-9}$
U-236	$2.15\ 10^{-12}$	$1.06\ 10^{-9}$	$1.06\ 10^{-9}$
U-238	$1.03\ 10^{-12}$	$9.92\ 10^{-10}$	$9.93\ 10^{-10}$
Np-237	$3.18\ 10^{-9}$	$2.51\ 10^{-8}$	$2.83\ 10^{-8}$
Pu-238	$6.18\ 10^{-12}$	$5.49\ 10^{-8}$	$5.49\ 10^{-8}$
Pu-239	$1.21\ 10^{-11}$	$5.99\ 10^{-8}$	$5.99\ 10^{-8}$
Pu-240	$5.99\ 10^{-12}$	$5.99\ 10^{-8}$	$5.99\ 10^{-8}$
Pu-241	$2.41\ 10^{-13}$	$1.15\ 10^{-9}$	$1.15\ 10^{-9}$
Pu-242	$5.23\ 10^{-12}$	$5.51\ 10^{-8}$	$5.51\ 10^{-8}$
Pu-244	$3.08\ 10^{-12}$	$5.51\ 10^{-8}$	$5.51\ 10^{-8}$
Am-241	$3.13\ 10^{-9}$	$8.54\ 10^{-8}$	$8.86\ 10^{-8}$
Am-242m	$1.21\ 10^{-10}$	$1.03\ 10^{-7}$	$1.03\ 10^{-7}$
Am-243	$1.02\ 10^{-8}$	$8.55\ 10^{-8}$	$9.57\ 10^{-8}$
Cm-242	$1.16\ 10^{-11}$	$4.95\ 10^{-9}$	$4.96\ 10^{-9}$
Cm-243	$4.17\ 10^{-8}$	$6.18\ 10^{-8}$	$1.04\ 10^{-7}$
Cm-244	$9.01\ 10^{-12}$	$5.07\ 10^{-8}$	$5.07\ 10^{-8}$
Cm-245	$2.44\ 10^{-8}$	$8.82\ 10^{-8}$	$1.13\ 10^{-7}$

Radionuclide	External irradiation DPUR	Inadvertent ingestion and inhalation DPUR	Total DPUR
Cm-246	8.33 10^{-12}	8.74 10^{-8}	8.74 10^{-8}
Cm-248	6.29 10^{-12}	3.21 10^{-7}	3.21 10^{-7}

664. The doses to leachate treatment facility workers are presented in Section E.3.7.3.

Farming family (soil treated with sludge)

665. Farm land is assumed to be treated repeatedly with contaminated sludge from the treatment works. The assessment of doses to a farming family using the treated land is based on the EA Initial Radiological Assessment Methodology (Environment Agency, 2006a) and (Environment Agency, 2006b). Members of the exposed group are assumed to be adults and the exposure pathways considered are:

- consumption of food produced on land conditioned with sludge and incorporating radionuclides, including milk, green vegetables, root vegetables and meat products;
- external irradiation from radionuclides incorporated in surface layers of sludge conditioned soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

666. The characteristics of the group are based on the EA methodology, with some modification to allow for more realistic rates of sludge application and food consumption.

667. The organic matter content of soil is an important part of its fertility. Farmers aim to enhance soil organic matter by reducing losses, minimising cultivations and adding organic carbon. Application of sewage sludge (commonly referred to as 'biosolids') to agricultural land is one method of maintaining soil organic matter but it is highly regulated. The application of solid or liquid sewage sludge is limited by many factors, including time of year, pH, potentially toxic element content, use of land and proximity to watercourses. It is common for the rate of application of biosolids to be limited in total to around 50 t ha^{-1} y^{-1}, equivalent to 5 kg m^{-2} y^{-1} (Defra, 2009). It is assumed that this rate also applies to sludge from the treatment facility.

668. Parameters characterising the application of treated sludge to agricultural land are summarised in Table 82. The area of land treated is not defined but is assumed to be sufficient to support food production at the levels implied by intake rates presented in Table 83.

Table 82 Parameters characterising the application of treated sludge to agricultural land: applicable during the Period of Authorisation

Parameter	Value	Comment
Rate of application of treated sludge ($kg\ m^{-2}\ y^{-1}$)	5	Amended from the DPUR default value of 8 $kg\ m^{-2}\ y^{-1}$ to comply with UK practice.
Delay between spreading sludge and animal grazing (d)	21	Standard assumption in [(Environment Agency, 2006a), (Environment Agency, 2006b)].
Delay between spreading sludge and animal grazing (d)	300	
Density of soil ($kg\ m^{-3}$)	1,250	
Transfer of strontium to next soil layer (y^{-1})	0.464	
Transfer of other radionuclides to next soil layer (y^{-1})	0.243	
Dust in air ($kg\ m^{-3}$)	$1\ 10^{-7}$	

669. Habit data for the farming family are summarised in Table 83. The habit data values used in the ESC for the inhalation and external exposure pathways differ from those in the DPUR, except for the child and infant inhalation rates. Scaling factors (F_P) were determined by dividing the ESC value by the value used in the EA IRAM to derive CPUR values.

670. The assessment uses the leachate contamination levels derived from the GoldSim groundwater model (see Section E.4.3) and a realistic throughput for the treatment works.

671. Consumption rates assumed for the farming family using biosolids from the treatment facility are consistent with the approach used throughout this report: the two most limiting pathways use consumption rates at the 97.5th percentile rate and average rates are used for consumption of all other foods. The Environment Agency IRAM adopted 97.5th percentile consumption rates for all foods and hence they use different values, see Table 83. Scaling factors for consumption (F_P) have been determined by dividing the assumed consumption rates by the EA IRAM default consumption rates and scaling is only applied to the pathways ranked third and below. The values for the mean and 97.5th percentile consumption rates are the generalised intake rates produced by the NRPB (Smith & Jones, 2003).

672. A biosolids application rate of 8 $kg\ m^{-2}\ y^{-1}$ was used as the default value in the Environment Agency IRAM methodology and hence the results are scaled to the assumed application rate of 5 $kg\ m^{-2}\ y^{-1}$ (F_{SAR} = 0.625) as discussed above (paragraph 667).

Table 83 Farming family habits data

Habit data	Adult			Child			Infant		
	DPUR basis	Mean	97.5th	DPUR basis	Mean	97.5th	DPUR basis	Mean	97.5th
Green vegetables ($kg\ y^{-1}$)	80	35	80	35	15	35	15	5	15
Root vegetables ($kg\ y^{-1}$)	130	65	130	95	50	95	45	15	45
Sheep meat ($kg\ y^{-1}$)	25	8	25	10	4	10	3	0.8	3
Sheep liver ($kg\ y^{-1}$)	10	2.75	10	5	1.5	5	2.75	0.5	2.75
Cow meat ($kg\ y^{-1}$)	45	15	45	30	15	30	10	3	10
Cow liver ($kg\ y^{-1}$)	10	2.75	10	5	1.5	5	2.75	0.5	2.75
Milk ($kg\ y^{-1}$)	240	122.5	240	240	110	240	320	130	320
Soil ingestion (inadvertent) ($kg\ y^{-1}$)	$7\ 10^{-5}$	$3\ 10^{-2}$	$3.0\ 10^{-2}$	$6\ 10^{-5}$	$2\ 10^{-2}$	$2\ 10^{-2}$	$7\ 10^{-5}$	$4\ 10^{-2}$	$4\ 10^{-2}$
Inhalation rate ($m^3\ h^{-1}$)	0.92	1.0		0.64	0.64		0.22	0.22	
Outdoor occupancy	0.5	0.25		0.2	0.16		0.1	0.09	

673. The radiation dose incurred by a farmer for each radionuclide ($D_{Rn,farmer}$) is given by:

$$Dose_{Rn,farmer,P} = F_{Rn} \cdot DF_{Rn,farmer} \cdot Dil \cdot F_{SAR} \cdot F_P \cdot (1 - F_E)$$

$$Dose_{Rn,farmer} = \sum_P Dose_{Rn,farmer,P}$$

where:

- F_{Rn} is the flux of the radionuclide to the treatment works ($Bq\ y^{-1}$), assuming no loss during leachate treatment;
- $DF_{Rn,\ farmer}$ is the dose per unit flux to the given exposed group ($Sv\ y^{-1}$ per $Bq\ y^{-1}$) using default values – as indicated in Table 84, Table 85 and Table 86 for adult, child and infant respectively from (Environment Agency, 2006b);
- Dil is a dilution factor that is given by the ratio of the assumed and actual throughputs;
- F_{SAR} is the scaling factor for the sewage application rate;
- F_P is the scaling factor for the specific pathway P; and,

- F_E is the fraction (Table 88) from raw sewage that is disposed in liquid effluent (the rest is disposed with biosolids).

674. The doses to each member of a farming family are presented in Section E.3.7.3.

Table 84 Dose per unit release factors for adult member of the farming family – leachate release to treatment facility scenario ($\mu Sv\ y^{-1}$ per $Bq\ y^{-1}$) given in the EA IRAM methodology

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
H-3	0	0	$1.90\ 10^{-12}$	$7.50\ 10^{-13}$	$2.20\ 10^{-12}$	$4.90\ 10^{-13}$	$1.40\ 10^{-11}$	0	$1.10\ 10^{-17}$	$1.80\ 10^{-18}$
C-14	$1.80\ 10^{-8}$	$2.40\ 10^{-8}$	$3.30\ 10^{-9}$	$1.30\ 10^{-9}$	$3.70\ 10^{-9}$	$8.30\ 10^{-10}$	$9.60\ 10^{-9}$	$2.70\ 10^{-13}$	$1.40\ 10^{-13}$	$6.70\ 10^{-15}$
Cl-36	$7.90\ 10^{-8}$	$1.00\ 10^{-7}$	$5.30\ 10^{-8}$	$2.10\ 10^{-8}$	$8.30\ 10^{-8}$	$1.80\ 10^{-8}$	$8.60\ 10^{-8}$	$4.40\ 10^{-10}$	$3.50\ 10^{-12}$	$7.70\ 10^{-14}$
Ca-41	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Mn-54	$1.80\ 10^{-10}$	$2.00\ 10^{-10}$	$7.60\ 10^{-10}$	$1.40\ 10^{-8}$	$9.80\ 10^{-10}$	$1.00\ 10^{-8}$	$4.00\ 10^{-9}$	$9.40\ 10^{-7}$	$7.90\ 10^{-13}$	$6.40\ 10^{-14}$
Fe-55	$8.10\ 10^{-11}$	$2.90\ 10^{-12}$	$1.20\ 10^{-11}$	$1.50\ 10^{-10}$	$1.30\ 10^{-11}$	$1.10\ 10^{-8}$	$8.20\ 10^{-11}$	0	$7.90\ 10^{-13}$	$1.20\ 10^{-13}$
Co-60	$1.90\ 10^{-9}$	$1.80\ 10^{-9}$	$9.50\ 10^{-10}$	$3.80\ 10^{-8}$	$9.10\ 10^{-10}$	$2.00\ 10^{-8}$	$1.20\ 10^{-8}$	$1.40\ 10^{-5}$	$2.50\ 10^{-11}$	$1.40\ 10^{-12}$
Ni-59	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ni-63	$2.00\ 10^{-10}$	$2.80\ 10^{-10}$	$5.10\ 10^{-14}$	$2.00\ 10^{-13}$	$2.20\ 10^{-13}$	$4.90\ 10^{-13}$	$3.70\ 10^{-12}$	0	$1.10\ 10^{-12}$	$6.00\ 10^{-14}$
Zn-65	$4.60\ 10^{-9}$	$3.50\ 10^{-9}$	$3.80\ 10^{-9}$	$1.50\ 10^{-9}$	$7.90\ 10^{-9}$	$1.80\ 10^{-9}$	$5.00\ 10^{-7}$	$5.50\ 10^{-7}$	$6.80\ 10^{-13}$	$2.80\ 10^{-13}$
Se-79	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sr-90	$1.30\ 10^{-7}$	$3.60\ 10^{-8}$	$1.20\ 10^{-9}$	$4.80\ 10^{-10}$	$3.30\ 10^{-9}$	$7.30\ 10^{-10}$	$8.30\ 10^{-8}$	$3.60\ 10^{-9}$	$8.60\ 10^{-12}$	$1.20\ 10^{-12}$
Mo-93	nd	nd	nd	nd	nd	nd	nd	nd	Nd	nd
Zr-93	nd	nd	nd	nd	nd	nd	nd	nd	Nd	nd
Nb-93m	nd	nd	nd	nd	nd	nd	nd	nd	Nd	nd
Nb-94	nd	nd	nd	nd	nd	nd	nd	nd	Nd	nd
Tc-99	$5.40\ 10^{-8}$	$7.10\ 10^{-8}$	$2.50\ 10^{-7}$	$3.00\ 10^{-7}$	$7.00\ 10^{-8}$	$6.20\ 10^{-8}$	$3.80\ 10^{-7}$	$1.90\ 10^{-11}$	$1.90\ 10^{-12}$	$5.30\ 10^{-14}$
Ru-106	$1.50\ 10^{-10}$	$5.30\ 10^{-11}$	$5.80\ 10^{-11}$	$2.30\ 10^{-11}$	$4.90\ 10^{-11}$	$1.10\ 10^{-11}$	$4.90\ 10^{-13}$	$5.50\ 10^{-8}$	$3.40\ 10^{-12}$	$1.40\ 10^{-13}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Ag-108m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ag-110m	$1.50\ 10^{-9}$	$1.90\ 10^{-9}$	$1.60\ 10^{-9}$	$2.00\ 10^{-7}$	$2.40\ 10^{-9}$	$2.10\ 10^{-7}$	$4.40\ 10^{-7}$	$4.60\ 10^{-6}$	$5.90\ 10^{-12}$	$3.70\ 10^{-13}$
Cd-109	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sb-125	$4.10\ 10^{-10}$	$2.80\ 10^{-10}$	$3.70\ 10^{-10}$	$1.50\ 10^{-8}$	$2.40\ 10^{-8}$	$5.30\ 10^{-9}$	$1.30\ 10^{-10}$	$1.60\ 10^{-6}$	$9.00\ 10^{-12}$	$3.50\ 10^{-13}$
Sn-119m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
I-129	$1.30\ 10^{-7}$	$2.00\ 10^{-7}$	$1.70\ 10^{-7}$	$6.60\ 10^{-8}$	$3.60\ 10^{-8}$	$7.90\ 10^{-9}$	$2.90\ 10^{-7}$	$3.40\ 10^{-9}$	$3.50\ 10^{-11}$	$1.80\ 10^{-11}$
Ba-133	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-134	$2.20\ 10^{-9}$	$2.70\ 10^{-9}$	$5.00\ 10^{-8}$	$2.00\ 10^{-8}$	$4.10\ 10^{-8}$	$9.10\ 10^{-9}$	$4.40\ 10^{-8}$	$1.90\ 10^{-6}$	$3.90\ 10^{-12}$	$1.90\ 10^{-12}$
Cs-135	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-137	$6.00\ 10^{-9}$	$9.20\ 10^{-9}$	$8.10\ 10^{-8}$	$3.30\ 10^{-8}$	$5.30\ 10^{-8}$	$1.20\ 10^{-8}$	$5.40\ 10^{-8}$	$1.60\ 10^{-6}$	$6.10\ 10^{-12}$	$2.90\ 10^{-12}$
Ce-144	$3.70\ 10^{-10}$	$1.30\ 10^{-11}$	$1.90\ 10^{-11}$	$1.50\ 10^{-9}$	$1.60\ 10^{-11}$	$7.20\ 10^{-10}$	$2.80\ 10^{-11}$	$5.80\ 10^{-8}$	$1.70\ 10^{-11}$	$4.30\ 10^{-13}$
Pm-147	$4.00\ 10^{-11}$	$1.10\ 10^{-11}$	$1.50\ 10^{-11}$	$3.60\ 10^{-11}$	$1.90\ 10^{-11}$	$3.40\ 10^{-11}$	$2.90\ 10^{-12}$	$1.80\ 10^{-11}$	$5.70\ 10^{-12}$	$5.10\ 10^{-14}$
Sm-147	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Eu-152	$4.20\ 10^{-10}$	$3.40\ 10^{-10}$	$1.60\ 10^{-10}$	$3.90\ 10^{-10}$	$3.10\ 10^{-10}$	$5.50\ 10^{-10}$	$2.80\ 10^{-11}$	$4.80\ 10^{-6}$	$8.30\ 10^{-11}$	$4.80\ 10^{-13}$
Eu-154	$5.10\ 10^{-10}$	$3.30\ 10^{-10}$	$2.10\ 10^{-10}$	$5.00\ 10^{-10}$	$3.70\ 10^{-10}$	$6.50\ 10^{-10}$	$3.60\ 10^{-11}$	$4.90\ 10^{-6}$	$9.60\ 10^{-11}$	$6.20\ 10^{-13}$
Eu-155	$6.40\ 10^{-11}$	$3.00\ 10^{-11}$	$2.60\ 10^{-11}$	$6.30\ 10^{-11}$	$4.10\ 10^{-11}$	$7.20\ 10^{-11}$	$4.70\ 10^{-12}$	$9.00\ 10^{-8}$	$1.00\ 10^{-11}$	$8.30\ 10^{-14}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Gd-153	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Pb-210	$1.10\ 10^{-6}$	$1.40\ 10^{-6}$	$2.90\ 10^{-7}$	$2.20\ 10^{-7}$	$3.50\ 10^{-7}$	$1.60\ 10^{-7}$	$7.00\ 10^{-7}$	$1.10\ 10^{-8}$	$4.20\ 10^{-9}$	$4.50\ 10^{-10}$
Po-210	$7.60\ 10^{-8}$	$7.70\ 10^{-8}$	$2.60\ 10^{-7}$	$1.30\ 10^{-6}$	$5.50\ 10^{-8}$	$3.40\ 10^{-7}$	$1.30\ 10^{-8}$	$8.10\ 10^{-12}$	$1.50\ 10^{-9}$	$9.10\ 10^{-11}$
Ra-226	$4.20\ 10^{-7}$	$6.10\ 10^{-8}$	$7.50\ 10^{-8}$	$3.00\ 10^{-8}$	$6.40\ 10^{-8}$	$1.40\ 10^{-8}$	$2.80\ 10^{-7}$	$9.40\ 10^{-6}$	$8.40\ 10^{-9}$	$1.20\ 10^{-10}$
Ra-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-230	$8.90\ 10^{-8}$	$4.20\ 10^{-8}$	$4.10\ 10^{-9}$	$1.60\ 10^{-8}$	$4.20\ 10^{-9}$	$9.30\ 10^{-9}$	$1.70\ 10^{-9}$	$1.70\ 10^{-9}$	$6.10\ 10^{-8}$	$1.60\ 10^{-10}$
Th-232	$9.70\ 10^{-8}$	$4.60\ 10^{-8}$	$4.40\ 10^{-9}$	$1.80\ 10^{-8}$	$4.60\ 10^{-9}$	$1.00\ 10^{-8}$	$1.80\ 10^{-9}$	$2.50\ 10^{-5}$	$1.10\ 10^{-7}$	$1.70\ 10^{-10}$
Pa-231	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-232	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-234	$3.00\ 10^{-9}$	$2.20\ 10^{-9}$	$8.90\ 10^{-10}$	$3.60\ 10^{-10}$	$3.00\ 10^{-10}$	$6.60\ 10^{-11}$	$4.70\ 10^{-9}$	$6.10\ 10^{-11}$	$1.70\ 10^{-9}$	$4.10\ 10^{-12}$
U-235	$2.80\ 10^{-9}$	$2.10\ 10^{-9}$	$8.50\ 10^{-10}$	$3.40\ 10^{-10}$	$2.80\ 10^{-10}$	$6.30\ 10^{-11}$	$4.50\ 10^{-9}$	$1.20\ 10^{-7}$	$1.50\ 10^{-9}$	$3.90\ 10^{-12}$
U-236	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-238	$2.70\ 10^{-9}$	$2.00\ 10^{-9}$	$8.20\ 10^{-10}$	$3.30\ 10^{-10}$	$2.70\ 10^{-10}$	$6.10\ 10^{-11}$	$4.30\ 10^{-9}$	$2.50\ 10^{-8}$	$1.40\ 10^{-9}$	$3.70\ 10^{-12}$
Np-237	$4.80\ 10^{-8}$	$2.40\ 10^{-8}$	$3.00\ 10^{-9}$	$8.70\ 10^{-8}$	$7.50\ 10^{-9}$	$2.00\ 10^{-7}$	$7.20\ 10^{-10}$	$9.00\ 10^{-7}$	$5.50\ 10^{-8}$	$4.60\ 10^{-11}$
Pu-238	$3.80\ 10^{-8}$	$2.10\ 10^{-9}$	$3.70\ 10^{-9}$	$1.00\ 10^{-7}$	$3.80\ 10^{-9}$	$1.00\ 10^{-7}$	$1.40\ 10^{-9}$	$1.00\ 10^{-10}$	$1.10\ 10^{-7}$	$9.20\ 10^{-11}$
Pu-239	$4.20\ 10^{-8}$	$2.80\ 10^{-9}$	$4.20\ 10^{-9}$	$1.20\ 10^{-7}$	$4.30\ 10^{-9}$	$1.20\ 10^{-7}$	$1.60\ 10^{-9}$	$2.30\ 10^{-10}$	$1.20\ 10^{-7}$	$1.00\ 10^{-10}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Pu-240	$4.20\ 10^{-8}$	$2.80\ 10^{-9}$	$4.20\ 10^{-9}$	$1.20\ 10^{-7}$	$4.30\ 10^{-9}$	$1.10\ 10^{-7}$	$1.60\ 10^{-9}$	$1.00\ 10^{-10}$	$1.20\ 10^{-7}$	$1.00\ 10^{-10}$
Pu-241	$7.50\ 10^{-10}$	$2.00\ 10^{-11}$	$6.30\ 10^{-11}$	$1.80\ 10^{-9}$	$5.50\ 10^{-11}$	$1.60\ 10^{-9}$	$2.10\ 10^{-11}$	$3.90\ 10^{-12}$	$1.80\ 10^{-9}$	$1.70\ 10^{-12}$
Pu-242	$4.00\ 10^{-8}$	$2.60\ 10^{-9}$	$4.00\ 10^{-9}$	$1.10\ 10^{-7}$	$4.10\ 10^{-9}$	$1.10\ 10^{-7}$	$1.60\ 10^{-9}$	$8.80\ 10^{-11}$	$1.20\ 10^{-7}$	$9.90\ 10^{-11}$
Pu-244	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-241	$6.20\ 10^{-8}$	$6.10\ 10^{-9}$	$6.20\ 10^{-9}$	$1.80\ 10^{-7}$	$7.70\ 10^{-9}$	$2.00\ 10^{-7}$	$3.00\ 10^{-9}$	$5.90\ 10^{-8}$	$1.80\ 10^{-7}$	$1.50\ 10^{-10}$
Am-242m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-243	$6.20\ 10^{-8}$	$6.30\ 10^{-9}$	$6.20\ 10^{-9}$	$1.80\ 10^{-7}$	$7.70\ 10^{-9}$	$2.10\ 10^{-7}$	$3.00\ 10^{-9}$	$1.30\ 10^{-6}$	$1.80\ 10^{-7}$	$1.50\ 10^{-10}$
Cm-242	$8.00\ 10^{-10}$	$4.90\ 10^{-13}$	$1.80\ 10^{-11}$	$5.10\ 10^{-10}$	$6.60\ 10^{-12}$	$1.80\ 10^{-10}$	$2.50\ 10^{-12}$	$2.50\ 10^{-11}$	$2.70\ 10^{-9}$	$1.10\ 10^{-12}$
Cm-243	$4.40\ 10^{-8}$	$1.00\ 10^{-9}$	$4.50\ 10^{-9}$	$1.30\ 10^{-7}$	$5.50\ 10^{-9}$	$1.50\ 10^{-7}$	$5.30\ 10^{-10}$	$7.80\ 10^{-7}$	$1.20\ 10^{-7}$	$1.00\ 10^{-10}$
Cm-244	$3.50\ 10^{-8}$	$6.50\ 10^{-10}$	$3.10\ 10^{-9}$	$9.00\ 10^{-8}$	$3.30\ 10^{-9}$	$9.00\ 10^{-8}$	$1.20\ 10^{-9}$	$1.20\ 10^{-10}$	$1.00\ 10^{-7}$	$7.70\ 10^{-11}$
Cm-245	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Table 85 Dose per unit release factors for child member of the farming family – leachate release to treatment facility scenario ($\mu Sv\ y^{-1}$ per $Bq\ y^{-1}$) given in the EA IRAM methodology

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
H-3	0	0	$9.60\ 10^{-13}$	$4.80\ 10^{-13}$	$1.90\ 10^{-12}$	$3.10\ 10^{-13}$	$1.70\ 10^{-11}$	0	$3.80\ 10^{-18}$	$2.00\ 10^{-18}$
C-14	$1.10\ 10^{-8}$	$2.40\ 10^{-8}$	$1.80\ 10^{-9}$	$9.10\ 10^{-10}$	$3.40\ 10^{-9}$	$5.70\ 10^{-10}$	$1.30\ 10^{-8}$	$1.40\ 10^{-13}$	$5.30\ 10^{-14}$	$8.10\ 10^{-15}$
Cl-36	$7.00\ 10^{-8}$	$1.50\ 10^{-7}$	$4.30\ 10^{-8}$	$2.20\ 10^{-8}$	$1.10\ 10^{-7}$	$1.90\ 10^{-8}$	$1.80\ 10^{-7}$	$2.20\ 10^{-10}$	$1.30\ 10^{-12}$	$1.40\ 10^{-13}$
Ca-41	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Mn-54	$1.40\ 10^{-10}$	$2.70\ 10^{-10}$	$5.50\ 10^{-10}$	$1.20\ 10^{-8}$	$1.20\ 10^{-9}$	$9.10\ 10^{-9}$	$7.20\ 10^{-9}$	$4.80\ 10^{-7}$	$3.50\ 10^{-13}$	$1.00\ 10^{-13}$
Fe-55	$1.20\ 10^{-10}$	$7.00\ 10^{-12}$	$1.60\ 10^{-11}$	$2.50\ 10^{-10}$	$2.80\ 10^{-11}$	$1.90\ 10^{-8}$	$2.70\ 10^{-10}$	0	$3.60\ 10^{-13}$	$3.40\ 10^{-13}$
Co-60	$2.70\ 10^{-9}$	$4.30\ 10^{-9}$	$1.20\ 10^{-9}$	$6.20\ 10^{-8}$	$2.00\ 10^{-9}$	$3.20\ 10^{-8}$	$3.80\ 10^{-8}$	$7.20\ 10^{-6}$	$1.00\ 10^{-11}$	$4.10\ 10^{-12}$
Ni-59	nd	nd	nd	nd	nd	nd	nd	nd	nd	Nd
Ni-63	$1.60\ 10^{-10}$	$3.80\ 10^{-10}$	$3.80\ 10^{-14}$	$1.90\ 10^{-13}$	$2.80\ 10^{-13}$	$4.60\ 10^{-13}$	$6.80\ 10^{-12}$	0	$4.60\ 10^{-13}$	$9.80\ 10^{-14}$
Zn-65	$3.30\ 10^{-9}$	$4.20\ 10^{-9}$	$2.50\ 10^{-9}$	$1.30\ 10^{-9}$	$8.60\ 10^{-9}$	$1.40\ 10^{-9}$	$8.20\ 10^{-7}$	$2.80\ 10^{-7}$	$2.80\ 10^{-13}$	$4.00\ 10^{-13}$
Se-79	nd	nd	nd	nd	nd	nd	nd	nd	nd	Nd
Sr-90	$1.20\ 10^{-7}$	$5.70\ 10^{-8}$	$1.00\ 10^{-9}$	$5.10\ 10^{-10}$	$4.70\ 10^{-9}$	$7.80\ 10^{-10}$	$1.80\ 10^{-7}$	$1.80\ 10^{-9}$	$3.40\ 10^{-12}$	$2.10\ 10^{-12}$
Mo-93	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Zr-93	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Nb-93m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Nb-94	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Tc-99	$4.80\ 10^{-8}$	$1.10\ 10^{-7}$	$2.00\ 10^{-7}$	$3.00\ 10^{-7}$	$9.50\ 10^{-8}$	$6.30\ 10^{-8}$	$7.60\ 10^{-7}$	$9.80\ 10^{-12}$	$7.70\ 10^{-13}$	$9.30\ 10^{-14}$
Ru-106	$1.40\ 10^{-10}$	$8.30\ 10^{-11}$	$5.00\ 10^{-11}$	$2.50\ 10^{-11}$	$7.00\ 10^{-11}$	$1.20\ 10^{-11}$	$1.10\ 10^{-12}$	$2.80\ 10^{-8}$	$1.40\ 10^{-12}$	$2.70\ 10^{-13}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Ag-108m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ag-110m	$1.20\ 10^{-9}$	$2.60\ 10^{-9}$	$1.20\ 10^{-9}$	$1.80\ 10^{-7}$	$2.90\ 10^{-9}$	$1.90\ 10^{-7}$	$8.20\ 10^{-7}$	$2.40\ 10^{-6}$	$2.60\ 10^{-12}$	$6.00\ 10^{-13}$
Cd-109	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sb-125	$3.40\ 10^{-10}$	$3.90\ 10^{-10}$	$2.80\ 10^{-10}$	$1.40\ 10^{-8}$	$3.10\ 10^{-8}$	$5.10\ 10^{-9}$	$2.50\ 10^{-10}$	$8.00\ 10^{-7}$	$3.50\ 10^{-12}$	$5.80\ 10^{-13}$
Sn-119m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
I-129	$9.70\ 10^{-8}$	$2.60\ 10^{-7}$	$1.10\ 10^{-7}$	$5.70\ 10^{-8}$	$4.10\ 10^{-8}$	$6.80\ 10^{-9}$	$5.00\ 10^{-7}$	$1.70\ 10^{-9}$	$1.80\ 10^{-11}$	$2.70\ 10^{-11}$
Ba-133	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-134	$7.10\ 10^{-10}$	$1.50\ 10^{-9}$	$1.50\ 10^{-8}$	$7.30\ 10^{-9}$	$2.00\ 10^{-8}$	$3.40\ 10^{-9}$	$3.20\ 10^{-8}$	$9.90\ 10^{-7}$	$8.70\ 10^{-13}$	$1.20\ 10^{-12}$
Cs-135	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-137	$2.00\ 10^{-9}$	$5.10\ 10^{-9}$	$2.50\ 10^{-8}$	$1.30\ 10^{-8}$	$2.70\ 10^{-8}$	$4.50\ 10^{-9}$	$4.20\ 10^{-8}$	$7.90\ 10^{-7}$	$1.40\ 10^{-12}$	$2.00\ 10^{-12}$
Ce-144	$3.40\ 10^{-10}$	$1.90\ 10^{-11}$	$1.60\ 10^{-11}$	$1.60\ 10^{-9}$	$2.30\ 10^{-11}$	$7.60\ 10^{-10}$	$5.90\ 10^{-11}$	$3.00\ 10^{-8}$	$7.40\ 10^{-12}$	$7.90\ 10^{-13}$
Pm-147	$3.80\ 10^{-11}$	$1.80\ 10^{-11}$	$1.30\ 10^{-11}$	$3.90\ 10^{-11}$	$2.80\ 10^{-11}$	$3.70\ 10^{-11}$	$6.50\ 10^{-12}$	$9.10\ 10^{-12}$	$2.20\ 10^{-12}$	$9.60\ 10^{-14}$
Sm-147	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Eu-152	$3.40\ 10^{-10}$	$4.60\ 10^{-10}$	$1.20\ 10^{-10}$	$3.60\ 10^{-10}$	$3.90\ 10^{-10}$	$5.10\ 10^{-10}$	$5.10\ 10^{-11}$	$2.50\ 10^{-6}$	$2.70\ 10^{-11}$	$7.70\ 10^{-13}$
Eu-154	$4.50\ 10^{-10}$	$5.00\ 10^{-10}$	$1.70\ 10^{-10}$	$5.10\ 10^{-10}$	$5.00\ 10^{-10}$	$6.70\ 10^{-10}$	$7.30\ 10^{-11}$	$2.50\ 10^{-6}$	$3.30\ 10^{-11}$	$1.10\ 10^{-12}$
Eu-155	$5.90\ 10^{-11}$	$4.60\ 10^{-11}$	$2.20\ 10^{-11}$	$6.70\ 10^{-11}$	$5.80\ 10^{-11}$	$7.70\ 10^{-11}$	$1.00\ 10^{-11}$	$4.60\ 10^{-8}$	$3.90\ 10^{-12}$	$1.50\ 10^{-13}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Gd-153	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Pb-210	$1.30\ 10^{-6}$	$2.80\ 10^{-6}$	$3.20\ 10^{-7}$	$3.10\ 10^{-7}$	$6.50\ 10^{-7}$	$2.20\ 10^{-7}$	$1.90\ 10^{-6}$	$5.40\ 10^{-9}$	$1.60\ 10^{-9}$	$1.10\ 10^{-9}$
Po-210	$7.20\ 10^{-8}$	$1.20\ 10^{-7}$	$2.30\ 10^{-7}$	$1.40\ 10^{-6}$	$8.00\ 10^{-8}$	$3.70\ 10^{-7}$	$2.70\ 10^{-8}$	$4.10\ 10^{-12}$	$5.70\ 10^{-10}$	$1.70\ 10^{-10}$
Ra-226	$5.30\ 10^{-7}$	$1.30\ 10^{-7}$	$8.60\ 10^{-8}$	$4.30\ 10^{-8}$	$1.20\ 10^{-7}$	$2.00\ 10^{-8}$	$8.00\ 10^{-7}$	$4.80\ 10^{-6}$	$3.30\ 10^{-9}$	$2.90\ 10^{-10}$
Ra-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-230	$4.40\ 10^{-8}$	$3.50\ 10^{-8}$	$1.90\ 10^{-9}$	$9.30\ 10^{-9}$	$3.20\ 10^{-9}$	$5.30\ 10^{-9}$	$1.90\ 10^{-9}$	$8.70\ 10^{-10}$	$1.90\ 10^{-8}$	$1.60\ 10^{-10}$
Th-232	$5.30\ 10^{-8}$	$4.20\ 10^{-8}$	$2.20\ 10^{-9}$	$1.10\ 10^{-8}$	$3.80\ 10^{-9}$	$6.40\ 10^{-9}$	$2.30\ 10^{-9}$	$1.20\ 10^{-5}$	$3.10\ 10^{-8}$	$1.90\ 10^{-10}$
Pa-231	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-232	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-234	$2.00\ 10^{-9}$	$2.40\ 10^{-9}$	$5.40\ 10^{-10}$	$2.70\ 10^{-10}$	$3.00\ 10^{-10}$	$5.00\ 10^{-11}$	$7.10\ 10^{-9}$	$3.10\ 10^{-11}$	$6.40\ 10^{-10}$	$5.30\ 10^{-12}$
U-235	$1.90\ 10^{-9}$	$2.30\ 10^{-9}$	$5.20\ 10^{-10}$	$2.60\ 10^{-10}$	$2.90\ 10^{-10}$	$4.80\ 10^{-11}$	$6.80\ 10^{-9}$	$6.30\ 10^{-8}$	$5.80\ 10^{-10}$	$5.10\ 10^{-12}$
U-236	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-238	$1.80\ 10^{-9}$	$2.20\ 10^{-9}$	$4.90\ 10^{-10}$	$2.50\ 10^{-10}$	$2.70\ 10^{-10}$	$4.60\ 10^{-11}$	$6.50\ 10^{-9}$	$1.30\ 10^{-8}$	$5.40\ 10^{-10}$	$4.90\ 10^{-12}$
Np-237	$2.10\ 10^{-8}$	$1.80\ 10^{-8}$	$1.20\ 10^{-9}$	$4.30\ 10^{-8}$	$5.00\ 10^{-9}$	$1.00\ 10^{-7}$	$7.20\ 10^{-10}$	$4.60\ 10^{-7}$	$1.50\ 10^{-8}$	$3.90\ 10^{-11}$
Pu-238	$1.70\ 10^{-8}$	$1.60\ 10^{-9}$	$1.50\ 10^{-9}$	$5.40\ 10^{-8}$	$2.60\ 10^{-9}$	$5.20\ 10^{-8}$	$1.50\ 10^{-9}$	$5.10\ 10^{-11}$	$2.90\ 10^{-8}$	$8.30\ 10^{-11}$
Pu-239	$2.00\ 10^{-8}$	$2.20\ 10^{-9}$	$1.80\ 10^{-9}$	$6.40\ 10^{-8}$	$3.10\ 10^{-9}$	$6.40\ 10^{-8}$	$1.80\ 10^{-9}$	$1.20\ 10^{-10}$	$3.20\ 10^{-8}$	$9.70\ 10^{-11}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. Inhalation	Inadv. ingestion
Pu-240	$2.00\ 10^{-8}$	$2.20\ 10^{-9}$	$1.80\ 10^{-9}$	$6.40\ 10^{-8}$	$3.10\ 10^{-9}$	$6.10\ 10^{-8}$	$1.80\ 10^{-9}$	$5.10\ 10^{-11}$	$3.20\ 10^{-8}$	$9.70\ 10^{-11}$
Pu-241	$3.50\ 10^{-10}$	$1.60\ 10^{-11}$	$2.70\ 10^{-11}$	$9.70\ 10^{-10}$	$3.90\ 10^{-11}$	$8.30\ 10^{-10}$	$2.20\ 10^{-11}$	$2.00\ 10^{-12}$	$4.60\ 10^{-10}$	$1.50\ 10^{-12}$
Pu-242	$1.90\ 10^{-8}$	$2.10\ 10^{-9}$	$1.70\ 10^{-9}$	$6.10\ 10^{-8}$	$3.00\ 10^{-9}$	$6.10\ 10^{-8}$	$1.70\ 10^{-9}$	$4.50\ 10^{-11}$	$3.00\ 10^{-8}$	$9.30\ 10^{-11}$
Pu-244	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-241	$3.00\ 10^{-8}$	$4.90\ 10^{-9}$	$2.70\ 10^{-9}$	$9.70\ 10^{-8}$	$5.60\ 10^{-9}$	$1.10\ 10^{-7}$	$3.30\ 10^{-9}$	$3.00\ 10^{-8}$	$4.80\ 10^{-8}$	$1.40\ 10^{-10}$
Am-242m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-243	$3.00\ 10^{-8}$	$5.10\ 10^{-9}$	$2.70\ 10^{-9}$	$1.00\ 10^{-7}$	$5.60\ 10^{-9}$	$1.20\ 10^{-7}$	$3.30\ 10^{-9}$	$6.60\ 10^{-7}$	$4.80\ 10^{-8}$	$1.40\ 10^{-10}$
Cm-242	$7.00\ 10^{-10}$	$7.20\ 10^{-13}$	$1.40\ 10^{-11}$	$5.10\ 10^{-10}$	$8.80\ 10^{-12}$	$1.80\ 10^{-10}$	$5.10\ 10^{-12}$	$1.30\ 10^{-11}$	$1.10\ 10^{-9}$	$1.90\ 10^{-12}$
Cm-243	$2.10\ 10^{-8}$	$8.10\ 10^{-10}$	$1.90\ 10^{-9}$	$6.90\ 10^{-8}$	$3.90\ 10^{-9}$	$8.00\ 10^{-8}$	$5.70\ 10^{-10}$	$4.00\ 10^{-7}$	$3.40\ 10^{-8}$	$9.40\ 10^{-11}$
Cm-244	$1.80\ 10^{-8}$	$5.50\ 10^{-10}$	$1.50\ 10^{-9}$	$5.20\ 10^{-8}$	$2.60\ 10^{-9}$	$5.20\ 10^{-8}$	$1.40\ 10^{-9}$	$6.30\ 10^{-11}$	$2.80\ 10^{-8}$	$7.80\ 10^{-11}$
Cm-245	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Table 86 Dose per unit release factors for infant member of the farming family – leachate release to treatment facility scenario (μSv y^{-1} per Bq y^{-1}) given in the EA IRAM methodology

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. inhalation	Inadv. ingestion
H-3	0	0	6.00 10^{-13}	5.50 10^{-13}	1.30 10^{-12}	3.60 10^{-13}	4.80 10^{-11}	0	1.40 10^{-18}	5.20 10^{-18}
C-14	9.40 10^{-9}	2.30 10^{-8}	1.10 10^{-9}	1.00 10^{-9}	2.30 10^{-9}	6.30 10^{-10}	3.50 10^{-8}	9.50 10^{-14}	2.10 10^{-14}	2.00 10^{-14}
Cl-36	1.00 10^{-7}	2.40 10^{-7}	4.30 10^{-8}	4.00 10^{-8}	1.20 10^{-7}	3.40 10^{-8}	7.80 10^{-7}	1.50 10^{-10}	6.00 10^{-13}	5.50 10^{-13}
Ca-41	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Mn-54	1.50 10^{-10}	3.10 10^{-10}	4.00 10^{-10}	1.60 10^{-8}	9.50 10^{-10}	1.20 10^{-8}	2.30 10^{-8}	3.20 10^{-7}	1.60 10^{-13}	3.00 10^{-13}
Fe-55	1.10 10^{-10}	7.20 10^{-12}	1.10 10^{-11}	3.00 10^{-10}	2.00 10^{-11}	2.20 10^{-8}	7.90 10^{-10}	0	1.40 10^{-13}	9.00 10^{-13}
Co-60	2.90 10^{-9}	4.90 10^{-9}	9.10 10^{-10}	8.30 10^{-8}	1.60 10^{-9}	4.40 10^{-8}	1.20 10^{-7}	4.90 10^{-6}	4.00 10^{-12}	1.20 10^{-11}
Ni-59	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ni-63	2.10 10^{-10}	5.40 10^{-10}	3.40 10^{-14}	3.10 10^{-13}	2.80 10^{-13}	7.60 10^{-13}	2.70 10^{-11}	0	2.10 10^{-13}	3.60 10^{-13}
Zn-65	3.50 10^{-9}	5.00 10^{-9}	1.90 10^{-9}	1.70 10^{-9}	7.20 10^{-9}	2.00 10^{-9}	2.70 10^{-6}	1.90 10^{-7}	1.30 10^{-13}	1.20 10^{-12}
Se-79	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sr-90	6.50 10^{-8}	3.30 10^{-8}	3.70 10^{-10}	3.40 10^{-10}	1.90 10^{-9}	5.20 10^{-10}	2.90 10^{-7}	1.20 10^{-9}	1.30 10^{-12}	3.20 10^{-12}
Mo-93	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Zr-93	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Nb-93m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Nb-94	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Tc-99	7.60 10^{-8}	1.90 10^{-7}	2.20 10^{-7}	6.20 10^{-7}	1.20 10^{-7}	1.30 10^{-7}	3.80 10^{-6}	6.70 10^{-12}	3.00 10^{-13}	4.20 10^{-13}
Ru-106	2.00 10^{-10}	1.30 10^{-10}	4.90 10^{-11}	4.50 10^{-11}	7.60 10^{-11}	2.10 10^{-11}	4.60 10^{-12}	1.90 10^{-8}	6.30 10^{-13}	1.10 10^{-12}

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. inhalation	Inadv. ingestion
Ag-108m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ag-110m	$1.40\ 10^{-9}$	$3.30\ 10^{-9}$	$9.90\ 10^{-10}$	$2.70\ 10^{-7}$	$2.60\ 10^{-9}$	$2.90\ 10^{-7}$	$2.90\ 10^{-6}$	$1.60\ 10^{-6}$	$1.00\ 10^{-12}$	$2.00\ 10^{-12}$
Cd-109	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sb-125	$4.20\ 10^{-10}$	$5.30\ 10^{-10}$	$2.50\ 10^{-10}$	$2.30\ 10^{-8}$	$3.00\ 10^{-8}$	$8.10\ 10^{-9}$	$9.70\ 10^{-10}$	$5.40\ 10^{-7}$	$1.40\ 10^{-12}$	$2.10\ 10^{-12}$
Sn-119m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
I-129	$4.80\ 10^{-8}$	$1.40\ 10^{-7}$	$4.00\ 10^{-8}$	$3.70\ 10^{-8}$	$1.60\ 10^{-8}$	$4.40\ 10^{-9}$	$7.70\ 10^{-7}$	$1.20\ 10^{-9}$	$4.00\ 10^{-12}$	$3.90\ 10^{-11}$
Ba-133	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-134	$3.50\ 10^{-10}$	$8.00\ 10^{-10}$	$5.00\ 10^{-9}$	$4.60\ 10^{-9}$	$7.70\ 10^{-9}$	$2.10\ 10^{-9}$	$4.90\ 10^{-8}$	$6.70\ 10^{-7}$	$2.10\ 10^{-13}$	$1.70\ 10^{-12}$
Cs-135	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cs-137	$1.00\ 10^{-9}$	$2.90\ 10^{-9}$	$9.00\ 10^{-9}$	$8.30\ 10^{-9}$	$1.10\ 10^{-8}$	$3.00\ 10^{-9}$	$6.70\ 10^{-8}$	$5.40\ 10^{-7}$	$3.40\ 10^{-13}$	$2.90\ 10^{-12}$
Ce-144	$5.20\ 10^{-10}$	$3.30\ 10^{-11}$	$1.70\ 10^{-11}$	$3.10\ 10^{-9}$	$2.70\ 10^{-11}$	$1.50\ 10^{-9}$	$2.80\ 10^{-10}$	$2.00\ 10^{-8}$	$3.70\ 10^{-12}$	$3.40\ 10^{-12}$
Pm-147	$5.40\ 10^{-11}$	$2.90\ 10^{-11}$	$1.30\ 10^{-11}$	$7.20\ 10^{-11}$	$3.10\ 10^{-11}$	$6.80\ 10^{-11}$	$2.90\ 10^{-11}$	$6.20\ 10^{-12}$	$9.80\ 10^{-13}$	$3.90\ 10^{-13}$
Sm-147	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Eu-152	$4.20\ 10^{-10}$	$6.20\ 10^{-10}$	$1.00\ 10^{-10}$	$5.70\ 10^{-10}$	$3.70\ 10^{-10}$	$8.00\ 10^{-10}$	$2.00\ 10^{-10}$	$1.70\ 10^{-6}$	$9.50\ 10^{-12}$	$2.70\ 10^{-12}$
Eu-154	$5.70\ 10^{-10}$	$6.90\ 10^{-10}$	$1.50\ 10^{-10}$	$8.20\ 10^{-10}$	$4.90\ 10^{-10}$	$1.10\ 10^{-9}$	$2.80\ 10^{-10}$	$1.70\ 10^{-6}$	$1.30\ 10^{-11}$	$4.00\ 10^{-12}$
Eu-155	$8.20\ 10^{-11}$	$7.00\ 10^{-11}$	$2.20\ 10^{-11}$	$1.20\ 10^{-10}$	$6.20\ 10^{-11}$	$1.40\ 10^{-10}$	$4.30\ 10^{-11}$	$3.10\ 10^{-8}$	$1.70\ 10^{-12}$	$6.10\ 10^{-13}$

Radionuclide$^{\$}$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. inhalation	Inadv. ingestion
Gd-153	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Pb-210	$1.10\ 10^{-6}$	$2.50\ 10^{-6}$	$1.80\ 10^{-7}$	$3.20\ 10^{-7}$	$4.10\ 10^{-7}$	$2.30\ 10^{-7}$	$4.90\ 10^{-6}$	$3.70\ 10^{-9}$	$6.80\ 10^{-10}$	$2.50\ 10^{-9}$
Po-210	$1.00\ 10^{-7}$	$1.90\ 10^{-7}$	$2.30\ 10^{-7}$	$2.50\ 10^{-6}$	$9.00\ 10^{-8}$	$6.80\ 10^{-7}$	$1.20\ 10^{-7}$	$2.80\ 10^{-12}$	$2.30\ 10^{-10}$	$7.10\ 10^{-10}$
Ra-226	$2.70\ 10^{-7}$	$7.30\ 10^{-8}$	$3.10\ 10^{-8}$	$2.80\ 10^{-8}$	$4.90\ 10^{-8}$	$1.30\ 10^{-8}$	$1.30\ 10^{-6}$	$3.30\ 10^{-6}$	$1.30\ 10^{-9}$	$4.20\ 10^{-10}$
Ra-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Th-230	$3.20\ 10^{-8}$	$2.80\ 10^{-8}$	$9.50\ 10^{-10}$	$8.70\ 10^{-9}$	$1.80\ 10^{-9}$	$5.00\ 10^{-9}$	$4.30\ 10^{-9}$	$5.90\ 10^{-10}$	$7.30\ 10^{-9}$	$3.20\ 10^{-10}$
Th-232	$3.60\ 10^{-8}$	$3.10\ 10^{-8}$	$1.00\ 10^{-9}$	$9.60\ 10^{-9}$	$2.00\ 10^{-9}$	$5.50\ 10^{-9}$	$4.70\ 10^{-9}$	$8.50\ 10^{-6}$	$1.00\ 10^{-8}$	$3.60\ 10^{-10}$
Pa-231	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-232	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-234	$1.50\ 10^{-9}$	$2.00\ 10^{-9}$	$2.80\ 10^{-10}$	$2.60\ 10^{-10}$	$1.70\ 10^{-10}$	$4.80\ 10^{-11}$	$1.70\ 10^{-8}$	$2.10\ 10^{-11}$	$2.50\ 10^{-10}$	$1.10\ 10^{-11}$
U-235	$1.50\ 10^{-9}$	$2.00\ 10^{-9}$	$2.80\ 10^{-10}$	$2.60\ 10^{-10}$	$1.70\ 10^{-10}$	$4.80\ 10^{-11}$	$1.70\ 10^{-8}$	$4.20\ 10^{-8}$	$2.30\ 10^{-10}$	$1.10\ 10^{-11}$
U-236	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
U-238	$1.40\ 10^{-9}$	$1.80\ 10^{-9}$	$2.60\ 10^{-10}$	$2.40\ 10^{-10}$	$1.60\ 10^{-10}$	$4.40\ 10^{-11}$	$1.50\ 10^{-8}$	$8.70\ 10^{-9}$	$2.20\ 10^{-10}$	$1.10\ 10^{-11}$
Np-237	$1.70\ 10^{-8}$	$1.60\ 10^{-8}$	$7.00\ 10^{-10}$	$4.60\ 10^{-8}$	$3.20\ 10^{-9}$	$1.10\ 10^{-7}$	$1.80\ 10^{-9}$	$3.10\ 10^{-7}$	$4.60\ 10^{-9}$	$9.20\ 10^{-11}$
Pu-238	$1.20\ 10^{-8}$	$1.30\ 10^{-9}$	$7.60\ 10^{-10}$	$5.00\ 10^{-8}$	$1.50\ 10^{-9}$	$4.80\ 10^{-8}$	$3.30\ 10^{-9}$	$3.50\ 10^{-11}$	$8.30\ 10^{-9}$	$1.70\ 10^{-10}$
Pu-239	$1.30\ 10^{-8}$	$1.60\ 10^{-9}$	$8.50\ 10^{-10}$	$5.50\ 10^{-8}$	$1.60\ 10^{-9}$	$5.50\ 10^{-8}$	$3.70\ 10^{-9}$	$8.10\ 10^{-11}$	$8.90\ 10^{-9}$	$1.80\ 10^{-10}$

Radionuclide$	Green vegetable	Root vegetable	Sheep meat	Sheep liver	Cow meat	Cow liver	Milk	External irradiation	Inadv. inhalation	Inadv. ingestion
Pu-240	$1.30\ 10^{-8}$	$1.60\ 10^{-9}$	$8.50\ 10^{-10}$	$5.50\ 10^{-8}$	$1.60\ 10^{-9}$	$5.20\ 10^{-8}$	$3.70\ 10^{-9}$	$3.50\ 10^{-11}$	$8.90\ 10^{-9}$	$1.80\ 10^{-10}$
Pu-241	$1.70\ 10^{-10}$	$8.40\ 10^{-12}$	$9.00\ 10^{-12}$	$5.90\ 10^{-10}$	$1.40\ 10^{-11}$	$5.10\ 10^{-10}$	$3.30\ 10^{-11}$	$1.40\ 10^{-12}$	$9.30\ 10^{-11}$	$2.10\ 10^{-12}$
Pu-242	$1.30\ 10^{-8}$	$1.50\ 10^{-9}$	$8.10\ 10^{-10}$	$5.20\ 10^{-8}$	$1.50\ 10^{-9}$	$5.20\ 10^{-8}$	$3.50\ 10^{-9}$	$3.00\ 10^{-11}$	$8.40\ 10^{-9}$	$1.80\ 10^{-10}$
Pu-244	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-241	$2.20\ 10^{-8}$	$3.90\ 10^{-9}$	$1.40\ 10^{-9}$	$9.00\ 10^{-8}$	$3.10\ 10^{-9}$	$1.00\ 10^{-7}$	$7.40\ 10^{-9}$	$2.00\ 10^{-8}$	$1.40\ 10^{-8}$	$2.90\ 10^{-10}$
Am-242m	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Am-243	$2.20\ 10^{-8}$	$4.10\ 10^{-9}$	$1.40\ 10^{-9}$	$9.30\ 10^{-8}$	$3.10\ 10^{-9}$	$1.10\ 10^{-7}$	$7.40\ 10^{-9}$	$4.50\ 10^{-7}$	$1.40\ 10^{-8}$	$2.90\ 10^{-10}$
Cm-242	$9.40\ 10^{-10}$	$1.10\ 10^{-12}$	$1.30\ 10^{-11}$	$8.90\ 10^{-10}$	$9.30\ 10^{-12}$	$3.10\ 10^{-10}$	$2.10\ 10^{-11}$	$8.50\ 10^{-12}$	$4.50\ 10^{-10}$	$7.20\ 10^{-12}$
Cm-243	$1.80\ 10^{-8}$	$8.00\ 10^{-10}$	$1.20\ 10^{-9}$	$7.80\ 10^{-8}$	$2.70\ 10^{-9}$	$9.10\ 10^{-8}$	$1.60\ 10^{-9}$	$2.70\ 10^{-7}$	$1.10\ 10^{-8}$	$2.40\ 10^{-10}$
Cm-244	$1.60\ 10^{-8}$	$5.40\ 10^{-10}$	$9.10\ 10^{-10}$	$6.00\ 10^{-8}$	$1.80\ 10^{-9}$	$6.00\ 10^{-8}$	$3.90\ 10^{-9}$	$4.30\ 10^{-11}$	$1.00\ 10^{-8}$	$2.00\ 10^{-10}$
Cm-245	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Anglers (discharge from treatment plant)

675. The assessment of doses to a coastal angler fishing in an estuary that receives discharges from the treatment plant is based on the EA Initial Radiological Assessment Methodology [(Environment Agency, 2006a) and (Environment Agency, 2006b)]. Members of the exposed group are assumed to be adults consuming fish and spending time on the banks of the estuary where water from the treatment works is discharged.

676. Habit data assumed for the angler family are summarised in Table 87. Adult consumption rates are taken from (Smith & Jones, 2003), as are the infant and child 97.5th percentile rates. The infant and child mean consumption rates were derived from the 97.5th rates assuming that the mean consumption rate is one third of the critical consumption rate, based on (Thorne, 2006). Beach occupancies are those assumed in (Environment Agency, 2006b).

Table 87 Habit data for the angling family: applicable during the Period of Authorisation

Pathway	Adult			Child			Infant		
	DPUR basis	mean	97.5th	DPUR basis	mean	97.5th	DPUR basis	mean	97.5th
Fish consumption (kg y^{-1})	100	61	100	20	6.67	20	5	1.67	5
Crustacean consumption (kg y^{-1})	20	18	20	5	1.67	5	0	0	0
Molluscs consumption (kg y^{-1})	20	14	15	5	1.67	5	0	0	0
Occupancy on beach (h y^{-1})	2000	2000		300	300		30	30	

677. The radiation dose incurred by an adult fisherman for each radionuclide ($Dose_{Rn,fisherman}$) is given by:

$$Dose_{Rn,fisherman} = F_{Rn} \cdot DF_{Rn,fisherman} \cdot Dil \cdot F_E \cdot F_p \cdot F_{exchange}$$

where:

- F_{Rn} is the flux of the radionuclide to the treatment works (Bq y^{-1});
- $DF_{Rn,\text{ fisherman}}$ is the dose per unit flux to the given exposed group (Sv y^{-1} per Bq y^{-1}) using default values – Total DPUR taken from EA methodology and given in Table 88, Table 89 and Table 90 for adult, child and infant members of the fishing family respectively;
- Dil is a dilution factor that is given by the ratio of the assumed and actual treatment throughputs, i.e. 60/3 10^5;

- F_E is the fraction from raw effluent that is disposed in liquid effluent;
- F_p is the consumption scaling factor; and,
- $F_{exchange}$ is the estuary exchange rate scaling factor, i.e. 100/51, to adjust for the assumed exchange rate in the Tees Estuary of 51 $m^3 s^{-1}$ (Dewar, et al., 2011).

Table 88 Dose per unit release factors for an adult member of the fishing family – leachate release to treatment facility scenario (μSv/y per Bq/y of discharge to sewer) given in the EA IRAM methodology

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
H-3	0.85	$2.90\ 10^{-16}$	$1.10\ 10^{-16}$	$1.10\ 10^{-16}$	0
C-14	0.85	$1.90\ 10^{-10}$	$7.30\ 10^{-11}$	$7.30\ 10^{-11}$	$1.60\ 10^{-16}$
Cl-36	0.9	$9.10\ 10^{-16}$	$3.60\ 10^{-16}$	$3.00\ 10^{-16}$	$3.10\ 10^{-17}$
Ca-41	nd	nd	nd	nd	nd
Mn-54	0.5	$2.20\ 10^{-13}$	$4.30\ 10^{-13}$	$4.30\ 10^{-12}$	$2.20\ 10^{-10}$
Fe-55	0.1	$2.10\ 10^{-14}$	$1.40\ 10^{-13}$	$1.40\ 10^{-13}$	0
Co-60	0.2	$4.60\ 10^{-12}$	$1.80\ 10^{-11}$	$5.20\ 10^{-11}$	$2.70\ 10^{-9}$
Ni-59	nd	nd	nd	nd	nd
Ni-63	0.5	$1.70\ 10^{-12}$	$6.60\ 10^{-13}$	$1.30\ 10^{-12}$	0
Zn-65	0.5	$2.20\ 10^{-11}$	$2.60\ 10^{-9}$	$6.90\ 10^{-10}$	$8.00\ 10^{-11}$
Se-79	nd	nd	nd	nd	nd
Sr-90	0.9	$1.40\ 10^{-12}$	$8.90\ 10^{-13}$	$1.80\ 10^{-12}$	$1.00\ 10^{-15}$
Mo-93	nd	nd	nd	nd	nd
Zr-93	nd	nd	nd	nd	nd
Nb-93m	nd	nd	nd	nd	nd
Nb-94	nd	nd	nd	nd	nd
Tc-99	0.9	$8.40\ 10^{-13}$	$4.10\ 10^{-12}$	$2.00\ 10^{-12}$	$1.00\ 10^{-16}$
Ru-106	0.9	$1.10\ 10^{-13}$	$2.20\ 10^{-12}$	$1.10\ 10^{-11}$	$3.50\ 10^{-11}$
Ag-108m	nd	nd	nd	nd	nd
Ag-110m	0.099	$3.50\ 10^{-10}$	$2.70\ 10^{-9}$	$8.20\ 10^{-10}$	$1.20\ 10^{-10}$
Cd-109	nd	nd	nd	nd	nd
Sb-125	0.2	$1.00\ 10^{-11}$	$2.00\ 10^{-12}$	$2.00\ 10^{-12}$	$1.50\ 10^{-11}$
Sn-119m	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd
I-129	0.8	$1.60\ 10^{-11}$	$2.10\ 10^{-12}$	$7.00\ 10^{-12}$	$5.40\ 10^{-15}$

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
Ba-133	nd	nd	nd	nd	nd
Cs-134	0.7	$2.80\ 10^{-11}$	$5.40\ 10^{-12}$	$6.50\ 10^{-12}$	$8.40\ 10^{-11}$
Cs-135	nd	nd	nd	nd	nd
Cs-137	0.7	$1.90\ 10^{-11}$	$3.80\ 10^{-12}$	$4.50\ 10^{-12}$	$1.20\ 10^{-10}$
Ce-144	0.5	$5.30\ 10^{-14}$	$4.20\ 10^{-13}$	$8.40\ 10^{-13}$	$1.40\ 10^{-11}$
Pm-147	0.5	$2.40\ 10^{-14}$	$1.30\ 10^{-13}$	$2.30\ 10^{-13}$	$6.00\ 10^{-15}$
Sm-147	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd
Eu-152	0.5	$1.40\ 10^{-13}$	$7.40\ 10^{-13}$	$1.30\ 10^{-12}$	$2.20\ 10^{-9}$
Eu-154	0.5	$2.00\ 10^{-13}$	$1.00\ 10^{-12}$	$1.80\ 10^{-12}$	$2.00\ 10^{-9}$
Eu-155	0.5	$3.10\ 10^{-14}$	$1.60\ 10^{-13}$	$2.90\ 10^{-13}$	$3.70\ 10^{-11}$
Gd-153	nd	nd	nd	nd	nd
Pb-210	0.1	$6.70\ 10^{-10}$	$1.20\ 10^{-7}$	$6.60\ 10^{-8}$	$2.50\ 10^{-12}$
Po-210	0.099	$6.00\ 10^{-11}$	$2.40\ 10^{-10}$	$2.40\ 10^{-10}$	$9.10\ 10^{-17}$
Ra-226	0.5	$4.40\ 10^{-10}$	$1.70\ 10^{-10}$	$1.70\ 10^{-10}$	$2.60\ 10^{-10}$
Ra-228	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd
Th-230	0.1	$3.20\ 10^{-11}$	$2.10\ 10^{-11}$	$2.10\ 10^{-11}$	$3.00\ 10^{-11}$
Th-232	0.1	$7.20\ 10^{-10}$	$4.60\ 10^{-10}$	$4.60\ 10^{-10}$	$5.10\ 10^{-9}$
Pa-231	nd	nd	nd	nd	nd
U-232	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd
U-234	0.9	$7.80\ 10^{-13}$	$3.10\ 10^{-12}$	$9.20\ 10^{-12}$	$4.90\ 10^{-15}$
U-235	0.9	$7.50\ 10^{-13}$	$2.90\ 10^{-12}$	$8.80\ 10^{-12}$	$9.60\ 10^{-12}$
U-236	nd	nd	nd	nd	nd
U-238	0.9	$7.20\ 10^{-13}$	$2.80\ 10^{-12}$	$8.50\ 10^{-12}$	$1.80\ 10^{-12}$
Np-237	0.5	$1.80\ 10^{-12}$	$6.90\ 10^{-11}$	$2.80\ 10^{-10}$	$1.40\ 10^{-11}$
Pu-238	0.5	$1.10\ 10^{-10}$	$9.00\ 10^{-11}$	$1.30\ 10^{-9}$	$5.00\ 10^{-14}$
Pu-239	0.5	$1.20\ 10^{-10}$	$9.90\ 10^{-11}$	$1.50\ 10^{-9}$	$1.20\ 10^{-13}$
Pu-240	0.5	$1.20\ 10^{-10}$	$9.90\ 10^{-11}$	$1.50\ 10^{-9}$	$5.30\ 10^{-14}$
Pu-241	0.5	$2.30\ 10^{-12}$	$1.80\ 10^{-12}$	$2.70\ 10^{-11}$	$2.40\ 10^{-13}$
Pu-242	0.5	$1.20\ 10^{-10}$	$9.50\ 10^{-11}$	$1.40\ 10^{-9}$	$4.70\ 10^{-14}$
Pu-244	nd	nd	nd	nd	nd
Am-241	0.1	$7.00\ 10^{-12}$	$1.10\ 10^{-11}$	$2.80\ 10^{-11}$	$2.50\ 10^{-11}$

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
Am-242m	nd	nd	nd	nd	nd
Am-243	0.1	7.10 10^{-12}	1.10 10^{-11}	2.80 10^{-11}	5.30 10^{-10}
Cm-242	0.099	4.40 10^{-13}	7.00 10^{-13}	1.80 10^{-12}	3.90 10^{-15}
Cm-243	0.1	5.10 10^{-12}	8.20 10^{-12}	2.00 10^{-11}	2.60 10^{-10}
Cm-244	0.1	4.00 10^{-12}	6.40 10^{-12}	1.60 10^{-11}	4.00 10^{-14}
Cm-245	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Table 89 Dose per unit release factors for a child member of the fishing family – leachate release to treatment facility scenario (μSv/y per Bq/y of discharge to sewer) given in the EA IRAM methodology

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
H-3	0.85	7.50 10^{-17}	3.70 10^{-17}	3.70 10^{-17}	0
C-14	0.85	5.10 10^{-11}	2.50 10^{-11}	2.50 10^{-11}	2.30 10^{-17}
Cl-36	0.9	3.70 10^{-16}	1.80 10^{-16}	1.50 10^{-16}	4.60 10^{-18}
Ca-41	nd	nd	nd	nd	nd
Mn-54	0.5	7.90 10^{-14}	2.00 10^{-13}	2.00 10^{-12}	3.40 10^{-11}
Fe-55	0.1	1.40 10^{-14}	1.20 10^{-13}	1.20 10^{-13}	0
Co-60	0.2	3.00 10^{-12}	1.50 10^{-11}	4.20 10^{-11}	4.10 10^{-10}
Ni-59	nd	nd	nd	nd	nd
Ni-63	0.5	6.20 10^{-13}	3.10 10^{-13}	6.10 10^{-13}	0
Zn-65	0.5	7.10 10^{-12}	1.10 10^{-9}	2.80 10^{-10}	1.20 10^{-11}
Se-79	nd	nd	nd	nd	nd
Sr-90	0.9	5.90 10^{-13}	4.80 10^{-13}	9.60 10^{-13}	1.50 10^{-16}
Mo-93	nd	nd	nd	nd	nd
Zr-93	nd	nd	nd	nd	nd
Nb-93m	nd	nd	nd	nd	nd
Nb-94	nd	nd	nd	nd	nd
Tc-99	0.9	3.40 10^{-13}	2.10 10^{-12}	1.00 10^{-12}	1.50 10^{-17}
Ru-106	0.9	4.70 10^{-14}	1.20 10^{-12}	5.80 10^{-12}	5.20 10^{-12}
Ag-108m	nd	nd	nd	nd	nd
Ag-110m	0.099	1.30 10^{-10}	1.30 10^{-9}	3.80 10^{-10}	1.80 10^{-11}

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
Cd-109	nd	nd	nd	nd	nd
Sb-125	0.2	$4.00\ 10^{-12}$	$9.80\ 10^{-13}$	$9.80\ 10^{-13}$	$2.20\ 10^{-12}$
Sn-119m	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd
I-129	0.8	$5.60\ 10^{-12}$	$9.10\ 10^{-13}$	$3.00\ 10^{-12}$	$8.10\ 10^{-16}$
Ba-133	nd	nd	nd	nd	nd
Cs-134	0.7	$4.10\ 10^{-12}$	$1.00\ 10^{-12}$	$1.20\ 10^{-12}$	$1.30\ 10^{-11}$
Cs-135	nd	nd	nd	nd	nd
Cs-137	0.7	$3.00\ 10^{-12}$	$7.30\ 10^{-13}$	$8.70\ 10^{-13}$	$1.90\ 10^{-11}$
Ce-144	0.5	$2.20\ 10^{-14}$	$2.20\ 10^{-13}$	$4.50\ 10^{-13}$	$2.10\ 10^{-12}$
Pm-147	0.5	$1.10\ 10^{-14}$	$7.10\ 10^{-14}$	$1.20\ 10^{-13}$	$9.10\ 10^{-16}$
Sm-147	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd
Eu-152	0.5	$5.20\ 10^{-14}$	$3.40\ 10^{-13}$	$6.00\ 10^{-13}$	$3.30\ 10^{-10}$
Eu-154	0.5	$8.00\ 10^{-14}$	$5.30\ 10^{-13}$	$9.30\ 10^{-13}$	$3.00\ 10^{-10}$
Eu-155	0.5	$1.30\ 10^{-14}$	$8.70\ 10^{-14}$	$1.50\ 10^{-13}$	$5.50\ 10^{-12}$
Gd-153	nd	nd	nd	nd	nd
Pb-210	0.1	$3.70\ 10^{-10}$	$8.20\ 10^{-8}$	$4.60\ 10^{-8}$	$3.70\ 10^{-13}$
Po-210	0.099	$2.60\ 10^{-11}$	$1.30\ 10^{-10}$	$1.30\ 10^{-10}$	$1.40\ 10^{-17}$
Ra-226	0.5	$2.50\ 10^{-10}$	$1.20\ 10^{-10}$	$1.20\ 10^{-10}$	$3.80\ 10^{-11}$
Ra-228	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd
Th-230	0.1	$8.10\ 10^{-12}$	$6.70\ 10^{-12}$	$6.70\ 10^{-12}$	$4.60\ 10^{-12}$
Th-232	0.1	$7.80\ 10^{-10}$	$6.30\ 10^{-10}$	$6.30\ 10^{-10}$	$7.60\ 10^{-10}$
Pa-231	nd	nd	nd	nd	nd
U-232	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd
U-234	0.9	$2.40\ 10^{-13}$	$1.20\ 10^{-12}$	$3.50\ 10^{-12}$	$7.30\ 10^{-16}$
U-235	0.9	$2.30\ 10^{-13}$	$1.10\ 10^{-12}$	$3.30\ 10^{-12}$	$1.40\ 10^{-12}$
U-236	nd	nd	nd	nd	nd
U-238	0.9	$2.20\ 10^{-13}$	$1.10\ 10^{-12}$	$3.20\ 10^{-12}$	$2.70\ 10^{-13}$
Np-237	0.5	$3.50\ 10^{-13}$	$1.70\ 10^{-11}$	$6.90\ 10^{-11}$	$2.00\ 10^{-12}$

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
Pu-238	0.5	$2.40\ 10^{-11}$	$2.30\ 10^{-11}$	$3.50\ 10^{-10}$	$7.50\ 10^{-15}$
Pu-239	0.5	$2.70\ 10^{-11}$	$2.70\ 10^{-11}$	$4.00\ 10^{-10}$	$1.80\ 10^{-14}$
Pu-240	0.5	$2.70\ 10^{-11}$	$2.70\ 10^{-11}$	$4.00\ 10^{-10}$	$8.00\ 10^{-15}$
Pu-241	0.5	$4.90\ 10^{-13}$	$4.90\ 10^{-13}$	$7.30\ 10^{-12}$	$3.50\ 10^{-14}$
Pu-242	0.5	$2.60\ 10^{-11}$	$2.60\ 10^{-11}$	$3.90\ 10^{-10}$	$7.00\ 10^{-15}$
Pu-244	nd	nd	nd	nd	nd
Am-241	0.1	$1.50\ 10^{-12}$	$3.10\ 10^{-12}$	$7.70\ 10^{-12}$	$3.70\ 10^{-12}$
Am-242m	nd	nd	nd	nd	nd
Am-243	0.1	$1.60\ 10^{-12}$	$3.10\ 10^{-12}$	$7.70\ 10^{-12}$	$8.00\ 10^{-11}$
Cm-242	0.099	$1.60\ 10^{-13}$	$3.20\ 10^{-13}$	$8.00\ 10^{-13}$	$5.90\ 10^{-16}$
Cm-243	0.1	$1.10\ 10^{-12}$	$2.20\ 10^{-12}$	$5.40\ 10^{-12}$	$3.90\ 10^{-11}$
Cm-244	0.1	$9.40\ 10^{-13}$	$1.90\ 10^{-12}$	$4.70\ 10^{-12}$	$5.90\ 10^{-15}$
Cm-245	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Table 90 Dose per unit release factors for an infant member of the fishing family – leachate release to treatment facility scenario (µSv/y per Bq/y of discharge to sewer) given in the EA IRAM methodology

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
H-3	0.85	$3.90\ 10^{-17}$	0	0	0
C-14	0.85	$2.60\ 10^{-11}$	0	0	$2.30\ 10^{-18}$
Cl-36	0.9	$3.10\ 10^{-16}$	0	0	$4.60\ 10^{-19}$
Ca-41	nd	nd	nd	nd	nd
Mn-54	0.5	$4.70\ 10^{-14}$	0	0	$3.40\ 10^{-12}$
Fe-55	0.1	$7.70\ 10^{-15}$	0	0	0
Co-60	0.2	$1.80\ 10^{-12}$	0	0	$4.10\ 10^{-11}$
Ni-59	nd	nd	nd	nd	nd
Ni-63	0.5	$4.60\ 10^{-13}$	0	0	0
Zn-65	0.5	$4.50\ 10^{-12}$	0	0	$1.20\ 10^{-12}$
Se-79	nd	nd	nd	nd	nd
Sr-90	0.9	$1.80\ 10^{-13}$	0	0	$1.50\ 10^{-17}$
Mo-93	nd	nd	nd	nd	nd

Radionuclide$	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
Zr-93	nd	nd	nd	nd	nd
Nb-93m	nd	nd	nd	nd	nd
Nb-94	nd	nd	nd	nd	nd
Tc-99	0.9	$3.10\ 10^{-13}$	0	0	$1.50\ 10^{-18}$
Ru-106	0.9	$3.80\ 10^{-14}$	0	0	$5.20\ 10^{-13}$
Ag-108m	nd	nd	nd	nd	nd
Ag-110m	0.099	$8.70\ 10^{-11}$	0	0	$1.80\ 10^{-12}$
Cd-109	nd	nd	nd	nd	nd
Sb-125	0.2	$2.90\ 10^{-12}$	0	0	$2.20\ 10^{-13}$
Sn-119m	nd	nd	nd	nd	nd
Sn-123	nd	nd	nd	nd	nd
Sn-126	nd	nd	nd	nd	nd
Te-127m	nd	nd	nd	nd	nd
I-129	0.8	$1.60\ 10^{-12}$	0	0	$8.10\ 10^{-17}$
Ba-133	nd	nd	nd	nd	nd
Cs-134	0.7	$1.20\ 10^{-12}$	0	0	$1.30\ 10^{-12}$
Cs-135	nd	nd	nd	nd	nd
Cs-137	0.7	$8.90\ 10^{-13}$	0	0	$1.90\ 10^{-12}$
Ce-144	0.5	$2.00\ 10^{-14}$	0	0	$2.10\ 10^{-13}$
Pm-147	0.5	$8.90\ 10^{-15}$	0	0	$9.10\ 10^{-17}$
Sm-147	nd	nd	nd	nd	nd
Sm-151	nd	nd	nd	nd	nd
Eu-152	0.5	$3.70\ 10^{-14}$	0	0	$3.30\ 10^{-11}$
Eu-154	0.5	$5.90\ 10^{-14}$	0	0	$3.00\ 10^{-11}$
Eu-155	0.5	$1.10\ 10^{-14}$	0	0	$5.50\ 10^{-13}$
Gd-153	nd	nd	nd	nd	nd
Pb-210	0.1	$1.70\ 10^{-10}$	0	0	$3.70\ 10^{-14}$
Po-210	0.099	$2.20\ 10^{-11}$	0	0	$1.40\ 10^{-18}$
Ra-226	0.5	$7.50\ 10^{-11}$	0	0	$3.80\ 10^{-12}$
Ra-228	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd
Th-228	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd
Th-230	0.1	$3.30\ 10^{-12}$	0	0	$4.60\ 10^{-13}$
Th-232	0.1	$2.90\ 10^{-10}$	0	0	$7.60\ 10^{-11}$
Pa-231	nd	nd	nd	nd	nd

Radionuclide[$]	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation
U-232	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd
U-234	0.9	$1.00\ 10^{-13}$	0	0	$7.30\ 10^{-17}$
U-235	0.9	$1.00\ 10^{-13}$	0	0	$1.40\ 10^{-13}$
U-236	nd	nd	nd	nd	nd
U-238	0.9	$9.60\ 10^{-14}$	0	0	$2.70\ 10^{-14}$
Np-237	0.5	$1.70\ 10^{-13}$	0	0	$2.00\ 10^{-13}$
Pu-238	0.5	$9.80\ 10^{-12}$	0	0	$7.50\ 10^{-16}$
Pu-239	0.5	$1.00\ 10^{-11}$	0	0	$1.80\ 10^{-15}$
Pu-240	0.5	$1.00\ 10^{-11}$	0	0	$8.00\ 10^{-16}$
Pu-241	0.5	$1.40\ 10^{-13}$	0	0	$3.50\ 10^{-15}$
Pu-242	0.5	$9.90\ 10^{-12}$	0	0	$7.00\ 10^{-16}$
Pu-244	nd	nd	nd	nd	nd
Am-241	0.1	$6.50\ 10^{-13}$	0	0	$3.70\ 10^{-13}$
Am-242m	nd	nd	nd	nd	nd
Am-243	0.1	$6.50\ 10^{-13}$	0	0	$8.00\ 10^{-12}$
Cm-242	0.099	$1.20\ 10^{-13}$	0	0	$5.90\ 10^{-17}$
Cm-243	0.1	$5.60\ 10^{-13}$	0	0	$3.90\ 10^{-12}$
Cm-244	0.1	$4.90\ 10^{-13}$	0	0	$5.90\ 10^{-16}$
Cm-245	nd	nd	nd	nd	nd
Cm-246	nd	nd	nd	nd	nd
Cm-248	nd	nd	nd	nd	nd

Note: $ Radionuclides that are not considered in IRAM are shown with no data (nd).

Reed bed facility worker

678. An assessment has been made of the radiological impact arising from treatment of contaminated leachate using a Reed Bed facility. The Billingham Reed Beds is used for leachate treatment (Scott Bros. Ltd) and under current normal operating circumstances approximately 2,600 $m^3\ y^{-1}$ of leachate is sent for off-site treatment. The treated leachate is then discharged to the estuary via Billingham Beck.

679. Output from a GoldSim groundwater model of the site provides an estimate of the maximum leachate activity concentration and this is used to assess the potential doses arising from leachate treatment. The calculations are conservative because they do not take into account sorption of radionuclides within waste materials whereas in reality the waste received at Port Clarence is likely to provide sorption sites within waste cells.

680. The flux of radionuclides to off-site treatment (Bq y^{-1}) uses the peak leachate activity concentrations (per MBq input to the landfill) during the active control period (60 years after capping) and the leachate export rate (2,647 m^3 y^{-1}) from the site. The ingrowth of daughters is modelled using GoldSim and the activity concentrations of the daughters are propagated through the model and the dose contributions summed.

681. The radiological assessment considers worker exposure to contamination of the Reed Beds (49,000 m^2 to a depth of 0.6 m) and accumulation over 7 years which is the anticipated operating life of the beds.

682. The reed bed operative receives a dose from external irradiation (SNIFFER, 2006):

$$Dose_{reed\ bed\ operator} = \left(\frac{D_{irr,slab}^{Rn}}{8766}\right) T C_{Rn,\text{reed bed}}$$

where:

- T is the time that the excavator is exposed to the material (104 h y^{-1});
- $D_{irr,slab}$, is the dose coefficient for radionuclide *Rn* (Sv y^{-1} Bq^{-1} kg^{-1});
- 8766 is the number of hours in a year (h y^{-1});
- $C_{Rn,waste}$ is the activity concentration of radionuclide *Rn* (Bq kg^{-1}) integrated over a period of 7 years, *t*:

$$C_{Rn,\text{reed bed}} = \frac{A_{Rn}(t)}{V_{reed\ bed}\rho_{reed\ bed}}$$

- $V_{reed\ bed}$ is the volume of the reed bed in which the activity is assumed to be concentrated (29,400 m^3); and,
- $\rho_{reed\ bed}$ is the density of the reed bed substrate (1680 kg m^{-3}).

E.3.7.3. Doses from leachate treatment

Dose per MBq Deposited at Port Clarence – Leachate Treatment

683. The calculated doses shown below for each of the assessed groups are per MBq input to Port Clarence landfills.

Table 91 Dose for exposure from the off-site treatment of leachate per MBq input to Port Clarence

Radionuclide	Leachate treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Farming family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group on farm	Fishing family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group fishing
H-3	$2.60\ 10^{-10}$	0	$1.67\ 10^{-12}$	Infant	$3.13\ 10^{-16}$	Adult
C-14	$2.65\ 10^{-11}$	$2.92\ 10^{-13}$	$6.91\ 10^{-12}$	Infant	$6.77\ 10^{-13}$	Adult
Cl-36	$3.29\ 10^{-8}$	$1.02\ 10^{-8}$	$1.61\ 10^{-8}$	Infant	$6.76\ 10^{-16}$	Adult
Ca-41	$4.25\ 10^{-10}$	0	$1.40\ 10^{-12}$	Infant	$5.82\ 10^{-18}$	Adult
Mn-54	$4.82\ 10^{-8}$	$9.23\ 10^{-11}$	$9.03\ 10^{-10}$	Adult	$2.57\ 10^{-13}$	Adult
Fe-55	$6.19\ 10^{-12}$	0	$1.25\ 10^{-13}$	Infant	$1.47\ 10^{-15}$	Adult
Co-60	$1.12\ 10^{-6}$	$8.22\ 10^{-9}$	$5.11\ 10^{-11}$	Infant	$1.52\ 10^{-14}$	Adult
Ni-59	$2.97\ 10^{-12}$	0	nd	nd	nd	nd
Ni-63	$7.26\ 10^{-12}$	0	$7.93\ 10^{-8}$	Infant	$3.17\ 10^{-12}$	Adult
Zn-65	$3.60\ 10^{-8}$	$5.43\ 10^{-11}$	$4.80\ 10^{-13}$	Adult	$2.06\ 10^{-14}$	Adult
Se-79	$1.75\ 10^{-10}$	$2.03\ 10^{-13}$	nd	nd	nd	nd
Sr-90	$4.30\ 10^{-9}$	$5.32\ 10^{-10}$	$7.38\ 10^{-10}$	Adult	$1.82\ 10^{-14}$	Adult
Mo-93	$2.78\ 10^{-10}$	$3.19\ 10^{-11}$	nd	nd	nd	nd
Zr-93	$8.36\ 10^{-11}$	0	$2.02\ 10^{-9}$	Infant	$6.62\ 10^{-13}$	Adult
Nb-93m	$2.78\ 10^{-12}$	$5.21\ 10^{-14}$	nd	nd	nd	nd
Nb-94	$1.56\ 10^{-7}$	$1.41\ 10^{-8}$	$1.66\ 10^{-11}$	Adult	$1.31\ 10^{-14}$	Adult
Tc-99	$6.94\ 10^{-9}$	$6.22\ 10^{-10}$	$2.02\ 10^{-11}$	Adult	$2.12\ 10^{-14}$	Adult
Ru-106	$1.49\ 10^{-8}$	$1.37\ 10^{-10}$	$2.84\ 10^{-14}$	Infant	$7.06\ 10^{-17}$	Adult

Radionuclide	Leachate treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Farming family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group on farm	Fishing family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group fishing
Ag-108m	$1.07\ 10^{-6}$	$5.27\ 10^{-8}$	$4.15\ 10^{-10}$	Adult	$1.11\ 10^{-12}$	Adult
Ag-110m	$7.48\ 10^{-7}$	$6.58\ 10^{-10}$	$4.12\ 10^{-10}$	Adult	$9.78\ 10^{-13}$	Adult
Cd-109	$3.63\ 10^{-10}$	$3.90\ 10^{-12}$	$7.13\ 10^{-12}$	Adult	$1.72\ 10^{-14}$	Adult
Sb-125	$1.16\ 10^{-6}$	$4.48\ 10^{-9}$	$3.76\ 10^{-10}$	Infant	$2.14\ 10^{-12}$	Adult
Sn-119m	$2.14\ 10^{-11}$	$3.63\ 10^{-14}$	$9.46\ 10^{-11}$	Adult	$1.88\ 10^{-12}$	Adult
Sn-123	$1.48\ 10^{-10}$	$1.19\ 10^{-13}$	nd	nd	nd	nd
Sn-126	$1.79\ 10^{-7}$	$1.62\ 10^{-8}$	nd	nd	nd	nd
Te-127m	$4.92\ 10^{-11}$	$2.78\ 10^{-14}$	nd	nd	nd	nd
I-129	$9.16\ 10^{-8}$	$3.98\ 10^{-9}$	$1.50\ 10^{-11}$	Infant	$3.34\ 10^{-14}$	Adult
Ba-133	$1.79\ 10^{-5}$	$1.65\ 10^{-6}$	nd	nd	nd	nd
Cs-134	$8.51\ 10^{-8}$	$6.21\ 10^{-10}$	$5.30\ 10^{-10}$	Adult	$8.83\ 10^{-14}$	Adult
Cs-135	$1.31\ 10^{-11}$	$6.98\ 10^{-14}$	nd	nd	nd	nd
Cs-137	$4.17\ 10^{-8}$	$3.29\ 10^{-9}$	nd	nd	nd	nd
Ce-144	$6.52\ 10^{-10}$	$1.10\ 10^{-12}$	$9.66\ 10^{-13}$	Infant	$1.20\ 10^{-14}$	Adult
Pm-147	$9.37\ 10^{-12}$	$1.19\ 10^{-14}$	$3.07\ 10^{-12}$	Adult	$2.23\ 10^{-14}$	Adult
Sm-147	$3.39\ 10^{-9}$	0	nd	nd	nd	nd
Sm-151	$2.36\ 10^{-12}$	$1.84\ 10^{-15}$	$1.01\ 10^{-9}$	Adult	$1.06\ 10^{-12}$	Adult
Eu-152	$6.68\ 10^{-7}$	$1.88\ 10^{-8}$	$2.05\ 10^{-11}$	Adult	$1.99\ 10^{-13}$	Adult
Eu-154	$7.10\ 10^{-7}$	$1.32\ 10^{-8}$	$1.18\ 10^{-12}$	Infant	$1.30\ 10^{-14}$	Adult
Eu-155	$1.58\ 10^{-8}$	$1.65\ 10^{-10}$	$2.56\ 10^{-11}$	Adult	$2.29\ 10^{-13}$	Adult

Radionuclide	Leachate treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Farming family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group on farm	Fishing family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group fishing
Gd-153	$4.59\ 10^{-9}$	$6.82\ 10^{-12}$	$2.26\ 10^{-11}$	Adult	$2.29\ 10^{-13}$	Adult
Pb-210	$2.09\ 10^{-8}$	$2.92\ 10^{-12}$	$8.25\ 10^{-13}$	Adult	$4.01\ 10^{-15}$	Adult
Po-210	$2.59\ 10^{-8}$	$1.10\ 10^{-15}$	$1.91\ 10^{-11}$	Adult	$6.18\ 10^{-16}$	Adult
Ra-226	$1.20\ 10^{-7}$	$9.66\ 10^{-9}$	$2.19\ 10^{-11}$	Adult	$2.16\ 10^{-13}$	Adult
Ra-228	$3.58\ 10^{-8}$	$1.74\ 10^{-9}$	$6.72\ 10^{-12}$	Adult	$9.56\ 10^{-16}$	Adult
Ac-227	$2.10\ 10^{-7}$	$9.13\ 10^{-10}$	$2.48\ 10^{-12}$	Adult	$5.85\ 10^{-16}$	Adult
Th-228	$1.67\ 10^{-7}$	$3.81\ 10^{-10}$	nd	nd	nd	nd
Th-229	$1.07\ 10^{-7}$	$1.82\ 10^{-9}$	$3.65\ 10^{-13}$	Adult	$1.13\ 10^{-14}$	Adult
Th-230	$2.77\ 10^{-8}$	$1.39\ 10^{-12}$	nd	nd	nd	nd
Th-232	$1.89\ 10^{-7}$	$6.95\ 10^{-9}$	$5.35\ 10^{-11}$	Infant	$1.71\ 10^{-13}$	Adult
Pa-231	$4.51\ 10^{-8}$	$2.08\ 10^{-10}$	nd	nd	nd	nd
U-232	$2.90\ 10^{-8}$	$7.33\ 10^{-12}$	nd	nd	nd	nd
U-233	$3.47\ 10^{-9}$	$1.53\ 10^{-11}$	nd	nd	nd	nd
U-234	$3.35\ 10^{-9}$	$4.39\ 10^{-12}$	nd	nd	nd	nd
U-235	$2.44\ 10^{-8}$	$7.88\ 10^{-9}$	$2.06\ 10^{-11}$	Infant	$9.79\ 10^{-14}$	Adult
U-236	$3.12\ 10^{-9}$	$2.35\ 10^{-12}$	nd	nd	nd	nd
U-238	$2.93\ 10^{-9}$	$1.13\ 10^{-12}$	nd	nd	nd	nd
Np-237	$4.74\ 10^{-7}$	$4.84\ 10^{-9}$	nd	nd	nd	nd
Pu-238	$4.35\ 10^{-8}$	$3.53\ 10^{-13}$	nd	nd	nd	nd
Pu-239	$4.78\ 10^{-8}$	$8.71\ 10^{-13}$	nd	nd	nd	nd

Radionuclide	Leachate treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Farming family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group on farm	Fishing family ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Limiting age group fishing
Pu-240	$4.78\ 10^{-8}$	$4.32\ 10^{-13}$	$2.07\ 10^{-14}$	Infant	$7.15\ 10^{-16}$	Adult
Pu-241	$8.72\ 10^{-10}$	$5.42\ 10^{-15}$	nd	nd	nd	nd
Pu-242	$4.40\ 10^{-8}$	$3.78\ 10^{-13}$	nd	nd	nd	nd
Pu-244	$4.40\ 10^{-8}$	$2.23\ 10^{-13}$	nd	nd	nd	nd
Am-241	$2.01\ 10^{-8}$	$3.50\ 10^{-11}$	$4.78\ 10^{-11}$	Infant	$1.21\ 10^{-14}$	Adult
Am-242m	$2.33\ 10^{-8}$	$1.22\ 10^{-12}$	nd	nd	nd	nd
Am-243	$2.17\ 10^{-8}$	$1.19\ 10^{-10}$	nd	nd	nd	nd
Cm-242	$8.48\ 10^{-11}$	$1.16\ 10^{-16}$	nd	nd	nd	nd
Cm-243	$6.43\ 10^{-9}$	$7.12\ 10^{-11}$	$5.37\ 10^{-12}$	Adult	$5.90\ 10^{-15}$	Adult
Cm-244	$3.10\ 10^{-9}$	$1.12\ 10^{-14}$	$5.97\ 10^{-16}$	Infant	$9.19\ 10^{-16}$	Adult
Cm-245	$7.15\ 10^{-9}$	$7.97\ 10^{-11}$	nd	nd	nd	nd
Cm-246	$5.55\ 10^{-9}$	$2.72\ 10^{-14}$	nd	nd	nd	nd
Cm-248	$2.04\ 10^{-8}$	$2.06\ 10^{-14}$	nd	nd	nd	nd

684. The doses calculated using illustrative inventories are considered further in Appendix D.

Dose from radiological capacity – Leachate Treatment

685. Off-site treatment of leachate is not used to determine radiological capacity of the landfills. The dose from disposal at the radiological capacity is given in Table 92.

Table 92 Dose for exposure from the off-site treatment of leachate for disposal of radiological capacity at Port Clarence

Radionuclide	Radiological capacity (MBq)	Leachate treatment worker ($\mu Sv\ y^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}$)	Farming family ($\mu Sv\ y^{-1}$)	Fishing family ($\mu Sv\ y^{-1}$)
H-3	$6.43\ 10^{9}$	$1.67\ 10^{0}$	0	$1.07\ 10^{-2}$	$2.01\ 10^{-6}$
C-14	$1.87\ 10^{8}$	$4.98\ 10^{-3}$	$5.48\ 10^{-5}$	$1.30\ 10^{-3}$	$1.27\ 10^{-4}$
Cl-36	$1.56\ 10^{8}$	$5.15\ 10^{0}$	$1.60\ 10^{0}$	$2.52\ 10^{0}$	$1.06\ 10^{-7}$
Ca-41	$5.77\ 10^{9}$	$2.45\ 10^{0}$	0	$2.59\ 10^{1}$	$3.36\ 10^{-8}$
Mn-54	$1.12\ 10^{13}$	$5.40\ 10^{5}$	$1.03\ 10^{3}$	$3.23\ 10^{2}$	$2.88\ 10^{0}$
Fe-55	$1.86\ 10^{13}$	$1.15\ 10^{2}$	0	$3.03\ 10^{-2}$	$2.73\ 10^{-2}$
Co-60	$3.58\ 10^{11}$	$4.00\ 10^{5}$	$2.94\ 10^{3}$	$1.96\ 10^{-2}$	$5.44\ 10^{-3}$
Ni-59	$1.95\ 10^{11}$	$5.79\ 10^{-1}$	0	nd	nd
Ni-63	$2.42\ 10^{11}$	$1.76\ 10^{0}$	0	$4.85\ 10^{1}$	$7.66\ 10^{-1}$
Zn-65	$8.95\ 10^{11}$	$3.22\ 10^{4}$	$4.86\ 10^{1}$	$4.39\ 10^{-1}$	$1.84\ 10^{-2}$
Se-79	$8.98\ 10^{8}$	$1.57\ 10^{-1}$	$1.83\ 10^{-4}$	nd	nd
Sr-90	$3.83\ 10^{8}$	$1.65\ 10^{0}$	$2.04\ 10^{-1}$	$3.08\ 10^{2}$	$6.97\ 10^{-6}$
Mo-93	$1.44\ 10^{9}$	$4.00\ 10^{-1}$	$4.59\ 10^{-2}$	nd	nd
Zr-93	$3.12\ 10^{11}$	$2.60\ 10^{1}$	0	$6.08\ 10^{-1}$	$2.06\ 10^{-1}$
Nb-93m	$5.06\ 10^{10}$	$1.41\ 10^{-1}$	$2.64\ 10^{-3}$	nd	nd
Nb-94	$6.09\ 10^{6}$	$9.48\ 10^{-1}$	$8.58\ 10^{-2}$	$1.68\ 10^{0}$	$7.98\ 10^{-8}$
Tc-99	$6.12\ 10^{8}$	$4.25\ 10^{0}$	$3.80\ 10^{-1}$	$1.96\ 10^{-2}$	$1.29\ 10^{-5}$
Ru-106	$9.14\ 10^{11}$	$1.36\ 10^{4}$	$1.26\ 10^{2}$	$6.08\ 10^{-1}$	$6.46\ 10^{-5}$
Ag-108m	$2.65\ 10^{8}$	$2.85\ 10^{2}$	$1.40\ 10^{1}$	$3.34\ 10^{0}$	$2.94\ 10^{-4}$
Ag-110m	$6.41\ 10^{12}$	$4.79\ 10^{6}$	$4.22\ 10^{3}$	$1.72\ 10^{1}$	$6.27\ 10^{0}$
Cd-109	$1.04\ 10^{12}$	$3.78\ 10^{2}$	$4.06\ 10^{0}$	$6.29\ 10^{1}$	$1.78\ 10^{-2}$
Sb-125	$4.17\ 10^{11}$	$4.83\ 10^{5}$	$1.87\ 10^{3}$	$1.82\ 10^{-1}$	$8.92\ 10^{-1}$
Sn-119m	$8.43\ 10^{12}$	$1.80\ 10^{2}$	$3.06\ 10^{-1}$	$3.68\ 10^{-4}$	$1.58\ 10^{1}$
Sn-123	$2.97\ 10^{12}$	$4.39\ 10^{2}$	$3.55\ 10^{-1}$	nd	nd
Sn-126	$4.60\ 10^{6}$	$8.22\ 10^{-1}$	$7.44\ 10^{-2}$	nd	nd
Te-127m	$4.07\ 10^{12}$	$2.00\ 10^{2}$	$1.13\ 10^{-1}$	nd	nd
I-129	$3.01\ 10^{8}$	$2.76\ 10^{1}$	$1.20\ 10^{0}$	$2.97\ 10^{-5}$	$1.01\ 10^{-5}$

Radionuclide	Radiological capacity (MBq)	Leachate treatment worker ($\mu Sv\ y^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}$)	Farming family ($\mu Sv\ y^{-1}$)	Fishing family ($\mu Sv\ y^{-1}$)
Ba-133	$7.18\ 10^{9}$	$1.29\ 10^{5}$	$1.18\ 10^{4}$	nd	nd
Cs-134	$1.01\ 10^{11}$	$8.60\ 10^{3}$	$6.28\ 10^{1}$	$4.21\ 10^{-3}$	$8.93\ 10^{-3}$
Cs-135	$1.55\ 10^{9}$	$2.03\ 10^{-2}$	$1.08\ 10^{-4}$	nd	nd
Cs-137	$9.69\ 10^{8}$	$4.04\ 10^{1}$	$3.19\ 10^{0}$	nd	nd
Ce-144	$4.81\ 10^{12}$	$3.14\ 10^{3}$	$5.28\ 10^{0}$	$1.40\ 10^{-4}$	$5.77\ 10^{-2}$
Pm-147	$2.14\ 10^{13}$	$2.01\ 10^{2}$	$2.54\ 10^{-1}$	$2.12\ 10^{-4}$	$4.78\ 10^{-1}$
Sm-147	$4.81\ 10^{8}$	$1.63\ 10^{0}$	0	nd	nd
Sm-151	$7.23\ 10^{11}$	$1.71\ 10^{0}$	$1.33\ 10^{-3}$	$1.43\ 10^{-2}$	$7.68\ 10^{-1}$
Eu-152	$8.05\ 10^{9}$	$5.38\ 10^{3}$	$1.51\ 10^{2}$	$1.55\ 10^{-2}$	$1.60\ 10^{-3}$
Eu-154	$4.18\ 10^{10}$	$2.97\ 10^{4}$	$5.53\ 10^{2}$	$1.89\ 10^{-3}$	$5.45\ 10^{-4}$
Eu-155	$8.81\ 10^{12}$	$1.39\ 10^{5}$	$1.45\ 10^{3}$	$3.98\ 10^{-3}$	$2.02\ 10^{0}$
Gd-153	$4.83\ 10^{13}$	$2.21\ 10^{5}$	$3.29\ 10^{2}$	$4.28\ 10^{-3}$	$1.11\ 10^{1}$
Pb-210	$4.85\ 10^{8}$	$1.01\ 10^{1}$	$1.42\ 10^{-3}$	$7.76\ 10^{-3}$	$1.94\ 10^{-6}$
Po-210	$6.17\ 10^{9}$	$1.60\ 10^{2}$	$6.79\ 10^{-6}$	$5.77\ 10^{-3}$	$3.81\ 10^{-6}$
Ra-226	$3.89\ 10^{6}$	$4.67\ 10^{-1}$	$3.76\ 10^{-2}$	$3.47\ 10^{-3}$	$8.39\ 10^{-7}$
Ra-228	$2.25\ 10^{10}$	$8.07\ 10^{2}$	$3.91\ 10^{1}$	$3.29\ 10^{-4}$	$2.15\ 10^{-5}$
Ac-227	$3.04\ 10^{9}$	$6.39\ 10^{2}$	$2.78\ 10^{0}$	$2.88\ 10^{-4}$	$1.78\ 10^{-6}$
Th-228	$1.72\ 10^{11}$	$2.88\ 10^{4}$	$6.55\ 10^{1}$	nd	nd
Th-229	$2.88\ 10^{7}$	$3.08\ 10^{0}$	$5.25\ 10^{-2}$	$4.09\ 10^{0}$	$3.26\ 10^{-7}$
Th-230	$1.98\ 10^{6}$	$5.48\ 10^{-2}$	$2.75\ 10^{-6}$	nd	nd
Th-232	$7.95\ 10^{6}$	$1.50\ 10^{0}$	$5.53\ 10^{-2}$	$4.78\ 10^{1}$	$1.36\ 10^{-6}$
Pa-231	$1.36\ 10^{7}$	$6.12\ 10^{-1}$	$2.83\ 10^{-3}$	nd	nd
U-232	$4.04\ 10^{8}$	$1.17\ 10^{1}$	$2.96\ 10^{-3}$	nd	nd
U-233	$1.02\ 10^{8}$	$3.55\ 10^{-1}$	$1.56\ 10^{-3}$	nd	nd
U-234	$1.45\ 10^{8}$	$4.87\ 10^{-1}$	$6.37\ 10^{-4}$	nd	nd
U-235	$6.93\ 10^{7}$	$1.69\ 10^{0}$	$5.46\ 10^{-1}$	$1.32\ 10^{2}$	$6.78\ 10^{-6}$
U-236	$1.48\ 10^{9}$	$4.62\ 10^{0}$	$3.48\ 10^{-3}$	nd	nd
U-238	$1.60\ 10^{9}$	$4.70\ 10^{0}$	$1.81\ 10^{-3}$	nd	nd
Np-237	$1.42\ 10^{7}$	$6.72\ 10^{0}$	$6.86\ 10^{-2}$	nd	nd
Pu-238	$7.56\ 10^{8}$	$3.28\ 10^{1}$	$2.67\ 10^{-4}$	nd	nd
Pu-239	$1.55\ 10^{8}$	$7.43\ 10^{0}$	$1.35\ 10^{-4}$	nd	nd
Pu-240	$1.89\ 10^{8}$	$9.03\ 10^{0}$	$8.16\ 10^{-5}$	$9.95\ 10^{-2}$	$1.35\ 10^{-7}$
Pu-241	$9.39\ 10^{9}$	$8.19\ 10^{0}$	$5.09\ 10^{-5}$	nd	nd
Pu-242	$1.58\ 10^{8}$	$6.96\ 10^{0}$	$5.99\ 10^{-5}$	nd	nd
Pu-244	$1.26\ 10^{8}$	$5.53\ 10^{0}$	$2.81\ 10^{-5}$	nd	nd
Am-241	$3.03\ 10^{8}$	$6.08\ 10^{0}$	$1.06\ 10^{-2}$	$2.95\ 10^{-1}$	$3.67\ 10^{-6}$

Radionuclide	Radiological capacity (MBq)	Leachate treatment worker ($\mu Sv\ y^{-1}$)	Reed bed treatment worker ($\mu Sv\ y^{-1}$)	Farming family ($\mu Sv\ y^{-1}$)	Fishing family ($\mu Sv\ y^{-1}$)
Am-242m	$1.75\ 10^{7}$	$4.08\ 10^{-1}$	$2.14\ 10^{-5}$	nd	nd
Am-243	$1.46\ 10^{8}$	$3.17\ 10^{0}$	$1.74\ 10^{-2}$	nd	nd
Cm-242	$1.48\ 10^{11}$	$1.25\ 10^{1}$	$1.71\ 10^{-5}$	nd	nd
Cm-243	$4.89\ 10^{7}$	$3.14\ 10^{-1}$	$3.48\ 10^{-3}$	$7.84\ 10^{-4}$	$2.88\ 10^{-7}$
Cm-244	$1.16\ 10^{8}$	$3.60\ 10^{-1}$	$1.30\ 10^{-6}$	$8.83\ 10^{-5}$	$1.07\ 10^{-7}$
Cm-245	$1.26\ 10^{7}$	$9.00\ 10^{-2}$	$1.00\ 10^{-3}$	nd	nd
Cm-246	$1.27\ 10^{7}$	$7.04\ 10^{-2}$	$3.45\ 10^{-7}$	nd	nd
Cm-248	$1.45\ 10^{7}$	$2.96\ 10^{-1}$	$3.00\ 10^{-7}$	nd	nd

E.3.8. Dose resulting from exposure to waste from a dropped container

686. This scenario applies during the pre-closure phase and the exposed groups are workers and the public.

687. This scenario was also addressed for the ENRMF using a radiological risk assessment for occupational exposure completed by the HPA (Annex C, (Augean, 2009)). Their conclusion was that with appropriate precautions the worker exposure can be kept within the site criterion under the unlikely circumstance of a dropped container which gives rise to a release.

688. This scenario is not used to constrain landfill capacity because it is independent of the tonnage disposed at Port Clarence. However, this scenario is one of the scenarios used to determine the proposed radionuclide activity concentration limits for packaged wastes and for loose tipped wastes (see Section 7.4.2.3 for further details).

689. The dose criteria are the legal dose limit to workers of 20 mSv y^{-1}, the site criterion of 1 mSv y^{-1} for workers and the dose constraint for the public of 0.3 mSv y^{-1}.

Potentially exposed group

690. The assessment of doses from waste released to atmosphere following a dropped load during the operational phase is based on that used in the ENRMF assessment (Augean, 2009). The exposed groups are the public and workers. Members of the exposed groups are assumed to be adults and be exposed as a result of inhalation of contaminated dust.

691. The load is assumed to be a flexible container that spills a proportion of its load, assumed to contain the maximum activity concentration of a single nuclide. The distance to the nearest exposed member of the public is cautiously assumed to be 50 m and the event duration is 30 minutes. The worker remains very close to the dropped waste without taking precautions or retreating for at least 30 minutes. The worker inhalation rate is used for both worker and the public in the assessment (Table 93).

E.3.8.1. Scenario assumptions for estimating doses following a dropped load

692. The scenario is not contained within the SNIFFER model and has been separately addressed. Exposure to both workers and the public has been calculated under the following assumptions using the UKAEA dropped load methodology from the safety assessment handbook [reference 22 of (Augean, 2009)].

693. The assumptions are as follows.

- A one cubic metre flexible container of wastes is dropped and spills 10% of its contents through broken seams.
- The bag is filled with a dry solid.
- The bag contains a single nuclide at 200 Bq g^{-1}.
- The bag weighs 1 tonne.
- The distance to the nearest public is 50 m and the event duration is 30 minutes.
- The worker remains very close to the dropped waste without taking precautions or retreating for at least 30 minutes.
- The atmospheric conditions are worst case, still conditions.

E.3.8.2. Assessment calculation involving a dropped load

694. The dose arising from the inhalation of contaminated material is given by:

$$Dose_{inh} = \frac{I \cdot RF_1 \cdot RF_2 \cdot C \cdot B \cdot D_{inh}^{Rn}}{DF}$$

where:

- I is the inventory of radionuclide *Rn* releasable (Bq), 10% of bag content (2 10^7 Bq);
- RF_1 is the release fraction;
- RF_2 is the respirable fraction;
- C is the dispersion coefficient (s m^{-3});
- B is the inhalation rate (m^3 s^{-1});
- D_{inh}^{Rn} is the inhalation dose coefficient for radionuclide *Rn* (Sv Bq^{-1}); and,
- DF is the decontamination factor.

695. The parameters used in this calculation are given in Table 93. The inhalation dose coefficients are given in Table 200.

Table 93 Dropped container parameters

Parameter	Units	Value	Description	
I	Bq	2 10^{8}	Radionuclide inventory	
RF_1		1 10^{-3}	Release fraction	
RF_2		0.1	Respirable fraction	
C	s m^{-3}	5	Dispersion coefficient	Worker
		1.7 10^{-2}		Public
B	m^3 s^{-1}	3.3 10^{-4}	Inhalation rate	Worker
		2.78 10^{-4}		Adult
		1.78 10^{-4}		Child
		6.11 10^{-5}		Infant
DF		1	Decontamination factor	

E.3.8.3. Dose from a dropped load

696. The effective doses arising from a dropped container are given in Table 94. The results for Ra-226 are independent of the Ra-226 placement depth in the site.

Table 94 Doses from a dropped container containing waste with 200 Bq g^{-1} of the radionuclide

Radionuclide	Dropped load dose assuming 200 Bq g^{-1} in the waste			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
H-3	8.67 10^{-7}	2.46 10^{-9}	3.59 10^{-9}	9.44 10^{-9}
C-14	1.93 10^{-5}	5.48 10^{-8}	6.99 10^{-8}	1.61 10^{-7}
Cl-36	2.43 10^{-5}	6.89 10^{-8}	9.44 10^{-8}	2.46 10^{-7}
Ca-41	6.00 10^{-7}	1.70 10^{-9}	3.12 10^{-9}	5.67 10^{-9}
Mn-54	5.00 10^{-6}	1.42 10^{-8}	2.27 10^{-8}	5.86 10^{-8}
Fe-55	2.57 10^{-6}	7.27 10^{-9}	1.32 10^{-8}	3.02 10^{-8}
Co-60	1.03 10^{-4}	2.93 10^{-7}	3.78 10^{-7}	8.12 10^{-7}
Ni-59	1.47 10^{-6}	4.16 10^{-9}	5.57 10^{-9}	1.42 10^{-8}
Ni-63	4.33 10^{-6}	1.23 10^{-8}	1.61 10^{-8}	4.06 10^{-8}
Zn-65	7.33 10^{-6}	2.08 10^{-8}	3.59 10^{-8}	9.44 10^{-8}
Se-79	2.27 10^{-5}	6.42 10^{-8}	8.22 10^{-8}	1.89 10^{-7}
Sr-90	5.38 10^{-4}	1.53 10^{-6}	1.73 10^{-6}	3.86 10^{-6}
Mo-93	7.67 10^{-6}	2.17 10^{-8}	2.64 10^{-8}	5.48 10^{-8}
Zr-93	8.33 10^{-5}	2.36 10^{-7}	9.16 10^{-8}	6.04 10^{-8}
Nb-93m	6.00 10^{-6}	1.70 10^{-8}	2.36 10^{-8}	6.14 10^{-8}
Nb-94	1.63 10^{-4}	4.63 10^{-7}	5.48 10^{-7}	1.13 10^{-6}
Tc-99	4.33 10^{-5}	1.23 10^{-7}	1.61 10^{-7}	3.49 10^{-7}

Radionuclide	Dropped load dose assuming 200 Bq g^{-1} in the waste			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
Ru-106	$2.20\ 10^{-4}$	$6.23\ 10^{-7}$	$8.59\ 10^{-7}$	$2.17\ 10^{-6}$
Ag-108m	$1.23\ 10^{-4}$	$3.49\ 10^{-7}$	$4.16\ 10^{-7}$	$8.22\ 10^{-7}$
Ag-110m	$4.00\ 10^{-5}$	$1.13\ 10^{-7}$	$1.70\ 10^{-7}$	$3.87\ 10^{-7}$
Cd-109	$2.70\ 10^{-5}$	$7.65\ 10^{-8}$	$1.32\ 10^{-7}$	$3.49\ 10^{-7}$
Sb-125	$4.32\ 10^{-5}$	$1.22\ 10^{-7}$	$1.64\ 10^{-7}$	$3.87\ 10^{-7}$
Sn-119m	$7.33\ 10^{-6}$	$2.08\ 10^{-8}$	$2.93\ 10^{-8}$	$7.46\ 10^{-8}$
Sn-123	$2.70\ 10^{-5}$	$7.65\ 10^{-8}$	$1.13\ 10^{-7}$	$2.93\ 10^{-7}$
Sn-126	$9.49\ 10^{-5}$	$2.69\ 10^{-7}$	$3.94\ 10^{-7}$	$9.65\ 10^{-7}$
Te-127m	$3.27\ 10^{-5}$	$9.26\ 10^{-8}$	$1.32\ 10^{-7}$	$3.12\ 10^{-7}$
I-129	$1.20\ 10^{-4}$	$3.40\ 10^{-7}$	$6.33\ 10^{-7}$	$8.12\ 10^{-7}$
Ba-133	$3.33\ 10^{-5}$	$9.44\ 10^{-8}$	$1.23\ 10^{-7}$	$2.74\ 10^{-7}$
Cs-134	$6.67\ 10^{-5}$	$1.89\ 10^{-7}$	$2.64\ 10^{-7}$	$5.95\ 10^{-7}$
Cs-135	$2.87\ 10^{-5}$	$8.12\ 10^{-8}$	$1.04\ 10^{-7}$	$2.27\ 10^{-7}$
Cs-137	$1.30\ 10^{-4}$	$3.68\ 10^{-7}$	$4.53\ 10^{-7}$	$9.44\ 10^{-7}$
Ce-144	$1.77\ 10^{-4}$	$5.01\ 10^{-7}$	$7.37\ 10^{-7}$	$2.55\ 10^{-6}$
Pm-147	$1.67\ 10^{-5}$	$4.72\ 10^{-8}$	$6.61\ 10^{-8}$	$1.70\ 10^{-7}$
Sm-147	$3.20\ 10^{-2}$	$9.07\ 10^{-5}$	$1.04\ 10^{-4}$	$2.17\ 10^{-4}$
Sm-151	$1.33\ 10^{-5}$	$3.78\ 10^{-8}$	$4.25\ 10^{-8}$	$9.44\ 10^{-8}$
Eu-152	$1.40\ 10^{-4}$	$3.97\ 10^{-7}$	$4.63\ 10^{-7}$	$9.44\ 10^{-7}$
Eu-154	$1.77\ 10^{-4}$	$5.01\ 10^{-7}$	$6.14\ 10^{-7}$	$1.42\ 10^{-6}$
Eu-155	$2.30\ 10^{-5}$	$6.52\ 10^{-8}$	$8.69\ 10^{-8}$	$2.17\ 10^{-7}$
Gd-153	$7.00\ 10^{-6}$	$1.98\ 10^{-8}$	$3.68\ 10^{-8}$	$1.13\ 10^{-7}$
Pb-210	$3.33\ 10^{-2}$	$9.44\ 10^{-5}$	$1.25\ 10^{-4}$	$3.05\ 10^{-4}$
Po-210	$1.43\ 10^{-2}$	$4.06\ 10^{-5}$	$5.57\ 10^{-5}$	$1.32\ 10^{-4}$
Ra-226	$6.51\ 10^{-2}$	$1.84\ 10^{-4}$	$2.39\ 10^{-4}$	$5.80\ 10^{-4}$
Ra-228	$1.99\ 10^{-1}$	$5.63\ 10^{-4}$	$7.53\ 10^{-4}$	$1.96\ 10^{-3}$
Ac-227	$1.90\ 10^{0}$	$5.37\ 10^{-3}$	$7.03\ 10^{-3}$	$1.56\ 10^{-2}$
Th-228	$1.45\ 10^{-1}$	$4.12\ 10^{-4}$	$5.64\ 10^{-4}$	$1.51\ 10^{-3}$
Th-229	$8.54\ 10^{-1}$	$2.42\ 10^{-3}$	$2.94\ 10^{-3}$	$5.24\ 10^{-3}$
Th-230	$3.33\ 10^{-1}$	$9.44\ 10^{-4}$	$1.04\ 10^{-3}$	$1.89\ 10^{-3}$
Th-232	$5.65\ 10^{-1}$	$1.60\ 10^{-3}$	$1.98\ 10^{-3}$	$4.04\ 10^{-3}$
Pa-231	$4.67\ 10^{-1}$	$1.32\ 10^{-3}$	$1.42\ 10^{-3}$	$2.17\ 10^{-3}$
U-232	$2.69\ 10^{-1}$	$7.61\ 10^{-4}$	$9.70\ 10^{-4}$	$2.43\ 10^{-3}$
U-233	$3.20\ 10^{-2}$	$9.07\ 10^{-5}$	$1.13\ 10^{-4}$	$2.83\ 10^{-4}$
U-234	$3.13\ 10^{-2}$	$8.88\ 10^{-5}$	$1.13\ 10^{-4}$	$2.74\ 10^{-4}$

Radionuclide	Dropped load dose assuming 200 Bq g^{-1} in the waste			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
U-235	$2.83\ 10^{-2}$	$8.03\ 10^{-5}$	$1.04\ 10^{-4}$	$2.46\ 10^{-4}$
U-236	$2.90\ 10^{-2}$	$8.22\ 10^{-5}$	$1.04\ 10^{-4}$	$2.55\ 10^{-4}$
U-238	$2.67\ 10^{-2}$	$7.56\ 10^{-5}$	$9.45\ 10^{-5}$	$2.36\ 10^{-4}$
Np-237	$1.67\ 10^{-1}$	$4.72\ 10^{-4}$	$4.72\ 10^{-4}$	$8.78\ 10^{-4}$
Pu-238	$3.67\ 10^{-1}$	$1.04\ 10^{-3}$	$1.04\ 10^{-3}$	$1.79\ 10^{-3}$
Pu-239	$4.00\ 10^{-1}$	$1.13\ 10^{-3}$	$1.13\ 10^{-3}$	$1.89\ 10^{-3}$
Pu-240	$4.00\ 10^{-1}$	$1.13\ 10^{-3}$	$1.13\ 10^{-3}$	$1.89\ 10^{-3}$
Pu-241	$7.67\ 10^{-3}$	$2.17\ 10^{-5}$	$2.27\ 10^{-5}$	$2.74\ 10^{-5}$
Pu-242	$3.67\ 10^{-1}$	$1.04\ 10^{-3}$	$1.13\ 10^{-3}$	$1.79\ 10^{-3}$
Pu-244	$3.67\ 10^{-1}$	$1.04\ 10^{-3}$	$1.13\ 10^{-3}$	$1.79\ 10^{-3}$
Am-241	$3.20\ 10^{-1}$	$9.07\ 10^{-4}$	$9.44\ 10^{-4}$	$1.70\ 10^{-3}$
Am-242m	$3.86\ 10^{-1}$	$1.09\ 10^{-3}$	$1.15\ 10^{-3}$	$1.89\ 10^{-3}$
Am-243	$3.20\ 10^{-1}$	$9.07\ 10^{-4}$	$9.44\ 10^{-4}$	$1.61\ 10^{-3}$
Cm-242	$1.97\ 10^{-2}$	$5.57\ 10^{-5}$	$7.74\ 10^{-5}$	$1.98\ 10^{-4}$
Cm-243	$2.31\ 10^{-1}$	$6.54\ 10^{-4}$	$6.92\ 10^{-4}$	$1.42\ 10^{-3}$
Cm-244	$1.90\ 10^{-1}$	$5.38\ 10^{-4}$	$5.76\ 10^{-4}$	$1.23\ 10^{-3}$
Cm-245	$3.30\ 10^{-1}$	$9.35\ 10^{-4}$	$9.44\ 10^{-4}$	$1.70\ 10^{-3}$
Cm-246	$3.27\ 10^{-1}$	$9.26\ 10^{-4}$	$9.44\ 10^{-4}$	$1.70\ 10^{-3}$
Cm-248	$1.20\ 10^{0}$	$3.40\ 10^{-3}$	$3.49\ 10^{-3}$	$6.14\ 10^{-3}$

697. The doses meet the site criterion for workers for all radionuclides except Ac-227 and Cm-248 (the estimated doses are is less than the criterion of 6 mSv for classifying workers as radiation workers), and all doses to the public are below 0.02 mSv. Ac-227 and Cm-248 are very unlikely to be present at 200 Bq g^{-1} given the low occurrence of these radionuclides (NDA, 2013). In addition, the above assessment calculations assume that the bag is filled with a loose dry material that disperses readily, that the package fails and that the worker does not respond correctly. These are highly conservative assumptions. Hence, studies at for the ENRMF (Augean, 2009) concluded that an activity concentration of 200 Bq g^{-1} could be applied to all radionuclides in the wastes disposed of at the site. The same conclusion is also valid for Port Clarence.

698. A key measure to mitigate dropped load dispersion events will be to engineer the waste containers such that they withstand or substantially withstand accidental drops during handling. Where drums are used these will be rated under existing dangerous good transport regulations for radioactive material to withstand a drop test. Flexible containers may only be used where this is acceptable under dangerous goods transport regulations.

699. This scenario has not been used to constrain the radiological capacity because it has a low probability of occurrence and is independent of the total tonnage and total activity received at Port Clarence.

700. However, this scenario is one of the scenarios used to determine the proposed radionuclide activity concentration limits for packaged wastes and for loose tipped wastes (see Section 7.4.2.3 for further details).

E.3.8.4. Dose from spillage from a tipper

701. The model for a dropped load has been adapted to consider a greater spillage (20%) from a tipper truck load (20 t) in order to assess the potential impact of a load destined for loose tipping. The potential impact from a different specific activity concentration in the waste is calculated by scaling the doses given in Table 95 by the activity concentration (see Section 7.4.2.3).

Table 95 Doses from a tipper truck load spillage assuming 200 Bq g^{-1} in the load

Radionuclide	Tipper truck load spillage			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
H-3	$3.47\ 10^{-5}$	$9.82\ 10^{-8}$	$1.44\ 10^{-7}$	$3.78\ 10^{-7}$
C-14	$7.73\ 10^{-4}$	$2.19\ 10^{-6}$	$2.80\ 10^{-6}$	$6.42\ 10^{-6}$
Cl-36	$9.73\ 10^{-4}$	$2.76\ 10^{-6}$	$3.78\ 10^{-6}$	$9.82\ 10^{-6}$
Ca-41	$2.40\ 10^{-5}$	$6.80\ 10^{-8}$	$1.25\ 10^{-7}$	$2.27\ 10^{-7}$
Mn-54	$2.00\ 10^{-4}$	$5.67\ 10^{-7}$	$9.07\ 10^{-7}$	$2.34\ 10^{-6}$
Fe-55	$1.03\ 10^{-4}$	$2.91\ 10^{-7}$	$5.29\ 10^{-7}$	$1.21\ 10^{-6}$
Co-60	$4.13\ 10^{-3}$	$1.17\ 10^{-5}$	$1.51\ 10^{-5}$	$3.25\ 10^{-5}$
Ni-59	$5.87\ 10^{-5}$	$1.66\ 10^{-7}$	$2.23\ 10^{-7}$	$5.67\ 10^{-7}$
Ni-63	$1.73\ 10^{-4}$	$4.91\ 10^{-7}$	$6.42\ 10^{-7}$	$1.62\ 10^{-6}$
Zn-65	$2.93\ 10^{-4}$	$8.31\ 10^{-7}$	$1.44\ 10^{-6}$	$3.78\ 10^{-6}$
Se-79	$9.07\ 10^{-4}$	$2.57\ 10^{-6}$	$3.29\ 10^{-6}$	$7.56\ 10^{-6}$
Sr-90	$2.15\ 10^{-2}$	$6.10\ 10^{-5}$	$6.90\ 10^{-5}$	$1.54\ 10^{-4}$
Mo-93	$3.07\ 10^{-4}$	$8.69\ 10^{-7}$	$1.06\ 10^{-6}$	$2.19\ 10^{-6}$
Zr-93	$3.33\ 10^{-3}$	$9.44\ 10^{-6}$	$3.66\ 10^{-6}$	$2.42\ 10^{-6}$
Nb-93m	$2.40\ 10^{-4}$	$6.80\ 10^{-7}$	$9.44\ 10^{-7}$	$2.46\ 10^{-6}$
Nb-94	$6.53\ 10^{-3}$	$1.85\ 10^{-5}$	$2.19\ 10^{-5}$	$4.53\ 10^{-5}$
Tc-99	$1.73\ 10^{-3}$	$4.91\ 10^{-6}$	$6.42\ 10^{-6}$	$1.40\ 10^{-5}$
Ru-106	$8.80\ 10^{-3}$	$2.49\ 10^{-5}$	$3.44\ 10^{-5}$	$8.69\ 10^{-5}$
Ag-108m	$4.93\ 10^{-3}$	$1.40\ 10^{-5}$	$1.66\ 10^{-5}$	$3.29\ 10^{-5}$
Ag-110m	$1.60\ 10^{-3}$	$4.53\ 10^{-6}$	$6.80\ 10^{-6}$	$1.55\ 10^{-5}$

Radionuclide	Tipper truck load spillage			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
Cd-109	$1.08\ 10^{-3}$	$3.06\ 10^{-6}$	$5.29\ 10^{-6}$	$1.40\ 10^{-5}$
Sb-125	$1.73\ 10^{-3}$	$4.90\ 10^{-6}$	$6.54\ 10^{-6}$	$1.55\ 10^{-5}$
Sn-119m	$2.93\ 10^{-4}$	$8.31\ 10^{-7}$	$1.17\ 10^{-6}$	$2.98\ 10^{-6}$
Sn-123	$1.08\ 10^{-3}$	$3.06\ 10^{-6}$	$4.53\ 10^{-6}$	$1.17\ 10^{-5}$
Sn-126	$3.80\ 10^{-3}$	$1.08\ 10^{-5}$	$1.58\ 10^{-5}$	$3.86\ 10^{-5}$
Te-127m	$1.31\ 10^{-3}$	$3.70\ 10^{-6}$	$5.29\ 10^{-6}$	$1.25\ 10^{-5}$
I-129	$4.80\ 10^{-3}$	$1.36\ 10^{-5}$	$2.53\ 10^{-5}$	$3.25\ 10^{-5}$
Ba-133	$1.33\ 10^{-3}$	$3.78\ 10^{-6}$	$4.91\ 10^{-6}$	$1.10\ 10^{-5}$
Cs-134	$2.67\ 10^{-3}$	$7.56\ 10^{-6}$	$1.06\ 10^{-5}$	$2.38\ 10^{-5}$
Cs-135	$1.15\ 10^{-3}$	$3.25\ 10^{-6}$	$4.16\ 10^{-6}$	$9.07\ 10^{-6}$
Cs-137	$5.20\ 10^{-3}$	$1.47\ 10^{-5}$	$1.81\ 10^{-5}$	$3.78\ 10^{-5}$
Ce-144	$7.07\ 10^{-3}$	$2.00\ 10^{-5}$	$2.95\ 10^{-5}$	$1.02\ 10^{-4}$
Pm-147	$6.67\ 10^{-4}$	$1.89\ 10^{-6}$	$2.64\ 10^{-6}$	$6.80\ 10^{-6}$
Sm-147	$1.28\ 10^{0}$	$3.63\ 10^{-3}$	$4.16\ 10^{-3}$	$8.69\ 10^{-3}$
Sm-151	$5.33\ 10^{-4}$	$1.51\ 10^{-6}$	$1.70\ 10^{-6}$	$3.78\ 10^{-6}$
Eu-152	$5.60\ 10^{-3}$	$1.59\ 10^{-5}$	$1.85\ 10^{-5}$	$3.78\ 10^{-5}$
Eu-154	$7.07\ 10^{-3}$	$2.00\ 10^{-5}$	$2.46\ 10^{-5}$	$5.67\ 10^{-5}$
Eu-155	$9.20\ 10^{-4}$	$2.61\ 10^{-6}$	$3.48\ 10^{-6}$	$8.69\ 10^{-6}$
Gd-153	$2.80\ 10^{-4}$	$7.93\ 10^{-7}$	$1.47\ 10^{-6}$	$4.53\ 10^{-6}$
Pb-210	$1.33\ 10^{0}$	$3.78\ 10^{-3}$	$5.00\ 10^{-3}$	$1.22\ 10^{-2}$
Po-210	$5.73\ 10^{-1}$	$1.62\ 10^{-3}$	$2.23\ 10^{-3}$	$5.29\ 10^{-3}$
Ra-226	$2.60\ 10^{0}$	$7.37\ 10^{-3}$	$9.55\ 10^{-3}$	$2.32\ 10^{-2}$
Ra-228	$7.95\ 10^{0}$	$2.25\ 10^{-2}$	$3.01\ 10^{-2}$	$7.86\ 10^{-2}$
Ac-227	$7.58\ 10^{1}$	$2.15\ 10^{-1}$	$2.81\ 10^{-1}$	$6.25\ 10^{-1}$
Th-228	$5.82\ 10^{0}$	$1.65\ 10^{-2}$	$2.26\ 10^{-2}$	$6.04\ 10^{-2}$
Th-229	$3.42\ 10^{1}$	$9.68\ 10^{-2}$	$1.18\ 10^{-1}$	$2.10\ 10^{-1}$
Th-230	$1.33\ 10^{1}$	$3.78\ 10^{-2}$	$4.16\ 10^{-2}$	$7.56\ 10^{-2}$
Th-232	$2.26\ 10^{1}$	$6.41\ 10^{-2}$	$7.92\ 10^{-2}$	$1.62\ 10^{-1}$
Pa-231	$1.87\ 10^{1}$	$5.29\ 10^{-2}$	$5.67\ 10^{-2}$	$8.69\ 10^{-2}$
U-232	$1.07\ 10^{1}$	$3.05\ 10^{-2}$	$3.88\ 10^{-2}$	$9.70\ 10^{-2}$
U-233	$1.28\ 10^{0}$	$3.63\ 10^{-3}$	$4.53\ 10^{-3}$	$1.13\ 10^{-2}$
U-234	$1.25\ 10^{0}$	$3.55\ 10^{-3}$	$4.53\ 10^{-3}$	$1.10\ 10^{-2}$
U-235	$1.13\ 10^{0}$	$3.21\ 10^{-3}$	$4.16\ 10^{-3}$	$9.82\ 10^{-3}$
U-236	$1.16\ 10^{0}$	$3.29\ 10^{-3}$	$4.16\ 10^{-3}$	$1.02\ 10^{-2}$
U-238	$1.07\ 10^{0}$	$3.03\ 10^{-3}$	$3.78\ 10^{-3}$	$9.46\ 10^{-3}$

Radionuclide	Tipper truck load spillage			
	Worker (mSv)	Public Adult (mSv)	Public Child (mSv)	Public Infant (mSv)
Np-237	$6.67\ 10^{0}$	$1.89\ 10^{-2}$	$1.89\ 10^{-2}$	$3.51\ 10^{-2}$
Pu-238	$1.47\ 10^{1}$	$4.16\ 10^{-2}$	$4.16\ 10^{-2}$	$7.18\ 10^{-2}$
Pu-239	$1.60\ 10^{1}$	$4.53\ 10^{-2}$	$4.53\ 10^{-2}$	$7.56\ 10^{-2}$
Pu-240	$1.60\ 10^{1}$	$4.53\ 10^{-2}$	$4.53\ 10^{-2}$	$7.56\ 10^{-2}$
Pu-241	$3.07\ 10^{-1}$	$8.69\ 10^{-4}$	$9.07\ 10^{-4}$	$1.10\ 10^{-3}$
Pu-242	$1.47\ 10^{1}$	$4.16\ 10^{-2}$	$4.53\ 10^{-2}$	$7.18\ 10^{-2}$
Pu-244	$1.47\ 10^{1}$	$4.16\ 10^{-2}$	$4.53\ 10^{-2}$	$7.18\ 10^{-2}$
Am-241	$1.28\ 10^{1}$	$3.63\ 10^{-2}$	$3.78\ 10^{-2}$	$6.80\ 10^{-2}$
Am-242m	$1.54\ 10^{1}$	$4.38\ 10^{-2}$	$4.59\ 10^{-2}$	$7.56\ 10^{-2}$
Am-243	$1.28\ 10^{1}$	$3.63\ 10^{-2}$	$3.78\ 10^{-2}$	$6.42\ 10^{-2}$
Cm-242	$7.87\ 10^{-1}$	$2.23\ 10^{-3}$	$3.10\ 10^{-3}$	$7.93\ 10^{-3}$
Cm-243	$9.24\ 10^{0}$	$2.62\ 10^{-2}$	$2.77\ 10^{-2}$	$5.69\ 10^{-2}$
Cm-244	$7.60\ 10^{0}$	$2.15\ 10^{-2}$	$2.30\ 10^{-2}$	$4.91\ 10^{-2}$
Cm-245	$1.32\ 10^{1}$	$3.74\ 10^{-2}$	$3.78\ 10^{-2}$	$6.80\ 10^{-2}$
Cm-246	$1.31\ 10^{1}$	$3.70\ 10^{-2}$	$3.78\ 10^{-2}$	$6.80\ 10^{-2}$
Cm-248	$4.80\ 10^{1}$	$1.36\ 10^{-1}$	$1.40\ 10^{-1}$	$2.46\ 10^{-1}$

E.3.9. Wound exposure

702. Radionuclides can enter the body via wounds and absorption through intact skin. This is not a reasonably foreseeable scenario under normal circumstances. However, it is a possible accident scenario.

703. Therefore, exposure due to radionuclides embedded in a wound is relevant to landfill site workers during the pre-closure phase. The scenario is not considered in the SNIFFER landfill assessment model.

704. While much of the material may be retained at the wound site, soluble material can be transferred to the blood and hence to other parts of the body. Insoluble material will be slowly translocated to regional lymphatic tissue, where it will gradually dissolve and eventually enter the blood (IAEA, 2004). A variable fraction of insoluble material can be retained at the wound site or in lymphatic tissue for the life of the individual. If the materials deposited in a wound are soluble, then they may translocate to the blood with a time course that depends on their dissolution rate in vivo. The distribution of this soluble component will, in most instances, be similar to that entering the blood from the lungs or GI tract. The biokinetic models developed by the ICRP can be used for the calculation of the effective dose arising from the soluble component once the systemic uptake has been determined. As a first approximation, data for direct uptake to blood (injection) can be used.

705. The National Council on Radiation Protection and Measurements Report No. 156 (NCRP, 2007), presents a biokinetic model for intakes of radionuclides via contaminated wounds. The model comprises seven categories that describe the behaviour of the injected radioactive material as a function of its physical and chemical form. Materials injected in soluble form are described by their retention at the wound site as either weakly retained, moderately retained, strongly retained or avidly retained, in order of increasing retention half-time, as determined primarily in rats. Three additional categories: Colloid, Particle, and Fragment, complete the classification. The first four categories reflect the compound's solubility in water, whereas the Colloid and Particle categories are based on the behaviour of injected plutonium compounds in animal models, and the Fragment category is based on the behaviour of uranium metal implants in animal models. A further distinction is made between particles and fragments in that fragments are too large to be ingested by connective tissue macrophages, i.e., fragments are greater than 100 μm in any dimension.

706. In order to assess the dose arising from contamination entering a wound it is necessary to estimate the quantity of material in the wound and the category of the contamination. A person would clean the wound, so removing some of the contamination. In one case (Schadilov AE, 2010), the residual contamination within the wound after cleaning amounted to 0.05% of the initial contamination; in another case about 70% of the initial wound activity was removed by physical (surgical) means (Bailey BR, 2003). It was also noted (Toohey RE, n.d.) that the activity in the body cleared more quickly than was assumed in the NCRP model. It has been remarked that more than 80% of contaminated puncture wounds exceed 1 mm in depth (Ilyn LA, 2001).

707. Two situations are considered here: a minor cut that is ignored and a more significant gash that is cleaned promptly.

708. A minor cut 10 mm long and 1mm wide is considered. The top layer of skin is the keratinised 'dead' layer so no transfer into the blood stream is assumed to take place in this layer. On the palm of the hand this depth is about 400 microns, and the average over the body is 70 microns. It is assumed that the wound extends 0.5 mm into the 'active' layer below this keratinised layer, causing a small amount of bleeding which soon stops. The wound is then left to 'heal itself' with the contaminated material (dust) still in place. The quantity of dust in the wound is 10x1x0.5 mm^3 i.e. 5 mm^3, corresponding to 0.0078 g using a density of 1.53 g cm^{-3}.

709. A gash 4 cm long is assumed to be contaminated to a depth of 2 mm and to be 1 mm wide. If full of contaminated dust this would contain 80 mm^3 (0.08 cm^3), corresponding to 0.2 g of dust. This would be attended to promptly as it would bleed and be painful. Assuming that the wound is cleaned up within a few hours and that 95% is removed, this leaves 0.0061 g of contaminated concrete in the wound.

710. The two different scenarios result in similar estimates of the quantity remaining in the wound. Hence, a reasonable assumption to use for this scenario would be to assume that 0.01 g of material is in the wound.

711. The effective dose coefficients using the NCRP model are presented in (Toohey RE, n.d.) and dose coefficients for injection are presented in (IAEA, 2004). Doses were

calculated from the injection dose coefficients and from each NCRP category dose coefficient.

712. In practice, material likely to be entering a wound would be dust or grit, which are not soluble. As such, using the 'fragment' category dose coefficient is the most realistic. The maximum dose from all NCRP categories and the dose using the fragment dose coefficient for a waste activity concentration of 1 Bq g^{-1} are presented in Table 96. The ratio of the fragment dose coefficient and the dose coefficient resulting in the highest dose is also presented. The fragment dose coefficients are two to three orders of magnitude smaller than the coefficients resulting in the highest dose.

Table 96 Wound doses

Radionuclide	Fragment dose coefficient (Sv/Bq)	Worst dose coefficient (Sv/Bq)	Ratio fragment dose to worst dose	Minimum dose from 1Bq/g (mSv)	Maximum dose from 1Bq/g (mSv)
H-3	$4.07\ 10^{-13}$	$1.84\ 10^{-11}$	0.022	$8.14\ 10^{-10}$	$3.68\ 10^{-8}$
C-14	$3.19\ 10^{-11}$	$5.77\ 10^{-10}$	0.055	$6.38\ 10^{-8}$	$1.15\ 10^{-6}$
Fe-55	nd	$3.00\ 10^{-9}$	n/a	n/a	$6.00\ 10^{-6}$
Co-60	$2.47\ 10^{-10}$	$1.94\ 10^{-8}$	0.013	$4.94\ 10^{-7}$	$3.88\ 10^{-5}$
Sr-90	$2.87\ 10^{-9}$	$8.81\ 10^{-8}$	0.033	$5.74\ 10^{-6}$	$1.76\ 10^{-4}$
Tc-99	nd	$8.70\ 10^{-10}$	n/a	n/a	$1.74\ 10^{-6}$
Ru-106	$1.65\ 10^{-10}$	$3.02\ 10^{-8}$	0.005	$3.30\ 10^{-7}$	$6.04\ 10^{-5}$
Sb-125	nd	$5.40\ 10^{-9}$	n/a	n/a	$1.08\ 10^{-5}$
I-129	$5.87\ 10^{-9}$	$1.07\ 10^{-7}$	0.055	$1.17\ 10^{-5}$	$2.14\ 10^{-4}$
Cs-134	$1.48\ 10^{-10}$	$1.94\ 10^{-8}$	0.008	$2.96\ 10^{-7}$	$3.88\ 10^{-5}$
Cs-137	$4.77\ 10^{-10}$	$1.40\ 10^{-8}$	0.034	$9.54\ 10^{-7}$	$2.80\ 10^{-5}$
Ce-144	nd	$1.70\ 10^{-7}$	n/a	n/a	$3.40\ 10^{-4}$
Gd-153	nd	$8.60\ 10^{-9}$	n/a	n/a	$1.72\ 10^{-5}$
Pb-210	nd	$3.50\ 10^{-6}$	n/a	n/a	$7.00\ 10^{-3}$
Po-210	$7.72\ 10^{-9}$	$2.40\ 10^{-6}$	0.003	$1.54\ 10^{-5}$	$4.80\ 10^{-3}$
Ra-226	$1.65\ 10^{-7}$	$2.64\ 10^{-6}$	0.063	$3.30\ 10^{-4}$	$5.28\ 10^{-3}$
Ra-228	$1.13\ 10^{-6}$	$4.56\ 10^{-5}$	0.025	$2.26\ 10^{-3}$	$9.12\ 10^{-2}$
Th-228	$8.64\ 10^{-7}$	$1.20\ 10^{-4}$	0.007	$1.73\ 10^{-3}$	$2.40\ 10^{-1}$
Th-230	$1.44\ 10^{-5}$	$4.19\ 10^{-4}$	0.034	$2.88\ 10^{-2}$	$8.38\ 10^{-1}$
Th-232	$1.92\ 10^{-5}$	$4.52\ 10^{-4}$	0.042	$3.84\ 10^{-2}$	$9.04\ 10^{-1}$
U-234	$8.75\ 10^{-8}$	$2.30\ 10^{-6}$	0.038	$1.75\ 10^{-4}$	$4.60\ 10^{-3}$
U-235	$8.13\ 10^{-8}$	$2.11\ 10^{-6}$	0.039	$1.63\ 10^{-4}$	$4.22\ 10^{-3}$
U-238	$7.89\ 10^{-8}$	$2.10\ 10^{-6}$	0.038	$1.58\ 10^{-4}$	$4.20\ 10^{-3}$
Np-237	$7.91\ 10^{-6}$	$2.10\ 10^{-4}$	0.038	$1.58\ 10^{-2}$	$4.20\ 10^{-1}$
Pu-238	$1.41\ 10^{-5}$	$4.50\ 10^{-4}$	0.031	$2.82\ 10^{-2}$	$9.00\ 10^{-1}$
Pu-239	$1.67\ 10^{-5}$	$4.90\ 10^{-4}$	0.034	$3.34\ 10^{-2}$	$9.80\ 10^{-1}$
Pu-240	$1.67\ 10^{-5}$	$4.90\ 10^{-4}$	0.034	$3.34\ 10^{-2}$	$9.80\ 10^{-1}$
Pu-241	$4.10\ 10^{-7}$	$9.68\ 10^{-6}$	0.042	$8.20\ 10^{-4}$	$1.94\ 10^{-2}$
Am-241	$1.41\ 10^{-5}$	$4.00\ 10^{-4}$	0.035	$2.82\ 10^{-2}$	$8.00\ 10^{-1}$
Cm-242	$1.02\ 10^{-7}$	$1.40\ 10^{-5}$	0.007	$2.04\ 10^{-4}$	$2.80\ 10^{-2}$
Cm-244	$5.72\ 10^{-6}$	$2.40\ 10^{-4}$	0.024	$1.14\ 10^{-2}$	$4.80\ 10^{-1}$

713. The highest doses result from thorium, plutonium and americium isotopes. For all radionuclides for which data is available, doses when using the fragment coefficient are less than 1 mSv even if the activity concentration in the waste is increased to 5000 Bq g^{-1}.

E.3.10. Exposure from leachate spillage

714. If leachate is accidentally spilled, for example during leachate transport, then land or a surface water body could become contaminated. Irrespective of the presence of radioactivity, landfill leachate poses a hazard to the environment if spilt and hence any accident involving loss of an entire load would be subject to mitigation measures. It is assumed that if the leachate is accidentally spilled onto land then the land will be remediated appropriately due to the radiological and non-radiological properties of leachate. The remediation process will also involve a dose assessment. Hence this situation is not assessed in the ESC.

715. If the leachate spillage results in contamination of nearby surface water then this is more difficult to remediate. The radiological impact on the public is therefore assessed. It is assumed that farmland adjacent to a water body that becomes contaminated by the spillage also becomes contaminated. Members of the exposed group are assumed to be a farming family who also use the water body for fishing. The leachate spillage pathway is highly uncertain, both in terms of the possibility of occurring and duration. The specific doses presented are illustrative, and might be considered in establishing mitigation measures, but should not be used to determine overall radiological capacities for the landfill site.

716. The dose criterion used for this scenario is the dose constraint for the public, 0.3 mSv y^{-1}.

Potentially exposed group

717. The assessment of doses from a leachate spillage to a water body, e.g. during leachate management work is based on the SNIFFER assessment methodology (SNIFFER, 2006). Members of the exposed group are assumed to be adults and it is assumed that farmland adjacent to the contaminated water body subsequently becomes contaminated through irrigation. The exposure pathways considered are:

- consumption of food produced on land contaminated by a contaminated water body, including fish, milk, green vegetables, root vegetables and meat products;
- external irradiation from radionuclides incorporated in contaminated soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

718. Table 83 details the habit data assumed for the farming family, assumptions concerning drinking water and fish consumption are in Table 97.

Table 97 Habit data for the leachate spillage: applicable during the Period of Authorisation

Pathway*	Age group	Average	97.5th	Comment
Fish consumption (kg y^{-1})	Adult	15	40	From (Smith & Jones, 2003).
	Child	6	20	
	Infant	3.5	15	
Drinking water consumption (m^3 y^{-1})	Adult	0.6	n/a	
	Child	0.35		
	Infant	0.26		

*Other data are the same as presented in Table 116.

E.3.10.1. Estimating activity concentrations after a leachate spillage

719. For this assessment, it is assumed that a tanker load of leachate (28 m^3 of leachate) enters a small reservoir (2 10^6 m^3) that is used for drinking water, irrigation and fishing. The dissolved radionuclide activity concentration, C_{Rn},leachate (Bq m^{-3}) in the leachate is based on the peak leachate activity concentrations (per MBq input to the landfill) from the GoldSim groundwater model during the site management period (Table 79). This is a very conservative set of assumptions.

720. Contamination is assumed to relate to a one-off event with the resulting radioactive contamination remaining constant for one year. The activity concentration ($C_{Rn,water,spill}$; Bq m^{-3}) in the water body is determined as follows:

$$C_{Rn,water,spill} = \frac{C_{Rn,leachate}(t) \cdot V_{spill}}{V_{water}}$$

where:

- $C_{Rn,leachate}$(t) is the activity concentration of radionuclide in the leachate at the time of the spill (Bq m^{-3});
- V_{spill} is the volume of leachate in the spill (m^3); and,
- V_{water} is the volume of the water body (m^3).

721. The resulting doses to the public then arise from water and fish consumption. If the water body is used for irrigation, then a one-off soil activity concentration, $C_{Rn,soil,spill}$ (Bq kg^{-1}), is calculated from:

$$C_{Rn,soil,spill} = C_{Rn,water,spill} \cdot \left\{\frac{Irrig_{rate}}{\rho_{soil} \cdot d_{soil}}\right\}$$

where:

- $Irrig_{rate}$ is the amount of irrigation in 1 year (m);
- ρ_{soil} is the density of the soil (kg m^{-3}); and,

- d_{soil} is the depth of the soil layer (m).

E.3.10.2. Assessment calculations for a farming family after a leachate spillage

Irrigation and Drinking Water

722. The exposure pathways for irrigation are the same as those detailed for groundwater contamination, see Section E.4.3.7. There is however no allowance for daughter radionuclide ingrowth.

723. Consumption of contaminated water by livestock direct from the water body is included at a rate of 0.06 m^3 d^{-1} (SNIFFER, 2006).

Fish Contamination

724. The dose from eating fish taken from the contaminated water body is given by:

$$Dose_{ing,fish} = Q_{fish} \cdot C_{Rn,water,spill} \cdot UF_{Rn,fish} \cdot D_{Rn,ing}$$

where:

- Q_{fish} is the consumption rate of fish (kg y^{-1});
- UF_{fish} is the water to fish transfer factor (m^3 kg^{-1}); and,
- D_{ing} is the dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

725. The transfer factors for freshwater fish are listed in Table 204. These are from (SNIFFER, 2006) except for the Ac-227 value which has been amended from 0.8 used in SNIFFER to a value of 0.24 which is the current IAEA recommendation for Americium (IAEA, 2010). There are no published data for the uptake of Actinium by fish and previous reviews have all adopted freshwater fish values based on Americium for which there are very few data (Smith, et al., 1988).

726. The two pathways resulting in greatest dose from the irrigation and fish pathways are used at critical group consumption rates. The remaining pathways use average consumptions rates.

E.3.10.3. Doses from Leachate spillage

727. It is expected that a spillage of landfill leachate will be subject to mitigation measures with an assessment of any ground contamination at the site. Leachate that enters water resources would become diluted and effective mitigation measures would be less likely. The dose (µSv per MBq) to an adult of a farming family is shown in Table 98. The public dose constraint is 0.3 mSv.

728. The spillage event has a low probability of occurring and clean-up actions would be taken to largely mitigate the event altogether. Without mitigation measures the scenario would constrain the radiological capacity for 21 radionuclides. The results for Ra-226 are independent of the Ra-226 placement depth in the site.

Table 98 Dose to farming family from leachate spillage

Radionuclide	Radiological capacity (MBq)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - infant ($\mu Sv\ MBq^{-1}$)	Limiting dose at radiological capacity ($\mu Sv\ y^{-1}$)
H-3	$6.43\ 10^{9}$	$5.37\ 10^{-10}$	$4.60\ 10^{-10}$	$5.73\ 10^{-10}$	$3.68\ 10^{0}$
C-14	$1.87\ 10^{8}$	$6.56\ 10^{-9}$	$4.54\ 10^{-9}$	$6.79\ 10^{-9}$	$1.27\ 10^{0}$
Cl-36	$1.56\ 10^{8}$	$2.83\ 10^{-8}$	$3.60\ 10^{-8}$	$7.49\ 10^{-8}$	$1.17\ 10^{1}$
Ca-41	$5.77\ 10^{9}$	$7.70\ 10^{-10}$	$1.06\ 10^{-9}$	$7.82\ 10^{-10}$	$6.11\ 10^{0}$
Mn-54	$1.12\ 10^{13}$	$1.65\ 10^{-11}$	$1.59\ 10^{-11}$	$2.68\ 10^{-11}$	$3.00\ 10^{2}$
Fe-55	$1.86\ 10^{13}$	$6.09\ 10^{-12}$	$1.16\ 10^{-11}$	$1.62\ 10^{-11}$	$3.00\ 10^{2}$
Co-60	$3.58\ 10^{11}$	$2.85\ 10^{-10}$	$4.90\ 10^{-10}$	$8.38\ 10^{-10}$	$3.00\ 10^{2}$
Ni-59	$1.95\ 10^{11}$	$4.73\ 10^{-12}$	$4.78\ 10^{-12}$	$9.73\ 10^{-12}$	$1.90\ 10^{0}$
Ni-63	$2.42\ 10^{11}$	$1.12\ 10^{-11}$	$1.21\ 10^{-11}$	$2.39\ 10^{-11}$	$5.77\ 10^{0}$
Zn-65	$8.95\ 10^{11}$	$2.18\ 10^{-10}$	$1.84\ 10^{-10}$	$3.35\ 10^{-10}$	$3.00\ 10^{2}$
Se-79	$8.98\ 10^{8}$	$1.57\ 10^{-9}$	$3.89\ 10^{-9}$	$5.65\ 10^{-9}$	$5.08\ 10^{0}$
Sr-90	$3.83\ 10^{8}$	$9.09\ 10^{-9}$	$1.16\ 10^{-8}$	$1.01\ 10^{-8}$	$4.45\ 10^{0}$
Mo-93	$1.44\ 10^{9}$	$7.27\ 10^{-10}$	$6.20\ 10^{-10}$	$5.88\ 10^{-10}$	$1.04\ 10^{0}$
Zr-93	$3.12\ 10^{11}$	$1.22\ 10^{-10}$	$3.43\ 10^{-11}$	$3.14\ 10^{-11}$	$3.82\ 10^{1}$
Nb-93m	$5.06\ 10^{10}$	$3.50\ 10^{-12}$	$4.19\ 10^{-12}$	$9.85\ 10^{-12}$	$4.98\ 10^{-1}$
Nb-94	$6.09\ 10^{6}$	$5.20\ 10^{-11}$	$5.52\ 10^{-11}$	$1.10\ 10^{-10}$	$6.68\ 10^{-4}$
Tc-99	$6.12\ 10^{8}$	$1.63\ 10^{-8}$	$2.11\ 10^{-8}$	$4.35\ 10^{-8}$	$2.66\ 10^{1}$
Ru-106	$9.14\ 10^{11}$	$1.33\ 10^{-10}$	$1.88\ 10^{-10}$	$3.28\ 10^{-10}$	$3.00\ 10^{2}$
Ag-108m	$2.65\ 10^{8}$	$5.57\ 10^{-11}$	$6.76\ 10^{-11}$	$9.04\ 10^{-11}$	$2.40\ 10^{-2}$
Ag-110m	$6.41\ 10^{12}$	$2.77\ 10^{-11}$	$3.33\ 10^{-11}$	$4.68\ 10^{-11}$	$3.00\ 10^{2}$
Cd-109	$1.04\ 10^{12}$	$1.66\ 10^{-10}$	$1.66\ 10^{-10}$	$2.88\ 10^{-10}$	$3.00\ 10^{2}$
Sb-125	$4.17\ 10^{11}$	$3.39\ 10^{-10}$	$3.77\ 10^{-10}$	$7.19\ 10^{-10}$	$3.00\ 10^{2}$
Sn-119m	$8.43\ 10^{12}$	$1.29\ 10^{-11}$	$1.46\ 10^{-11}$	$3.56\ 10^{-11}$	$3.00\ 10^{2}$
Sn-123	$2.97\ 10^{12}$	$3.55\ 10^{-11}$	$3.97\ 10^{-11}$	$1.01\ 10^{-10}$	$3.00\ 10^{2}$
Sn-126	$4.60\ 10^{6}$	$4.22\ 10^{-10}$	$4.49\ 10^{-10}$	$1.00\ 10^{-9}$	$4.60\ 10^{-3}$
Te-127m	$4.07\ 10^{12}$	$2.55\ 10^{-11}$	$3.14\ 10^{-11}$	$7.38\ 10^{-11}$	$3.00\ 10^{2}$
I-129	$3.01\ 10^{8}$	$1.91\ 10^{-7}$	$2.08\ 10^{-7}$	$1.42\ 10^{-7}$	$6.26\ 10^{1}$
Ba-133	$7.18\ 10^{9}$	$2.09\ 10^{-8}$	$4.18\ 10^{-8}$	$2.94\ 10^{-8}$	$3.00\ 10^{2}$
Cs-134	$1.01\ 10^{11}$	$2.97\ 10^{-9}$	$1.11\ 10^{-9}$	$9.36\ 10^{-10}$	$3.00\ 10^{2}$
Cs-135	$1.55\ 10^{9}$	$4.31\ 10^{-10}$	$1.85\ 10^{-10}$	$1.86\ 10^{-10}$	$6.69\ 10^{-1}$
Cs-137	$9.69\ 10^{8}$	$2.74\ 10^{-9}$	$1.07\ 10^{-9}$	$9.47\ 10^{-10}$	$2.65\ 10^{0}$

Radionuclide	Radiological capacity (MBq)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - infant ($\mu Sv\ MBq^{-1}$)	Limiting dose at radiological capacity ($\mu Sv\ y^{-1}$)
Ce-144	$4.81\ 10^{12}$	$2.31\ 10^{-11}$	$3.04\ 10^{-11}$	$6.23\ 10^{-11}$	$3.00\ 10^{2}$
Pm-147	$2.14\ 10^{13}$	$5.34\ 10^{-12}$	$7.27\ 10^{-12}$	$1.40\ 10^{-11}$	$3.00\ 10^{2}$
Sm-147	$4.81\ 10^{8}$	$6.28\ 10^{-10}$	$5.09\ 10^{-10}$	$6.45\ 10^{-10}$	$3.10\ 10^{-1}$
Sm-151	$7.23\ 10^{11}$	$1.25\ 10^{-12}$	$1.58\ 10^{-12}$	$2.93\ 10^{-12}$	$2.12\ 10^{0}$
Eu-152	$8.05\ 10^{9}$	$6.69\ 10^{-11}$	$7.68\ 10^{-11}$	$1.26\ 10^{-10}$	$1.01\ 10^{0}$
Eu-154	$4.18\ 10^{10}$	$9.25\ 10^{-11}$	$1.17\ 10^{-10}$	$1.98\ 10^{-10}$	$8.29\ 10^{0}$
Eu-155	$8.81\ 10^{12}$	$1.38\ 10^{-11}$	$1.82\ 10^{-11}$	$3.40\ 10^{-11}$	$3.00\ 10^{2}$
Gd-153	$4.83\ 10^{13}$	$2.60\ 10^{-12}$	$3.46\ 10^{-12}$	$6.22\ 10^{-12}$	$3.00\ 10^{2}$
Pb-210	$4.85\ 10^{8}$	$4.19\ 10^{-8}$	$5.30\ 10^{-8}$	$1.02\ 10^{-7}$	$4.94\ 10^{1}$
Po-210	$6.17\ 10^{9}$	$1.83\ 10^{-8}$	$2.38\ 10^{-8}$	$4.86\ 10^{-8}$	$3.00\ 10^{2}$
Ra-226	$3.89\ 10^{6}$	$1.25\ 10^{-8}$	$1.84\ 10^{-8}$	$2.81\ 10^{-8}$	$1.09\ 10^{-1}$
Ra-228	$2.25\ 10^{10}$	$4.28\ 10^{-9}$	$1.33\ 10^{-8}$	$1.27\ 10^{-8}$	$3.00\ 10^{2}$
Ac-227	$3.04\ 10^{9}$	$7.47\ 10^{-8}$	$6.26\ 10^{-8}$	$9.87\ 10^{-8}$	$3.00\ 10^{2}$
Th-228	$1.72\ 10^{11}$	$6.36\ 10^{-10}$	$1.16\ 10^{-9}$	$1.75\ 10^{-9}$	$3.00\ 10^{2}$
Th-229	$2.88\ 10^{7}$	$3.85\ 10^{-9}$	$4.58\ 10^{-9}$	$5.37\ 10^{-9}$	$1.55\ 10^{-1}$
Th-230	$1.98\ 10^{6}$	$1.32\ 10^{-9}$	$9.36\ 10^{-10}$	$9.25\ 10^{-10}$	$2.61\ 10^{-3}$
Th-232	$7.95\ 10^{6}$	$6.68\ 10^{-9}$	$1.80\ 10^{-8}$	$1.64\ 10^{-8}$	$1.43\ 10^{-1}$
Pa-231	$1.36\ 10^{7}$	$3.34\ 10^{-9}$	$2.83\ 10^{-9}$	$2.18\ 10^{-9}$	$4.54\ 10^{-2}$
U-232	$4.04\ 10^{8}$	$2.21\ 10^{-8}$	$3.02\ 10^{-8}$	$3.18\ 10^{-8}$	$1.28\ 10^{1}$
U-233	$1.02\ 10^{8}$	$2.40\ 10^{-9}$	$2.40\ 10^{-9}$	$2.35\ 10^{-9}$	$2.46\ 10^{-1}$
U-234	$1.45\ 10^{8}$	$2.31\ 10^{-9}$	$2.28\ 10^{-9}$	$2.18\ 10^{-9}$	$3.35\ 10^{-1}$
U-235	$6.93\ 10^{7}$	$2.23\ 10^{-9}$	$2.21\ 10^{-9}$	$2.22\ 10^{-9}$	$1.54\ 10^{-1}$
U-236	$1.48\ 10^{9}$	$2.21\ 10^{-9}$	$2.15\ 10^{-9}$	$2.18\ 10^{-9}$	$3.28\ 10^{0}$
U-238	$1.60\ 10^{9}$	$2.28\ 10^{-9}$	$2.32\ 10^{-9}$	$2.43\ 10^{-9}$	$3.90\ 10^{0}$
Np-237	$1.42\ 10^{7}$	$2.97\ 10^{-8}$	$1.96\ 10^{-8}$	$2.06\ 10^{-8}$	$4.21\ 10^{-1}$
Pu-238	$7.56\ 10^{8}$	$2.79\ 10^{-9}$	$1.90\ 10^{-9}$	$1.65\ 10^{-9}$	$2.11\ 10^{0}$
Pu-239	$1.55\ 10^{8}$	$3.06\ 10^{-9}$	$2.16\ 10^{-9}$	$1.75\ 10^{-9}$	$4.75\ 10^{-1}$
Pu-240	$1.89\ 10^{8}$	$3.06\ 10^{-9}$	$2.16\ 10^{-9}$	$1.75\ 10^{-9}$	$5.77\ 10^{-1}$
Pu-241	$9.39\ 10^{9}$	$5.59\ 10^{-11}$	$3.88\ 10^{-11}$	$2.26\ 10^{-11}$	$5.25\ 10^{-1}$
Pu-242	$1.58\ 10^{8}$	$2.93\ 10^{-9}$	$2.08\ 10^{-9}$	$1.66\ 10^{-9}$	$4.65\ 10^{-1}$
Pu-244	$1.26\ 10^{8}$	$2.95\ 10^{-9}$	$2.09\ 10^{-9}$	$1.74\ 10^{-9}$	$3.71\ 10^{-1}$
Am-241	$3.03\ 10^{8}$	$9.15\ 10^{-10}$	$6.25\ 10^{-10}$	$6.09\ 10^{-10}$	$2.77\ 10^{-1}$
Am-242m	$1.75\ 10^{7}$	$1.10\ 10^{-9}$	$7.51\ 10^{-10}$	$7.12\ 10^{-10}$	$1.92\ 10^{-2}$
Am-243	$1.46\ 10^{8}$	$9.20\ 10^{-10}$	$6.31\ 10^{-10}$	$6.19\ 10^{-10}$	$1.34\ 10^{-1}$
Cm-242	$1.48\ 10^{11}$	$3.75\ 10^{-11}$	$3.85\ 10^{-11}$	$8.87\ 10^{-11}$	$1.31\ 10^{1}$
Cm-243	$4.89\ 10^{7}$	$1.71\ 10^{-9}$	$9.37\ 10^{-10}$	$1.40\ 10^{-9}$	$8.36\ 10^{-2}$

Radionuclide	Radiological capacity (MBq)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose to Farming family - infant ($\mu Sv\ MBq^{-1}$)	Limiting dose at radiological capacity ($\mu Sv\ y^{-1}$)
Cm-244	$1.16\ 10^{8}$	$1.34\ 10^{-9}$	$8.05\ 10^{-10}$	$1.21\ 10^{-9}$	$1.56\ 10^{-1}$
Cm-245	$1.26\ 10^{7}$	$2.44\ 10^{-9}$	$1.37\ 10^{-9}$	$1.61\ 10^{-9}$	$3.07\ 10^{-2}$
Cm-246	$1.27\ 10^{7}$	$2.44\ 10^{-9}$	$1.31\ 10^{-9}$	$1.61\ 10^{-9}$	$3.10\ 10^{-2}$
Cm-248	$1.45\ 10^{7}$	$8.95\ 10^{-9}$	$5.01\ 10^{-9}$	$6.08\ 10^{-9}$	$1.30\ 10^{-1}$

E.4. Radiological impacts after the period of authorisation {R6}

729. As described in Section E.2, the ESC considers the exposure of adults, children and infants. Hence the ESC calculates the radiological capacity of Port Clarence based on the greatest radiological impact to one of three age groups.

730. During the post closure period the site will be actively managed and monitored whilst the Permit is in force. The active management phase ends when the site has stabilised to the extent that active management is no longer necessary and a Permit is no longer relevant (the end of the period of authorisation). The process leading to the end of the period of authorisation will be gradual with a progressive decrease in monitoring and controls as appropriate and where agreed with the Environment Agency.

731. Under the planning permission requirements the Port Clarence site must be restored to areas of grassland, scrub and woodland and the surrounding areas will be restored to areas of open water, aquatic marginal vegetation, scrub, wet meadow and ruderal grassland with small hollows, banks and ridges suitable for nature conservation use. The restoration is carried out progressively during the life of the site and will be completed by 2070 or earlier. The aftercare of the restored site also continues under the planning requirements for at least 10 years after closure to ensure that the land use and vegetation is properly established.

732. At some point in time after site restoration is complete members of the public will have access to this land for its intended recreational use. This may or may not occur before the end of the period of authorisation. The principal risk to site users could arise from direct radiation from the disposed waste and gas migration. The exposed groups considered for this scenario are recreational site users.

733. During the post closure period gradual degradation of the non-mineral components of the site cap and liner may occur, eventually leading to infiltration of rainwater into the landfill site, leaching of the waste and migration of radionuclides in the groundwater below the site. The characteristics of the site cap and engineered barriers mean that contamination of the groundwater is not expected to occur before the end of the period of authorisation, and probably not until sometime afterwards.

734. The groundwater downstream of the site is subject to saline intrusion, is not potable and will not be used for drinking, irrigation or livestock. Nevertheless, a what-if calculation is performed that considers members of the public drinking groundwater abstracted from a well and using it for irrigation of land. This assessment is not used to calculate the radiological capacity of the site.

735. Contaminated groundwater will migrate downstream of the site, along the aquifer, and enter the estuary. The exposure of a member of the public through ingestion of seafood, external irradiation on the bank and inhalation of seaspray is considered.

736. The future possible erosion of the site is also considered. Two assessments are performed: one to a person walking close to the eroded material and one to a member of the public exposed to contamination leached from eroded waste.

737. The assessment scenarios for the period following the period of authorisation are summarised in the table below.

Table 99 Summary of scenarios and exposure pathways after the period of authorisation

Event/scenario	Exposure pathway	Description
Access to undisturbed site: recreational use	External irradiation	A member of the public is exposed to external radiation whilst walking over the undisturbed site.
	Gas (including radon) inhalation	A member of the public is exposed to gases emanating from contaminated material in the landfill.
Site erosion: recreational use	External irradiation	A member of the public is exposed to external radiation whilst walking close to the estuary.
	Inadvertent ingestion	A member of the public inadvertently ingests contaminated soil whilst walking close to the estuary.
	Inhalation	A member of the public inhales contaminated soil whilst walking close to the estuary.
Site erosion: fishing family/estuary user	Ingestion of contaminated seafood	A member of the public ingests contaminated seafood obtained from the estuary.
	External irradiation	A member of the public is exposed to external radiation during activities near the estuary. This includes fishing for the adult member of the public.
	Inhalation of spray	A member of the public is exposed to radionuclides incorporated in spray from the estuary.
Bathtubbing: residential occupant	Land contaminated with leachate overspill	A member of the public ingests contaminated foodstuffs as a result of growing crops on contaminated soil, inadvertently ingests or inhales contaminated soil and is exposed through external irradiation to soil.

Event/scenario	Exposure pathway	Description
Release to groundwater: abstraction 100m from site boundary	Ingestion of contaminated water	Drinking water contaminated as a result of radionuclide migration into the aquifer and abstracted from a well.
	Irrigation of land with contaminated groundwater	A member of the public ingests contaminated foodstuffs as a result of growing crops on contaminated soil, inadvertently ingests or inhales contaminated soil and is exposed through external irradiation to soil.
Release to groundwater: migration into estuary	Ingestion of contaminated seafood	A member of the public ingests contaminated seafood obtained from the estuary.
	External irradiation	A member of the public is exposed to external radiation during activities near the estuary. This includes fishing for the adult member of the public.
	Inhalation of seaspray	A member of the public is exposed to radionuclides incorporated in spray from the estuary.

738. Intrusion scenarios are considered separately in Section E.5, exposure to heterogeneous wastes is addressed in Section E.6, and the assessment of wildlife exposure is discussed in Section E.7. Additional scenarios which were considered, but not explicitly assessed, are discussed in the following sections.

Inundation from the sea

739. The effects of very long term climate change are assessed due to the location of the site close to the Tees Estuary. Consideration has been given to the timescale over which sea level rise could occur (see Section 2.9) and lead to the erosion of the site. Future glaciation would have similar or lesser effects than the “residential intrusion scenario” considered in Section E.5.6 since it could also remove the cap but it would occur much later (e.g. 10,000s of years in the future).

740. With sea level rise the area surrounding the landfill is likely in due course to be subject to periodic flooding. At some stage the peak flood height will begin to overlap the basal liner and water may enter the base of the landfill. The bathtubbing and the groundwater scenarios are both assumed to occur earlier and would have similar or greater effects than inundation. Hence, inundation is not explicitly assessed.

Enhanced rainfall due to climate change

741. The HRA (MJCA, 2019b) does not explicitly consider long term effective rainfall changes in response to climate change. Notwithstanding that there may be changes to rates of effective rainfall the assumptions made in the HRA with regard to cap infiltration are generally conservative. An infiltration to grassland value of 74.3 mm y^{-1} has been considered based on long term effective rainfall, and an infiltration to grassland that is equal to the effective rainfall is considered representative of the long term situation where the geomembrane element of the cap has fully degraded. The landfill capping system incorporates a GCL and it is likely that the long term infiltration

through the capping system will be less than the effective rainfall because the GCL will be less susceptible to chemical or physical degradation than a LLDPE geomembrane.

742. The HRA does not consider a potential increased risk of flooding resulting from climate change. A flood risk assessment (FRA) for the area is presented in Figure 10. A loss of a significant depth of cover materials through rainfall induced erosion is unlikely where a restored landform is vegetated and subject to periodic inspections and maintenance activities as appropriate. The site is located at the topographic high in the catchment and therefore is not subject to uncontrolled runoff from adjacent areas.

Seismic Events

743. The engineered containment structures at the site are not formed of brittle materials such as concrete that may fracture as a result of a severe earthquake. The HDPE and clay lining materials have a high shear strength and have the flexibility to withstand the stresses which would be imposed during the types of earthquake which occur in the UK. Two fracking licences have been granted in the immediate vicinity of Port Clarence. These are administered by Egdon Resources U.K. Limited and Third Energy UK Gas Limited (Licence references PEDL68 and PEDL259, respectively). We are not aware of any test drilling or site developments being carried out or planned under these licences. The engineered containment structures at the site would be resilient to any minor seismic activity which might result from future fracking. Hence this scenario is not considered in the ESC.

Transport Accidents

744. Transport accidents occurring prior to delivery are not discussed in the ESC because transport is outside of the scope of the permit and is regulated under an existing regime of Dangerous Goods Regulations. Transport accidents on the site are considered as part of the dropped load scenario (see Section 684) and a transport accident involving leachate sent to a aqueous waste treatment facility is specifically considered (see Section E.3.9).

Criticality Event

745. Criticality and heat generation are processes that are mentioned in the guidance (NS-GRA para. 6.4.21 and 7.3.31). Criticality conditions can arise when sufficient fissile materials are present and are arranged in idealised configurations. Criticality results in the release of increased radiation over the period that criticality is sustained. Criticality is not realistically considered feasible for LLW facilities under practical circumstances and has not been observed in such facilities.

746. An analysis presented in 2009 (Augean, 2009) for ENRMF showed that criticality is not an issue given the very low content of fissile material and low activity concentrations proposed for the waste to be disposed of. The analysis referred to studies on the issue of criticality control which indicated that if the % by weight of U235 in U238 is maintained on average over the wasteform at less than 1 % by weight then the criticality potential in repositories is eliminated under long term future scenarios. The natural level in Uranium is 0.7% by weight U235.

747. Criticality conditions during transport, handling and site operations are prevented effectively through adherence to radioactive materials transport package limits for fissile materials (ref IAEA SSR6 2018).

748. Consignments requiring marking as ‘fissile’ under the transport regulations (ref IAEA SSR6 2018) would not be accepted for disposal at the Port Clarence landfills without prior agreement with the Environment Agency. This condition limits the U235, U233 and Pu content of the wastes and/or the mass of fissile material in each package and each consignment. Since the LLW that will be sent for disposal at Port Clarence is broadly homogenous decommissioning wastes, the most relevant provisions excepting material and packages from classification as ‘fissile’ are given in SSR 6 para 417 (a, c, d). These are reproduced in Table 100.

Table 100 Provisions whereby material and packages are excepted from classification as fissile

SSR 6 Para	Requirement
417 (a)	Uranium enriched in uranium-235 to a maximum of 1% by mass, and with a total plutonium and uranium-233 content not exceeding 1% of the mass of uranium-235, provided that the fissile nuclides are distributed essentially homogeneously throughout the material. In addition, if uranium-235 is present in metallic, oxide or carbide forms, it shall not form a lattice arrangement
417 (c)	Uranium with a maximum uranium enrichment of 5% by mass of uranium-235 provided: (i) There is no more than 3.5 g of uranium-235 per package. (ii) The total plutonium and uranium-233 content does not exceed 1% of the mass of uranium-235 per package. (iii) Transport of the package is subject to the consignment limit provided in para. 570(c).
417 (d)	Fissile nuclides with a total mass not greater than 2.0 g per package, provided the package is transported subject to the consignment limit provided in para. 570(d).
570 (c)	45 g per consignment
570 (d)	15 g per consignment

749. Criticality conditions in the emplaced waste at the Port Clarence landfills will be effectively prevented because insufficient material is present to cause criticality. The material will be distributed within the landfills in a large number of waste packages at low concentrations and in a broadly homogeneous form.

750. Heat generating wastes would not be accepted for disposal since these would not be classified as LLW.

E.4.1. Presentation of results of dose assessments

751. The radiological capacity for individual radionuclides present in the LLW is obtained from the results of the ESC and depends on the radiological characteristics of the radionuclide. The radiological capacity is calculated on the basis that the LLW only contains this one radionuclide. The overall radiological capacity for an individual

radionuclide is the minimum of the radionuclide capacities calculated for each of the different scenarios. The results of the assessment are presented as effective doses per MBq disposed (µSv y^{-1} MBq^{-1}). The dose calculated for each radionuclide would only be achieved if that radionuclide was the only one disposed of. Actual waste disposal will be controlled using a sum of fractions approach (see paragraph 0).

752. Estimates of radiological impact based on 'illustrative inventories' for waste streams that might be typical of those contributing to the total impact from disposals at the facility have been produced. These estimates are presented in Appendix D.

E.4.2. Exposure of the public on the undisturbed site

753. Radiation exposure of members of the public spending time on the site after the end of the period of authorisation could occur. Two exposure pathways are considered; exposure through inhalation of gases (H-3, C-14 and radon) and direct irradiation to a casual user who walks over the restored site (e.g. on a footpath). The possibility of housing being built on the site after the period of authorisation is considered in the assessment of intrusion scenarios (Sections E.5.6 and E.5.7).

754. The assessment assumes that the waste is shielded by a 1.3 m thick capping layer and a further layer of cover material to a depth of 1 m. This scenario also covers occupancy of agricultural land by farmers since activities such as ploughing will not disturb the waste.

755. The dose criterion used is a dose of 0.02 mSv y^{-1} for the public (this is equivalent to the risk guidance level of 10^{-6} y^{-1} for exposure of the public post closure, for situations that are expected to occur).

Potentially exposed group

756. The restored site will include rough grassland, scrub and woodland and the surrounding areas will be restored to areas of open water, aquatic marginal vegetation, scrub, wet meadow and ruderal grassland with small hollows, banks and ridges suitable for nature conservation use, that will be available for access by the public. The area could be used for walking and this scenario considers an occupancy of 750 hours per year on the site for all age groups, equivalent to about 2 hours per day (Oatway & Mobbs, 2003).

757. This occupancy applies to exposure to release of gases through the intact cap and direct exposure whilst using the restored site for recreational purposes. Table 101 details the habit data assumed for the exposed group. Exposure is assessed both immediately after site restoration (in 2130) and 60 years after closure. The time on site at 60 years for the child and infant is adjusted to 1400 and 788 hr y^{-1}.

Table 101 Habit data for exposure to gas releases: applicable after the Period of Authorisation

Parameter	Value	Comment
Inhalation rate – adult ($m^3\ h^{-1}$)	1.0	
Inhalation rate – child ($m^3\ h^{-1}$)	0.64	
Inhalation rate – infant ($m^3\ h^{-1}$)	0.22	
Time on site – public ($h\ y^{-1}$)	750	About 2 hours per day (all ages)

E.4.2.1. Assessment calculations for recreational site use

758. The impact on a member of the public using the site for recreation has been included to illustrate the doses expected from what is likely to be the most probable public use of the site after the period of institutional control.

759. It is expected that the public will get access to the site soon after site restoration is complete. The doses are therefore assessed both at site closure and after 60 years (at the end of the period of authorisation).

Gas generation

760. The method in Section E.3.5 is used to assess the impact of gas generation for recreational site users. The release rate of gases from a landfill is expected to vary over time. A conservative assumption for the operational period assumed all C-14 and H-3 that was associated with organic material would be released over a ten year period. Gas generation within the ENRMF landfill has been simulated using the GasSim model (Augean, 2010) which shows a rapid build-up in the rate of release after capping followed by an exponential decline. The waste cells are capped sequentially so a series of peaks during the operational period could be expected. A longer timescale for gas generation (20 years) has been applied to the period after closure using the value recommended by IAEA (IAEA, 2003).

761. The exposure time (Table 101) is reduced to reflect recreational use, assumed to be about 2 hours per day (equivalent to an outdoor occupancy factor of 0.0856 based on 750 hours per year).

External Irradiation

762. The external irradiation calculation is presented in Section E.4.3.7 was used by setting indoor occupation to zero and using the same outdoor occupancy factor.

E.4.2.2. Dose to recreational user from exposure to gas release and external radiation

763. The dose to a recreational user immediately after the site closes and at the end of the period of authorisation (60 years after closure) are given in Table 102 and Table 77, respectively. Note that the results after 60 years include the effects of ingrowth upon the calculated doses. The expected dose from the radiological capacity is also shown.

The highest dose at site closure is from H-3 and C-14 gas and Nb-93m and this scenario limits the radionuclide capacity for these three radionuclides. The dose from wastes disposed of at Port Clarence will always be lower than 20 µSv y^{-1} due to application of the sum of fractions approach.

764. The dose was also assessed assuming that waste at the maximum activity concentration given in Table 32 was disposed at the top layer of the landfill (at 2.3 m below the restored surface for all radionuclides). Under these circumstances all doses are less than 1 µSv. Hence, no additional restrictions on the activity concentration in the waste are required.

Table 102 Doses to recreational users of restored site at site closure

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose from radiological capacity (µSv y^{-1})
		Gas	External	Total	
H-3	$6.43\ 10^{9}$	$3.11\ 10^{-9}$	0	$3.11\ 10^{-9}$	$2.00\ 10^{1}$
C-14	$1.87\ 10^{8}$	$1.07\ 10^{-7}$	$7.41\ 10^{-67}$	$1.07\ 10^{-7}$	$2.00\ 10^{1}$
Cl-36	$1.56\ 10^{8}$	0	$3.80\ 10^{-29}$	$3.80\ 10^{-29}$	$5.94\ 10^{-21}$
Ca-41	$5.77\ 10^{9}$	0	0	0	0
Mn-54	$1.12\ 10^{13}$	0	$5.35\ 10^{-19}$	$5.35\ 10^{-19}$	$5.99\ 10^{-6}$
Fe-55	$1.86\ 10^{13}$	0	0	0	0
Co-60	$3.58\ 10^{11}$	0	$6.04\ 10^{-17}$	$6.04\ 10^{-17}$	$2.16\ 10^{-5}$
Ni-59	$1.95\ 10^{11}$	0	0	0	0
Ni-63	$2.42\ 10^{11}$	0	0	0	0
Zn-65	$8.95\ 10^{11}$	0	$3.81\ 10^{-18}$	$3.81\ 10^{-18}$	$3.41\ 10^{-6}$
Se-79	$8.98\ 10^{8}$	0	$8.49\ 10^{-65}$	$8.49\ 10^{-65}$	$7.63\ 10^{-56}$
Sr-90	$3.83\ 10^{8}$	0	$3.22\ 10^{-26}$	$3.22\ 10^{-26}$	$1.23\ 10^{-17}$
Mo-93	$1.44\ 10^{9}$	0	$2.24\ 10^{-9}$	$2.24\ 10^{-9}$	$3.23\ 10^{0}$
Zr-93	$3.12\ 10^{11}$	0	0	0	0
Nb-93m	$5.06\ 10^{10}$	0	$3.95\ 10^{-10}$	$3.95\ 10^{-10}$	$2.00\ 10^{1}$
Nb-94	$6.09\ 10^{6}$	0	$5.54\ 10^{-19}$	$5.54\ 10^{-19}$	$3.38\ 10^{-12}$
Tc-99	$6.12\ 10^{8}$	0	$1.64\ 10^{-48}$	$1.64\ 10^{-48}$	$1.01\ 10^{-39}$
Ru-106	$9.14\ 10^{11}$	0	$9.80\ 10^{-21}$	$9.80\ 10^{-21}$	$8.96\ 10^{-9}$
Ag-108m	$2.65\ 10^{8}$	0	$4.52\ 10^{-20}$	$4.52\ 10^{-20}$	$1.20\ 10^{-11}$
Ag-110m	$6.41\ 10^{12}$	0	$3.82\ 10^{-18}$	$3.82\ 10^{-18}$	$2.45\ 10^{-5}$
Cd-109	$1.04\ 10^{12}$	0	$3.41\ 10^{-49}$	$3.41\ 10^{-49}$	$3.54\ 10^{-37}$
Sb-125	$4.17\ 10^{11}$	0	$3.84\ 10^{-21}$	$3.84\ 10^{-21}$	$1.60\ 10^{-9}$
Sn-119m	$8.43\ 10^{12}$	0	0	0	0
Sn-123	$2.97\ 10^{12}$	0	$1.06\ 10^{-20}$	$1.06\ 10^{-20}$	$3.14\ 10^{-8}$

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose from radiological capacity ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
Sn-126	$4.60\ 10^{6}$	0	$1.26\ 10^{-19}$	$1.26\ 10^{-19}$	$5.80\ 10^{-13}$
Te-127m	$4.07\ 10^{12}$	0	$6.76\ 10^{-39}$	$6.76\ 10^{-39}$	$2.75\ 10^{-26}$
I-129	$3.01\ 10^{8}$	0	$1.04\ 10^{-138}$	$1.04\ 10^{-138}$	$3.14\ 10^{-130}$
Ba-133	$7.18\ 10^{9}$	0	$9.28\ 10^{-24}$	$9.28\ 10^{-24}$	$6.66\ 10^{-14}$
Cs-134	$1.01\ 10^{11}$	0	$2.21\ 10^{-19}$	$2.21\ 10^{-19}$	$2.23\ 10^{-8}$
Cs-135	$1.55\ 10^{9}$	0	$4.67\ 10^{-57}$	$4.67\ 10^{-57}$	$7.25\ 10^{-48}$
Cs-137	$9.69\ 10^{8}$	0	$4.47\ 10^{-20}$	$4.47\ 10^{-20}$	$4.33\ 10^{-11}$
Ce-144	$4.81\ 10^{12}$	0	$5.50\ 10^{-35}$	$5.50\ 10^{-35}$	$2.65\ 10^{-22}$
Pm-147	$2.14\ 10^{13}$	0	$1.12\ 10^{-47}$	$1.12\ 10^{-47}$	$2.39\ 10^{-34}$
Sm-147	$4.81\ 10^{8}$	0	0	0	0
Sm-151	$7.23\ 10^{11}$	0	0	0	0
Eu-152	$8.05\ 10^{9}$	0	$2.49\ 10^{-18}$	$2.49\ 10^{-18}$	$2.00\ 10^{-8}$
Eu-154	$4.18\ 10^{10}$	0	$3.46\ 10^{-18}$	$3.46\ 10^{-18}$	$1.45\ 10^{-7}$
Eu-155	$8.81\ 10^{12}$	0	$1.58\ 10^{-40}$	$1.58\ 10^{-40}$	$1.39\ 10^{-27}$
Gd-153	$4.83\ 10^{13}$	0	$2.12\ 10^{-42}$	$2.12\ 10^{-42}$	$1.02\ 10^{-28}$
Pb-210	$4.85\ 10^{8}$	0	$3.30\ 10^{-22}$	$3.30\ 10^{-22}$	$1.60\ 10^{-13}$
Po-210	$6.17\ 10^{9}$	0	$2.82\ 10^{-24}$	$2.82\ 10^{-24}$	$1.74\ 10^{-14}$
Ra-226	$3.89\ 10^{6}$	$5.36\ 10^{-11}$	$6.46\ 10^{-17}$	$5.36\ 10^{-11}$	$2.08\ 10^{-4}$
Ra-228	$2.25\ 10^{10}$	0	$3.09\ 10^{-15}$	$3.09\ 10^{-15}$	$6.96\ 10^{-5}$
Ac-227	$3.04\ 10^{9}$	0	$1.21\ 10^{-20}$	$1.21\ 10^{-20}$	$3.67\ 10^{-11}$
Th-228	$1.72\ 10^{11}$	0	$1.95\ 10^{-15}$	$1.95\ 10^{-15}$	$3.35\ 10^{-4}$
Th-229	$2.88\ 10^{7}$	0	$5.16\ 10^{-18}$	$5.16\ 10^{-18}$	$1.49\ 10^{-10}$
Th-230	$1.98\ 10^{6}$	0	$2.56\ 10^{-38}$	$2.56\ 10^{-38}$	$5.07\ 10^{-32}$
Th-232	$7.95\ 10^{6}$	0	$1.16\ 10^{-15}$	$1.16\ 10^{-15}$	$9.18\ 10^{-9}$
Pa-231	$1.36\ 10^{7}$	0	$5.85\ 10^{-26}$	$5.85\ 10^{-26}$	$7.94\ 10^{-19}$
U-232	$4.04\ 10^{8}$	0	$1.73\ 10^{-19}$	$1.73\ 10^{-19}$	$6.97\ 10^{-11}$
U-233	$1.02\ 10^{8}$	0	$6.63\ 10^{-32}$	$6.63\ 10^{-32}$	$6.79\ 10^{-24}$
U-234	$1.45\ 10^{8}$	0	$2.63\ 10^{-45}$	$2.63\ 10^{-45}$	$3.81\ 10^{-37}$
U-235	$6.93\ 10^{7}$	0	$1.83\ 10^{-29}$	$1.83\ 10^{-29}$	$1.27\ 10^{-21}$
U-236	$1.48\ 10^{9}$	0	$2.06\ 10^{-41}$	$2.06\ 10^{-41}$	$3.06\ 10^{-32}$
U-238	$1.60\ 10^{9}$	0	$9.52\ 10^{-24}$	$9.52\ 10^{-24}$	$1.53\ 10^{-14}$
Np-237	$1.42\ 10^{7}$	0	$1.41\ 10^{-26}$	$1.41\ 10^{-26}$	$2.00\ 10^{-19}$
Pu-238	$7.56\ 10^{8}$	0	$3.01\ 10^{-47}$	$3.01\ 10^{-47}$	$2.28\ 10^{-38}$
Pu-239	$1.55\ 10^{8}$	0	$1.86\ 10^{-31}$	$1.86\ 10^{-31}$	$2.89\ 10^{-23}$
Pu-240	$1.89\ 10^{8}$	0	$2.28\ 10^{-54}$	$2.28\ 10^{-54}$	$4.31\ 10^{-46}$

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose from radiological capacity ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
Pu-241	$9.39\ 10^{9}$	0	$6.35\ 10^{-41}$	$6.35\ 10^{-41}$	$5.96\ 10^{-31}$
Pu-242	$1.58\ 10^{8}$	0	$8.21\ 10^{-66}$	$8.21\ 10^{-66}$	$1.30\ 10^{-57}$
Pu-244	$1.26\ 10^{8}$	0	$2.28\ 10^{-24}$	$2.28\ 10^{-24}$	$2.87\ 10^{-16}$
Am-241	$3.03\ 10^{8}$	0	$4.23\ 10^{-61}$	$4.23\ 10^{-61}$	$1.28\ 10^{-52}$
Am-242m	$1.75\ 10^{7}$	0	$5.98\ 10^{-22}$	$5.98\ 10^{-22}$	$1.04\ 10^{-14}$
Am-243	$1.46\ 10^{8}$	0	$7.32\ 10^{-30}$	$7.32\ 10^{-30}$	$1.07\ 10^{-21}$
Cm-242	$1.48\ 10^{11}$	0	$1.78\ 10^{-41}$	$1.78\ 10^{-41}$	$2.64\ 10^{-30}$
Cm-243	$4.89\ 10^{7}$	0	$2.72\ 10^{-29}$	$2.72\ 10^{-29}$	$1.33\ 10^{-21}$
Cm-244	$1.16\ 10^{8}$	0	0	0	0
Cm-245	$1.26\ 10^{7}$	0	$1.57\ 10^{-34}$	$1.57\ 10^{-34}$	$1.97\ 10^{-27}$
Cm-246	$1.27\ 10^{7}$	0	$1.35\ 10^{-138}$	$1.35\ 10^{-138}$	$1.71\ 10^{-131}$
Cm-248	$1.45\ 10^{7}$	0	0	0	0

Table 103 Doses to recreational users of restored site 60 years after closure

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose from radiological capacity ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
H-3	$6.43\ 10^{9}$	$7.43\ 10^{-11}$	0	$7.43\ 10^{-11}$	$4.78\ 10^{-1}$
C-14	$1.87\ 10^{8}$	$3.23\ 10^{-8}$	$1.37\ 10^{-66}$	$3.23\ 10^{-8}$	$6.05\ 10^{0}$
Cl-36	$1.56\ 10^{8}$	0	$7.09\ 10^{-29}$	$7.09\ 10^{-29}$	$1.11\ 10^{-20}$
Ca-41	$5.77\ 10^{9}$	0	0	0	0
Mn-54	$1.12\ 10^{13}$	0	$7.38\ 10^{-40}$	$7.38\ 10^{-40}$	$8.27\ 10^{-27}$
Fe-55	$1.86\ 10^{13}$	0	0	0	0
Co-60	$3.58\ 10^{11}$	0	$4.22\ 10^{-20}$	$4.22\ 10^{-20}$	$1.51\ 10^{-8}$
Ni-59	$1.95\ 10^{11}$	0	0	0	0
Ni-63	$2.42\ 10^{11}$	0	0	0	0
Zn-65	$8.95\ 10^{11}$	0	$6.42\ 10^{-45}$	$6.42\ 10^{-45}$	$5.74\ 10^{-33}$
Se-79	$8.98\ 10^{8}$	0	$1.58\ 10^{-64}$	$1.58\ 10^{-64}$	$1.42\ 10^{-55}$
Sr-90	$3.83\ 10^{8}$	0	$1.42\ 10^{-26}$	$1.42\ 10^{-26}$	$5.43\ 10^{-18}$
Mo-93	$1.44\ 10^{9}$	0	$4.14\ 10^{-9}$	$4.14\ 10^{-9}$	$5.96\ 10^{0}$
Zr-93	$3.12\ 10^{11}$	0	0	0	0
Nb-93m	$5.06\ 10^{10}$	0	$5.60\ 10^{-11}$	$5.60\ 10^{-11}$	$2.83\ 10^{0}$
Nb-94	$6.09\ 10^{6}$	0	$1.03\ 10^{-18}$	$1.03\ 10^{-18}$	$6.29\ 10^{-12}$
Tc-99	$6.12\ 10^{8}$	0	$3.07\ 10^{-48}$	$3.07\ 10^{-48}$	$1.88\ 10^{-39}$
Ru-106	$9.14\ 10^{11}$	0	$4.04\ 10^{-38}$	$4.04\ 10^{-38}$	$3.69\ 10^{-26}$
Ag-108m	$2.65\ 10^{8}$	0	$7.64\ 10^{-20}$	$7.64\ 10^{-20}$	$2.03\ 10^{-11}$
Ag-110m	$6.41\ 10^{12}$	0	$2.80\ 10^{-44}$	$2.80\ 10^{-44}$	$1.80\ 10^{-31}$
Cd-109	$1.04\ 10^{12}$	0	$3.18\ 10^{-63}$	$3.18\ 10^{-63}$	$3.31\ 10^{-51}$
Sb-125	$4.17\ 10^{11}$	0	$2.04\ 10^{-27}$	$2.04\ 10^{-27}$	$8.50\ 10^{-16}$
Sn-119m	$8.43\ 10^{12}$	0	0	0	0
Sn-123	$2.97\ 10^{12}$	0	$1.72\ 10^{-71}$	$1.72\ 10^{-71}$	$5.09\ 10^{-59}$
Sn-126	$4.60\ 10^{6}$	0	$2.36\ 10^{-19}$	$2.36\ 10^{-19}$	$1.08\ 10^{-12}$
Te-127m	$4.07\ 10^{12}$	0	$8.38\ 10^{-101}$	$8.38\ 10^{-101}$	$3.41\ 10^{-88}$
I-129	$3.01\ 10^{8}$	0	$1.95\ 10^{-138}$	$1.95\ 10^{-138}$	$5.86\ 10^{-130}$
Ba-133	$7.18\ 10^{9}$	0	$3.32\ 10^{-25}$	$3.32\ 10^{-25}$	$2.39\ 10^{-15}$
Cs-134	$1.01\ 10^{11}$	0	$7.39\ 10^{-28}$	$7.39\ 10^{-28}$	$7.47\ 10^{-17}$
Cs-135	$1.55\ 10^{9}$	0	$8.72\ 10^{-57}$	$8.72\ 10^{-57}$	$1.35\ 10^{-47}$
Cs-137	$9.69\ 10^{8}$	0	$2.10\ 10^{-20}$	$2.10\ 10^{-20}$	$2.04\ 10^{-11}$
Ce-144	$4.81\ 10^{12}$	0	$7.19\ 10^{-58}$	$7.19\ 10^{-58}$	$3.46\ 10^{-45}$
Pm-147	$2.14\ 10^{13}$	0	$2.71\ 10^{-54}$	$2.71\ 10^{-54}$	$5.80\ 10^{-41}$

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose from radiological capacity (µSv y^{-1})
		Gas	External	Total	
Sm-147	4.81 10^{8}	0	0	0	0
Sm-151	7.23 10^{11}	0	0	0	0
Eu-152	8.05 10^{9}	0	2.15 10^{-19}	2.15 10^{-19}	1.73 10^{-9}
Eu-154	4.18 10^{10}	0	5.11 10^{-20}	5.11 10^{-20}	2.14 10^{-9}
Eu-155	8.81 10^{12}	0	4.75 10^{-44}	4.75 10^{-44}	4.18 10^{-31}
Gd-153	4.83 10^{13}	0	1.43 10^{-69}	1.43 10^{-69}	6.88 10^{-56}
Pb-210	4.85 10^{8}	0	9.45 10^{-23}	9.45 10^{-23}	4.58 10^{-14}
Po-210	6.17 10^{9}	0	1.13 10^{-71}	1.13 10^{-71}	6.97 10^{-62}
Ra-226	3.89 10^{6}	7.56 10^{-11}	1.17 10^{-16}	7.56 10^{-11}	2.94 10^{-4}
Ra-228	2.25 10^{10}	0	4.17 10^{-18}	4.17 10^{-18}	9.39 10^{-8}
Ac-227	3.04 10^{9}	0	3.34 10^{-21}	3.34 10^{-21}	1.01 10^{-11}
Th-228	1.72 10^{11}	0	1.30 10^{-24}	1.30 10^{-24}	2.24 10^{-13}
Th-229	2.88 10^{7}	0	9.58 10^{-18}	9.58 10^{-18}	2.76 10^{-10}
Th-230	1.98 10^{6}	0	3.09 10^{-18}	3.09 10^{-18}	6.11 10^{-12}
Th-232	7.95 10^{6}	0	2.16 10^{-15}	2.16 10^{-15}	1.71 10^{-8}
Pa-231	1.36 10^{7}	0	1.91 10^{-20}	1.91 10^{-20}	2.60 10^{-13}
U-232	4.04 10^{8}	0	1.76 10^{-19}	1.76 10^{-19}	7.11 10^{-11}
U-233	1.02 10^{8}	0	5.44 10^{-20}	5.44 10^{-20}	5.57 10^{-12}
U-234	1.45 10^{8}	0	2.58 10^{-41}	2.58 10^{-41}	3.75 10^{-33}
U-235	6.93 10^{7}	0	1.73 10^{-28}	1.73 10^{-28}	1.20 10^{-20}
U-236	1.48 10^{9}	0	6.40 10^{-24}	6.40 10^{-24}	9.49 10^{-15}
U-238	1.60 10^{9}	0	1.78 10^{-23}	1.78 10^{-23}	2.85 10^{-14}
Np-237	1.42 10^{7}	0	2.64 10^{-26}	2.64 10^{-26}	3.74 10^{-19}
Pu-238	7.56 10^{8}	0	3.56 10^{-47}	3.56 10^{-47}	2.69 10^{-38}
Pu-239	1.55 10^{8}	0	3.46 10^{-31}	3.46 10^{-31}	5.38 10^{-23}
Pu-240	1.89 10^{8}	0	6.82 10^{-47}	6.82 10^{-47}	1.29 10^{-38}
Pu-241	9.39 10^{9}	0	6.53 10^{-42}	6.53 10^{-42}	6.13 10^{-32}
Pu-242	1.58 10^{8}	0	1.65 10^{-31}	1.65 10^{-31}	2.62 10^{-23}
Pu-244	1.26 10^{8}	0	4.26 10^{-24}	4.26 10^{-24}	5.37 10^{-16}
Am-241	3.03 10^{8}	0	4.88 10^{-31}	4.88 10^{-31}	1.48 10^{-22}
Am-242m	1.75 10^{7}	0	8.30 10^{-22}	8.30 10^{-22}	1.45 10^{-14}
Am-243	1.46 10^{8}	0	1.36 10^{-29}	1.36 10^{-29}	1.98 10^{-21}
Cm-242	1.48 10^{11}	0	1.79 10^{-49}	1.79 10^{-49}	2.64 10^{-38}
Cm-243	4.89 10^{7}	0	1.22 10^{-29}	1.22 10^{-29}	5.95 10^{-22}
Cm-244	1.16 10^{8}	0	1.05 10^{-56}	1.05 10^{-56}	1.22 10^{-48}

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose from radiological capacity (µSv y^{-1})
		Gas	External	Total	
Cm-245	1.26 10^{7}	0	2.91 10^{-34}	2.91 10^{-34}	3.67 10^{-27}
Cm-246	1.27 10^{7}	0	1.69 10^{-69}	1.69 10^{-69}	2.14 10^{-62}
Cm-248	1.45 10^{7}	0	2.22 10^{-30}	2.22 10^{-30}	3.22 10^{-23}

765. The doses calculated using illustrative inventories are considered further in Appendix D.

E.4.3. Groundwater pathways

766. A mathematical model has been implemented in the GoldSim program (GoldSim Technology Group, 2013) that considers the groundwater pathways. GoldSim is considered to be appropriate because:

- it provides a flexible modelling environment and was used for the ENRMF ESC;
- decay and ingrowth of radionuclides can be modelled in the standard application; and,
- models for well-mixed compartment and one-dimensional transport are available in the Contaminant Transport Module (GoldSim Technology Group, 2013).

767. The model has been developed for the Port Clarence site taking into account the characteristics described in Section 2 of the main report. Calculations have been undertaken for the combined landfill comprising the non-hazardous and the hazardous waste cells and sensitivity studies have considered the hazardous and non-hazardous waste cells separately.

768. The scenarios consider the exposures resulting from contaminated groundwater taken from a hypothetical abstraction point close to the site boundary. There are no abstraction wells near to the site and there are unlikely to be any in the future for the following reasons:

- the strata from which abstraction of contaminated water would occur is subject to the subsurface saline interface with the Tees Estuary (MJCA, 2019b) meaning the waters are not suitable for drinking, irrigation or livestock;
- sea level rise is likely to isolate the site over long timescales; and,
- erosion scenarios are considered more likely to occur than water abstraction between the landfill and the estuary.

769. Results of groundwater calculations are used in other assessments, such as assessments of releases to the estuary (see Section E.4.4) and the non-human biota

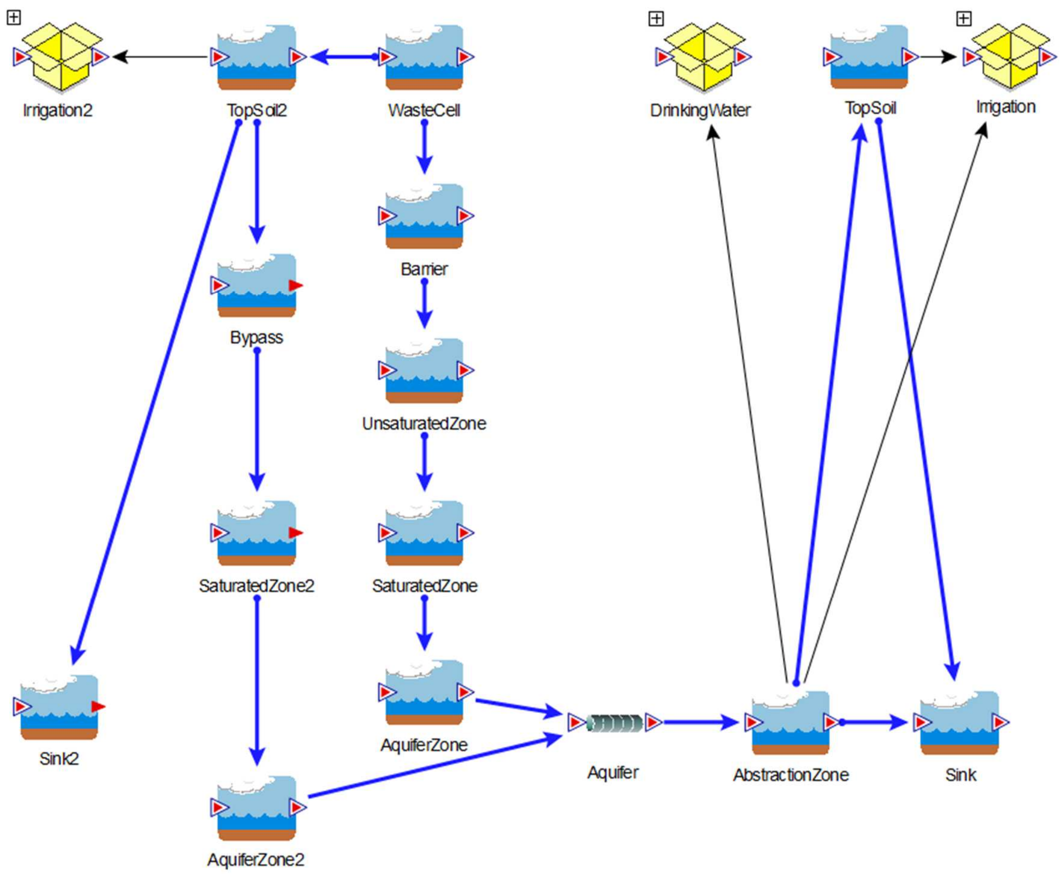
assessments (see Section E.7). Calculations consider the concentration of radionuclides in leachate and groundwater as specified below:

- Activity concentrations in leachate at the end of the period of authorisation and peak activity concentrations after the period of authorisation are calculated.
- Peak activity concentrations in groundwater are calculated at the interface with the estuary, which is 520 m from the centre of the non-hazardous waste cell, 360 m from the centre of the hazardous waste cell and 440 m from the centre of the combined landfill.
- Peak activity concentrations in subsoil are calculated, which occur due to bathtubbing.

770. Although the presence of an abstraction point is not considered credible either during the period of authorization or afterwards, activity concentrations are calculated for information purposes at a point 100 m from the edge of the landfill. This location is 340 m from the centre of the non-hazardous waste cell, 180 m from the centre of the hazardous waste cell and 260 m from the centre of the combined landfill.

771. The structure of the GoldSim model is shown in Figure 18. All compartments are assumed to be well mixed cells, apart from the aquifer, in which one-dimensional flow is assumed to occur. More details about the compartments are given below.

Figure 18 Compartments as modelled in GoldSim



772. Table 6 lists the radionuclides of interest with their half-lives, short-lived daughters where applicable and radioactive daughters considered explicitly. GoldSim adds the appropriate terms for radioactive decay and ingrowth to the equations governing the dynamics of the compartments. The equation for radioactive decay and ingrowth is:

$$\left(\frac{dN_{Rn,Comp}}{dt}\right)_{Decay} = \lambda_{PN} \cdot N_{PN,Comp} - \lambda_{RN} \cdot N_{Rn,Comp}$$

$$A_{Rn,Comp} = N_{Rn,Comp} \cdot \lambda_{Rn}$$

where:

- $N_{Rn,Comp}$ is the number of atoms of radionuclide Rn;
- $N_{PN,Comp}$ is the number of atoms of the parent radionuclide PN;
- λ_{Rn} is the decay constant of radionuclide Rn (s^{-1});

- λ_{PN} is the decay constant of the parent radionuclide PN (s^{-1}); and,
- $A_{Rn,Comp}$ the activity of radionuclide Rn.

773. Decay systems corresponding to a number of radionuclide chains are illustrated in Section 3.2 of the main report. Short-lived daughters that are assumed to be in secular equilibrium with a longer-lived parent radionuclide have been omitted from the figure.

774. In all the calculations, the quantities of long-lived daughters that have ingrown from specific parents or were directly disposed were distinguished. For example, the model considers seven variants of U-234, all with identical decay and sorption properties:

- U-234 directly disposed;
- U-234 ingrown from Pu-238;
- U-234 ingrown from U-238;
- U-234 ingrown from Pu-242;
- U-234 ingrown from Cm-242;
- U-234 ingrown from Am-242m; and,
- U-234 ingrown from Cm-246.

775. The dose coefficients include the contribution of all listed short-lived daughters assuming that those daughters are in secular equilibrium. Thus, the dose coefficient for U-238 includes the contributions from Th-234, Pa-234m and Pa-234.

E.4.3.1. Waste cells

776. The engineered cap and the waste cell design are discussed in Section E.1.3. Further details are presented below on the relevant equations and parameter values used in the GoldSim model. Compartments have been defined corresponding to the different landfill components identified above. In each compartment, the waste is assumed to be well mixed. The compartment is assumed to be saturated and contaminants are distributed between pore water and soil according to a linear equilibrium sorption model.

Activity in the waste inventories

777. Calculations were undertaken for a nominal disposal inventory of 1 MBq, distributed evenly through the landfill. As all radiological impacts from the groundwater pathway scale with the disposed inventory, the results of these calculations serve as a basis for calculation of:

- the radiological capacity for disposal of specific radionuclides; and,
- the contribution to radiological impact from the disposal of example wastes (similar to those disposed at the ENRMF).

Water flux

778. The water flux (q) through the waste cell is determined by the infiltration flux through the cap and by the efficiency of the basal liner and the clay barrier.

779. The infiltration flux through the cap, $q_{Infiltration}$, ($m^3\ y^{-1}$) has been defined as:

$$q_{Infiltration} = P_{eff} \cdot A_{Surface}$$

where

- $A_{Surface}$ represents the surface area of the component of the landfill being considered (m^2), and
- P_{eff} represents the effective infiltration into the waste cell, defined as:

$$P_{eff} = \begin{cases} P_{Cap} \text{ if } t < t_{StartCapDegradation} \\ P_{Grassland} \text{ if } t > t_{EndCapDegradation} \\ P_{Cap} + \dfrac{(t - t_{StartCapDegradation}) \cdot (P_{Grassland} - P_{Cap})}{t_{EndCapDegradation} - t_{StartCapDegradation}} \text{ otherwise} \end{cases}$$

780. Before cap degradation starts (i.e. before $t_{StartCapDegradation}$), the cap design infiltration rate P_{Cap} is assumed to be valid. When the HDPE component of the cap has fully degraded at $t_{EndCapDegradation}$, the vegetation on top of the landfill area is assumed to be grassland, and hence the infiltration into the waste cells would be defined by the infiltration to grassland $P_{Grassland}$. The cap is assumed to degrade in such a way that the infiltration increases linearly between $t_{StartCapDegradation}$ and $t_{EndCapDegradation}$ (Eden NE, 2015a).

781. The parameters used to calculate the effective infiltration have been assigned values as defined in Table 104. All these parameter values are taken from the HRA (MJCA, 2019a).

Table 104 Parameters to calculate the effective infiltration through the cap.

Parameter	Description	Value
P_{Cap}	Cap design infiltration (consistent with HRA)	0.0315 m y^{-1}
$P_{Grassland}$	Infiltration to grassland (consistent with HRA)	0.202 m y^{-1}
$t_{StartCapDegradation}$	Start of cap degradation	250 y
$t_{EndCapDegradation}$	End of cap degradation	1,000 y

782. The same formula for the potential flux through the HDPE liner adopted in the LandSim model (Gane, 2014) is used. The maximum water flux q_{Liner} through the basal liner ($m^3\ y^{-1}$) is defined in LandSim as:

$$q_{Liner} = \sum_{Defect} n_{Defect} \cdot q_{Defect}$$

where:

$$q_{Defect} = c \cdot a_{Defect}^{0.1} \cdot h^{0.9} \cdot K_{Barrier}^{0.74} \cdot 3.16E+07$$

783. The initial number and type of defects are as defined in Table 105 and other parameters defined in Table 106. The area of defects and the number of defects at the end of the period of authorisation were derived from LandSim data (Environment Agency, 2003).

Table 105 Assumptions regarding initial defects in the liner

Defect	Area defect a_{Defect} (mm^2)	Number of defects at time of installation n_{Defect} (ha^{-1})
Pinhole	2.55	12.5
Hole	52.5	2.5
Tear	5050	0.1

Table 106 Parameters to calculate the flow through the waste cells.

Parameter	Units	Value	Description
C		1.05	Contact quality parameter (between 0.21 and 1.15)
H	m	1	Leachate head
$K_{Barrier}$	$m\ s^{-1}$	$5.95\ 10^{-11}$	Hydraulic conductivity of the clay barrier
$d_{Barrier}$	m	1 or 1.5	Thickness of the clay barrier, 1 m for the non-hazardous waste cell and 1.5 m for the hazardous waste cell.

784. The parameter values in Table 106 were taken from (MJCA, 2019a), except for the liner contact quality parameter *c*, which is given a credible central value, based on minimum and maximum values that were established in LandSim (Environment Agency, 2003).

785. The number of defects is assumed to double every 100 years (Augean, 2011b).

786. The potential flux through the clay barrier ($m^3\ y^{-1}$) is defined as:

$$q_{Barrier} = K_{Barrier} \cdot A_{Basal} \cdot \frac{h}{d_{Barrier}} \cdot 3.16\text{E}7$$

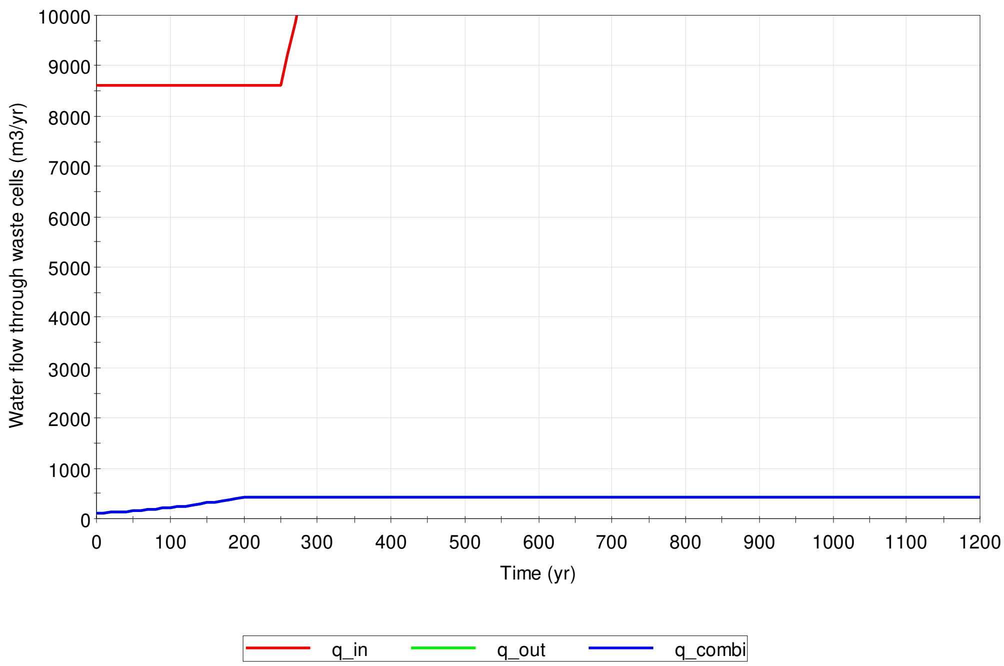
where:

- A_{Basal} the basal area (m^2) of the landfill component being considered; and,
- 3.16E7 is the conversion from s to years (s/y).

787. It is assumed that the infiltration through the cap controls the flow of water through the base of the landfill unless either $q_{Barrier}$ or q_{Liner} is less than $q_{Infiltration}$. The flow of water through the base of the landfill (*q*) is determined as the minimum of $q_{Infiltration}$, $q_{Barrier}$ and q_{Liner}.

788. The water flux from the waste cell to the clay barrier for the combined landfill is shown in Figure 19. In these figures, q_in represents the infiltration through the cap, q_out represents the potential flow of water through the base of the cells and q_combi represents the actual flow of water through the base of the cells.

Figure 19 Water flux through the waste cells for the currently permitted landfill



789. Until the end of the management phase (MJCA, 2019b), leachate is monitored and managed to ensure that leachate levels remain below the permitted limits (trigger levels: head < 2 m in sumps and head < 1 m in boreholes). Excess leachate is pumped off and either used in the adjacent treatment plant or transported off-site by tanker for treatment. The fate of leachate after the end of the management phase is addressed in Section E.4.3.5.

Materials

790. Proportions and properties of waste and filling materials are given in Table 107. The proportions of waste and soil are based on (Wilson, 2013). The density, water content and hydraulic conductivity of waste and clay were taken from the HRA (MJCA, 2019b) and the density of soil was taken from a previous assessment (Eden NE, 2015a).

Table 107 Proportions and properties of waste and filling materials

Material	Proportion	Density (kg m^{-3})	Water Content	Hydraulic conductivity (m s^{-1})
Waste	20%	920	0.1	1 10^{-5}
Soil (incl. soil type waste)	80%	1,300	0.25	1 10^{-5}
Clay	0%	1,810	0.17	5.95 10^{-11}

791. The sorption distribution coefficients (K_d's) for the filling materials are given in Table 199.

792. Radionuclides sorb to different materials in the waste cells such that the activity concentration in leachate ($a_{Leachate}$) in Bq m^{-3} is:

$$a_{RN,Leachate} = \frac{A_{Rn,Cell}}{\sum_{Mat} M_{Mat,Cell} \cdot K_{d,Rn,Mat} + V_{Water,Cell}}$$

with:

- $A_{Rn,Cell}$ the total activity of radionuclide *Rn* in the waste cell (Bq);
- $M_{Mat,Cell}$ the mass of material *Mat* in the waste cell (kg);
- $K_{d,Rn,Mat}$ the distribution coefficient for radionuclide *Rn* in material Mat (m^3 kg^{-1}); and,
- $V_{Water,Cell}$ the volume of water in the waste cell (m^3).

793. The mass of the different materials ($M_{Mat,Cell}$) and the volume of water in the waste cell ($V_{Water,Cell}$) are determined by the proportions and properties of the materials:

$$M_{Mat,Cell} = \rho_{Mat} \cdot V_{Cell} \cdot pr_{Mat}$$

$$V_{Water,Cell} = V_{Cell} \cdot \sum_{Mat} \varepsilon_{Mat} \cdot pr_{Mat}$$

where:

- ρ_{Mat} the density of material *Mat* (kg m^{-3});
- V_{Cell} the volume of the waste *cell* (m^3);
- pr_{Mat} the proportion of material *Mat* (dimensionless); and,
- ε_{Mat} the water content in material *Mat* (dimensionless).

794. The leaching rate of radionuclide *Rn* is defined as:

$$\left(\frac{dN_{Rn,Cell}}{dt}\right)_{Leaching} = -n_{Rn,Leachate} \cdot q = -\frac{N_{Rn,Cell} \cdot q}{\sum_{Mat} M_{Mat,Cell} \cdot K_{d,Rn,Mat} + V_{Water,Cell}}$$

where:

- $N_{Rn,Cell}$ the number of atoms of radionuclide *Rn* in the waste cells;
- $n_{Rn,Leachate}$ the number of atoms per unit volume of leachate (m^{-3}); and,
- q the flow of water through the base of the landfill ($m^3\ y^{-1}$).

795. The same differential equation applies to activity. Radioactive decay and ingrowth are also applied to the atoms of radionuclide *Rn* in the waste cells.

E.4.3.2. Clay barrier

796. The clay barrier is represented as a well-mixed compartment with equilibrium sorption of contaminants to the clay. Flow through the barrier is subvertical from the waste cell to the unsaturated zone.

797. The dimensions of the barrier for the different calculation cases are defined in Table 108. The barrier thicknesses for the non-hazardous and hazardous waste cells have been taken from the HRA (MJCA, 2019a). The minimum barrier thickness has been selected for the assessment of the combined landfill as a conservative assumption.

Table 108 Dimensions of the barrier

Site	Basal area (m^2)	Clay barrier thickness (m)
Non-hazardous waste cell	142,594	1
Hazardous waste cell	130,874	1.5
Combined landfill	273,468	1

798. The clay barrier is assumed to be in hydrological equilibrium, which means that the water flux through the barrier is assumed to be equal to the water flux through the waste cells, i.e. *q*.

799. The barrier is constructed out of Upper Lias Clay. The properties and K_d values for clay have been defined in Table 107 and Table 199.

800. The behaviour of radionuclide *Rn* in the barrier is described by the following differential equation.

$$\left(\frac{dN_{RN,Barrier}}{dt}\right)_{GW} = \frac{N_{RN,Cell} \cdot q}{\sum_{Mat} M_{Mat,Cell} \cdot K_{d,RN,Mat} + V_{Water,Cell}} - \frac{N_{RN,Barrier} \cdot q}{M_{Clay,Barrier} \cdot K_{d,RN,Clay} + V_{Water,Barrier}}$$

801. The first term relates to the flux from the waste cell into the clay barrier. The second term relates to the flux from the clay barrier into the unsaturated zone. The subscript index *GW* indicates that the equation includes the contributions from groundwater movement, and not radioactive decay. Radioactive decay and ingrowth are also applied to the atoms of radionuclide *Rn* in the barrier. The clay mass in the barrier ($M_{Clay,Barrier}$) and the water volume in the clay barrier ($V_{Water,Barrier}$) are given by:

$$M_{Clay,Barrier} = \rho_{Clay} \cdot V_{Barrier}$$

$$V_{Water,Barrier} = \varepsilon_{Clay} \cdot V_{Barrier}$$

where:

- ρ_{Clay} is the density of clay (kg m^{-3});
- ε_{Clay} is the water content of clay (dimensionless); and,
- $V_{Barrier}$ is the volume of the clay barrier (m^3).

E.4.3.3. Unsaturated zone

802. An unsaturated zone underlies the clay barrier and flow in the unsaturated zone is subvertical. The zone is represented as a well-mixed compartment.

803. The thicknesses of the unsaturated zone for the non-hazardous and hazardous waste cells are the same, see Table 109, and have been taken from the HRA (MJCA, 2019a). This thickness has also been selected for the assessment of the combined landfill.

Table 109 Dimensions of the unsaturated zone

Site	Basal area (m^2)	Unsaturated zone thickness (m)
Non-hazardous waste cell	142,594	1
Hazardous waste cell	130,874	1
Combined landfill	273,468	1

804. The water flux through the unsaturated zone is assumed to be equal to the water flux through the barrier and the waste cells.

805. The unsaturated zone consists of made ground, the properties of which have been taken from (MJCA, 2019a) and reproduced in Table 110.

Table 110 Properties of unsaturated made ground

Material	Density (kg^{-1} m^3)	Water content	Hydraulic conductivity (m s^{-1})
Unsaturated made ground	1.155	0.19	1.36 10^{-6}

806. The behaviour of radionuclide *Rn* in the unsaturated zone is represented by the following equation:

$$\left(\frac{dN_{RN,Unsat}}{dt}\right)_{GW} = \frac{N_{RN,Barrier} \cdot q}{M_{Clay,Barrier} \cdot K_{d,RN,Clay} + V_{Water,Barrier}} - \frac{N_{RN,Unsat} \cdot q}{V_{Water,Unsat}}$$

807. The first term relates to the flux from the clay barrier into the unsaturated zone. The second term relates to the flux from the unsaturated zone into the saturated zone. The volume of water in the unsaturated zone ($V_{Water,Unsat}$, m^3) is given by:

$$V_{Water,Unsat} = \varepsilon_{UnsatMadeGround} \cdot V_{Unsat}$$

where:

- $\varepsilon_{UnsatMadeGround}$ is the water content of unsaturated made ground; and,
- V_{Unsat} is the volume of the unsaturated zone (m^3).

808. Radioactive decay and ingrowth is also applied to radionuclide Rn, separately.

E.4.3.4. Saturated zone

809. A saturated zone underlies the unsaturated zone and flow in the saturated zone is subvertical. The zone is represented as a well-mixed compartment.

810. The thicknesses of the saturated zone for the non-hazardous and hazardous waste cells are the same, see Table 111, and have been taken from the HRA (MJCA, 2019a). The thickness has been selected for the assessment of the combined landfill.

Table 111 Dimensions of the saturated zone

Site	Basal area (m^2)	Saturated zone thickness (m)
Non-hazardous waste cell	142,594	2.7
Hazardous waste cell	130,874	2.7
Combined landfill	273,468	2.7

811. The saturated zone is assumed to be in hydrological equilibrium, which means that the water flux through the saturated zone is assumed to be equal to the water flux through the unsaturated zone, the barrier and the waste cells.

812. The saturated zone consists of made ground, the properties of which have been taken from (MJCA, 2019b) and reproduced in Table 112.

Table 112 Properties of saturated made ground

Material	Density (kg m^{-3})	Water content	Hydraulic conductivity (m s^{-1})
Saturated made ground	1155	0.25	1.36 10^{-6}

813. The behaviour of radionuclide *Rn* in the saturated zone is represented by the following equation:

$$\left(\frac{dN_{RN,Sat}}{dt}\right)_{GW} = \frac{N_{RN,Unsat} \cdot q}{M_{UnsatMadeGround,Unsat} \cdot K_{d,RN,UnsatMadeGround} + V_{Water,Unsat}} - \frac{N_{RN,Sat} \cdot q}{V_{Water,Sat}}$$

814. The first term relates to the flux from the unsaturated zone into the saturated zone. The second term relates to the flux from the saturated zone into the aquifer. The volume of water in the saturated zone ($V_{Water,Sat}$, m^3) is given by:

$$V_{Water,Sat} = \varepsilon_{SatMadeGround} \cdot V_{Sat}$$

where:

- $\varepsilon_{SatMadeGround}$ is the water content of saturated made ground; and,
- V_{Sat} is the volume of the saturated zone (m^3).

815. Radioactive decay and ingrowth are also applied to radionuclide *Rn*, separately.

E.4.3.5. Bathtubbing

816. After the management control period, the leachate level is no longer controlled, and the base of a waste cell may gradually fill up if the infiltration flow through the cap is higher than the flow through the basal liner and the clay barrier. When the clay barrier basin is full (the leachate reaches a height of 3 m), leachate overflow will occur, and leachate will disperse into the subsoil around the area. From the subsoil, the leachate will percolate through the soil into the saturated layer, bypassing the engineered barriers put in place for the landfill. Note that the clay layer proposed for the landfill engineered barrier (below the liner) will not degrade in the same way as an artificial membrane.

817. The GoldSim model calculates the water volume in the clay barrier basin and activates a switch when the capacity of the basin is reached. At this time the potential annual volume of leachate overflow ($V_{overflow}$) is determined as:

$$V_{overflow} = q_{in} - q_{out} \qquad if \;\; q_{in} > q_{out}$$

where:

- $V_{overflow}$ is the annual volume of leachate overflow;

- q_{in} is the inflow from precipitation at that time; and,
- q_{out} is the outflow through the liner at that time.

818. It is considered likely that overtopping will drain to subsoil rather than flood and saturate an extensive area or percolate to the site drainage channels. At this time the engineered barrier will be bypassed.

819. Water in the affected subsoil originates from leachate overflow and from infiltration. The water outflow from subsoil is limited by the maximum flow through unsaturated made ground. The affected area (A_{Flood}) is calculated by GoldSim by assuming hydrological equilibrium, expressed as:

$$P_{Farmland} \cdot A_{Flood} + q_{Leachate} = K_{UnsatMadeGround} \cdot 3.16E7 \cdot A_{Flood} \cdot \frac{h_{subsoil}}{d_{overflow}}$$

where:

- the first term represents infiltration;
- the second term represents leachate overflow; and,
- the third term represents the maximum flow through unsaturated made ground.

820. The affected area (m^2) is calculated as:

$$A_{Flood} = q_{Leachate} / \left(K_{UnsatMadeGround} \cdot 3.16\text{E}7 \cdot \frac{h_{subsoil}}{d_{overflow}} - P_{Farmland} \right)$$

where:

- A_{Flood} the affected area (m^2);
- $q_{Leachate}$ the leachate overflow rate ($m^3\ y^{-1}$);
- $K_{UnsatMadeGround}$ the hydraulic conductivity of unsaturated made ground ($m\ s^{-1}$);
- $h_{subsoil}$ the hydraulic head in the subsoil (m), set to be 10% of the initial leachate head in the waste cells;
- $d_{overflow}$ the thickness of the receiving overflow compartment (m);
- $P_{Farmland}$ the infiltration in farmland ($m\ y^{-1}$); and,
- 3.16E7 is the conversion from s to years (s/y).

821. The thickness of the bypass compartment is defined as:

$$d_{overflow} = d_{UnsatZone} + d_{Barrier} + d_{Basin} - d_{subsoil}$$

where:

- $d_{overflow}$ is the thickness of the receiving overflow compartment (m);
- $d_{UnsatZone}$ is the thickness of the unsaturated zone (m);
- $d_{Barrier}$ is the thickness of the clay barrier (m);
- d_{Basin} is the height of the basin (m); and,
- $d_{Topsoil}$ is the thickness of the subsoil (m).

822. The bypass compartment has the same water content and the hydraulic conductivity as the unsaturated zone as it contains unsaturated made ground. A saturated zone is located beneath the bypass compartment unsaturated zone, with the same thickness, water content and hydraulic conductivity as the saturated zone beneath the landfill.

E.4.3.6. Aquifer

823. The aquifer consists of alluvium. In addition to the properties defined in Table 113, the hydraulic gradient in the aquifer is taken as 0.0107 (Eden NE, 2015a) based on (British Geological Survey and the Environment Agency, 2006).

824. The horizontal groundwater volume flux (m^3 s^{-1}) in the aquifer is defined as:

$$q_{Aquifer} = K_{Alluvium} \cdot \Delta_H \cdot W_{Aquifer} \cdot d_{Aquifer}$$

where:

- $K_{Alluvium}$ is the hydraulic conductivity of Alluvium (m s^{-1});
- Δ_H is the hydraulic gradient (dimensionless);
- $W_{Aquifer}$ is the width of the aquifer pathway (m); and,
- $d_{Aquifer}$ is the thickness of the aquifer (m).

825. Although the aquifer is assumed to be a continuous medium, it is modelled in four zones:

- The aquifer zone: this is the volume of alluvium right beneath the waste cells. This zone is modelled as a single aquifer cell. Modelling this zone separately allows us to define the length of the aquifer transport zone as the migration distance down gradient from the edge of the landfill to the point of interest. The water flux into this zone is assumed to be equal to the water flux out of the saturated zone. This contaminated water is also mixed with clean water from the upstream part of the aquifer. The horizontal water flux in the aquifer is much higher than the vertical water flux into the aquifer.
- The overflow aquifer zone: this is the volume of alluvium right beneath the flooded area when leachate overflow occurs. This zone is modelled in a similar way to the aquifer zone.

- The aquifer transport zone: perpendicular diffusion has not been accounted for in the model. A one-dimensional transport model has been used to represent the transport in the aquifer away down gradient from the landfill, modelled as a sequence of 10 aquifer cells. The groundwater migration distance is assumed to be equal to the distance between the centre of the landfill and the well.
- The abstraction zone: in order to evaluate the activity concentration at the position where the well is located, an additional aquifer cell is introduced in the model.

Aquifer Zone

826. The aquifer zone is a compartment corresponding to the area of the aquifer located beneath the waste cells and serves as an interface between the leaching zone and the aquifer.

827. The dimensions of the aquifer zone for the different calculation cases are defined in Table 113. The thicknesses of the aquifer zone for the non-hazardous and hazardous waste cells have been taken from the HRA (MJCA, 2019b) and are the same value. This thickness has been selected for the assessment of the combined landfill.

Table 113 Dimensions of the aquifer zone

Site	Basal area (m^2)	Aquifer zone thickness (m)	Width of aquifer perpendicular to flow direction (m)
Non-hazardous waste cell	142,594	7.05	404
Hazardous waste cell	130,874	7.05	689
Combined landfill	273,468	7.05	689

828. The water flux into the aquifer zone has two components; the vertical water flux through the saturated zone (coming from the unsaturated zone, the barrier and the waste cells), and the horizontal groundwater flux.

829. In order to conserve water, the outward flux is equal to the sum of the two inward fluxes. Since the vertical flux through the landfill is a small fraction of the horizontal flux, this adjustment is negligible.

830. The behaviour of radionuclide *Rn* in the aquifer zone is represented by the following equation:

$$\left(\frac{dN_{Rn,AquiferZone}}{dt}\right)_{GW} = \frac{N_{Rn,Sat} \cdot q}{V_{Water,Sat}} - \frac{N_{Rn,\mathrm{Aquifer}} \cdot q_{Aquifer}}{V_{Water,AquiferZone}}$$

831. The first term relates to the flux from the saturated zone into the aquifer zone right beneath the waste cell. The second term relates to the flux from the aquifer zone horizontally downgradient and away from the landfill, and into the aquifer transport zone. The volume of water in the aquifer zone ($V_{Water,AquiferZone}$, m^3) is given by:

$$V_{Water,AquiferZone} = \varepsilon_{Alluvium} \cdot V_{AquiferZone}$$

where:

- $\varepsilon_{Alluvium}$ is the water content of alluvium (dimensionless); and,
- $V_{AquiferZone}$ is the volume of the dilution zone (m^3).

832. Radioactive decay and ingrowth are also applied to radionuclide *Rn*, separately.

Overflow Aquifer Zone

833. The overflow aquifer zone is a compartment corresponding to the area of the aquifer located beneath the flooded area when leachate overflow occurs; it serves as an interface to the aquifer transport zone.

834. The surface area of the overflow aquifer zone is equal to the flooded area when leachate overflow occurs. The width of the overflow aquifer is defined as:

$$w_{OverAquifer} = \sqrt{A_{Flood}}$$

835. The water flux into the aquifer zone has two components; the vertical water flux through the saturated bypass zone (from the unsaturated overflow zone and the flooded subsoil), and the horizontal groundwater flux.

836. In order to conserve water, the outward flux is equal to the sum of the two inward fluxes.

837. Leachate overflow is assumed to be perpendicular to the groundwater flow, so that the bypass flow and bypass groundwater flow are added to the groundwater flow in the aquifer.

838. The behaviour of radionuclide *Rn* in the aquifer zone is represented by the following equation:

$$\left(\frac{dN_{Rn,\mathrm{Overflow}AquiferZone}}{dt}\right)_{GW} = \frac{N_{Rn,\mathrm{Over}Sat} \cdot q}{V_{Water,\mathrm{Over}Sat}} - \frac{N_{Rn,\mathrm{OverAquifer}} \cdot q_{OverAquifer}}{V_{Water,\mathrm{Over}AquiferZone}}$$

839. The first term relates to the flux from the saturated zone into the bypass aquifer zone right beneath the flooded area. The second term relates to the flux from the aquifer zone horizontally away from the flooded area and into the aquifer transport zone. The volume of water in the bypass aquifer zone ($V_{Water,\mathrm{Bypass}AquiferZone}$, m^3) is given by:

$$V_{Water,\mathrm{Over}AquiferZone} = \varepsilon_{Alluvium} \cdot V_{OverAquiferZone}$$

where:

- $\varepsilon_{Alluvium}$ is the water content of alluvium (dimensionless); and

- $V_{BypassAquiferZone}$ is the volume of the dilution zone (m^3).

840. Radioactive decay and ingrowth are also applied to radionuclide *Rn*, separately.

Aquifer Transport Zone

841. The aquifer transport zone is the region of the aquifer between the aquifer zone beneath the waste cells and the flooded area, and the abstraction zone, where the abstraction well is located.

842. The dimensions of the aquifer transport zone for the different calculation cases are given in Table 114. As noted previously, the length of the pathway has been taken to be equivalent to the maximum length considered in the HRA.

843. The "aquifer" model in GoldSim represents a one-dimensional transport pathway as a series of cells. Advection and dispersion in the aquifer are modelled, together with radioactive decay and ingrowth. The GoldSim model was set up with a default value of ten cells. The dispersivity was set to the length of the aquifer pathway divided by ten.

Table 114 Dimensions of the aquifer transport zone

Site	Length of Aquifer Pathway (m)	Width of Aquifer Pathway (m)	Thickness of Aquifer (m)
Abstraction well			
Non-hazardous waste cell	340	404 + $w_{OverAquiferZone}$	7.05
Hazardous waste cell	180	689 + $w_{OverAquiferZone}$	7.05
Combined landfill	260	689 + $w_{OverAquiferZone}$	7.05
Discharge into the estuary			
Non-hazardous waste cell	520	404 + $w_{OverAquiferZone}$	7.05
Hazardous waste cell	360	689 + $w_{OverAquiferZone}$	7.05
Combined landfill	440	689 + $w_{OverAquiferZone}$	7.05

Abstraction Zone

844. The abstraction zone is the section of the aquifer in which an abstraction well is assumed to be located. This compartment has been introduced in the GoldSim model to allow evaluation of the activity concentration in the groundwater at the location of abstraction.

845. The dimensions of the abstraction zone for the different calculation cases are defined in Table 115. The width and thicknesses are the same as the values for the aquifer transport zone and the saturated zone. The length of the abstraction zone will not significantly affect the results because a strong gradient is not expected. An arbitrary value of 10 m is chosen.

Table 115 Dimensions of the abstraction zone

Site	Length of abstraction zone (m)	Width of abstraction zone (m)	Thickness of abstraction zone (m)
Non-hazardous waste cell	10	404 + $w_{OverAquiferZone}$	7.05
Hazardous waste cell	10	689 + $w_{OverAquiferZone}$	7.05
Combined landfill	10	689 + $w_{OverAquiferZone}$	7.05

846. The abstraction zone is assumed to be in hydrological equilibrium, which means that the water flux into the abstraction zone is equal to the water flux out of the aquifer transport zone. The water fluxes out are the flux abstracted by the well and the remaining horizontal flux in the aquifer. Radioactive decay and ingrowth are also applied, separately.

E.4.3.7. Assessment calculations for a hypothetical abstraction point

847. The contamination of groundwater under the landfill is expected to occur at some point in the future. The HRA shows degradation of the landfill basal liner and cap over time resulting in leachate flows to the underlying substrate and then to groundwater.

848. If contaminated groundwater discharges to a surface water body (spring, river, sea), then ingestion of drinking water and foodstuffs from the surface water body is also a potential exposure pathway and this is considered. These discharges would be subject to additional dilution by groundwater, surface runoff and drainage water thereby reducing exposure relative to an abstraction point. Discharge of groundwater to the estuary is considered in Section E.4.3.9.

849. The dose criterion used is a dose of 0.02 mSv y^{-1} for the public (this is equivalent to the risk guidance level of 10^{-6} y^{-1} for exposure of the public post closure, for situations that are expected to occur). However, the groundwater pathway is not used to limit the radiological capacity for the reasons given above.

Exposed group

850. Exposure of members of the public is assumed to occur as a result of using abstracted water for irrigation and drinking water. Members of the exposed group are assumed to be adults and to be exposed as a result of:

- consumption of drinking water from the borehole;
- consumption of food produced on irrigated land including milk, green vegetables, root vegetables and meat products;
- external irradiation from radionuclides incorporated in contaminated soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

851. The drinking water consumption rate for adults used in the assessment is 600 l y^{-1} (Smith & Jones, 2003) and the habit assumptions applied to an adult in a farming family irrigating soil are used for the irrigation pathways (see Table 83).

852. The National Dose Assessments Working Group published guidance recently on the use of habit data in prospective dose assessments (NDAWG, 2013). This suggested that the two foodstuffs likely to be most restrictive in terms of their radionuclide content (hence dose potential), should be assumed to be consumed at an elevated rate and all other foodstuffs, that may be reasonably assumed to be sourced locally, are assumed to be consumed at average consumption rates expressed on a per consumer basis.

853. The HPA have issued generic consumption data (Smith & Jones, 2003). In general, the consumption rates assumed in the EA methodology represent, for every food group considered, the 97.5th percentile consumption rate. Summing over foodstuffs will therefore give a conservative dose assessment that is appropriate for preliminary scoping assessments. For more realistic assessments it is not appropriate to assume that all foods are consumed at this high rate, in terms of diet and calorific intake, particularly for longer term assessments. The ESC has therefore followed the approach in the NDAWG guidance.

854. Table 83 details the habit data assumed for a farming family irrigating land with abstracted groundwater. They are also used for scenarios involving the application of sewage sludge to farmland, spillage resulting in contamination of a water body and intrusion after the period of authorisation leading to contamination of land used by a smallholding. For each of these cases, the two most restrictive pathways use 97.5th percentile consumption rates and the mean consumption rate is used for the remaining pathways.

Table 116 Habit data for the farming family irrigating soil with groundwater

Pathway	Adult average	Adult 97.5th	Child average	Child 97.5th	Infant Average	Infant 97.5th
Milk consumption[1] (l y^{-1})	122.5	240	127	240	148	320
Meat consumption[1] (kg y^{-1})	23	70	19	40	3.8	13
Green & other domestic veg consumption[1] (kg y^{-1})	35	80	15	35	5	15
Root veg & potatoes consumption[1] (kg y^{-1})	60	130	50	95	15	45
Breathing rate[2] (m^3 h^{-1})	1		0.64		0.22	
Inadvertent soil ingestion[2] (kg y^{-1})	0.03		0.018		0.044	
Occupancy[3] (h y^{-1})	8,760		8,760		8,760	
Indoor shielding factor[3]	0.1		0.1		0.1	
Fraction of time spent indoors[2]	0.75		0.84		0.9	

Notes: 1) From (Smith & Jones, 2003). 2) From (Augean, 2009). 3) Standard assumption in (Environment Agency, 2006b).

855. The GoldSim model used to model the groundwater migration scenario also includes a soil compartment which receives inputs from abstracted water that is used for irrigation and losses due to leaching from top soil. Direct contamination of crops (green vegetables and root vegetables) by irrigation water is also considered. The applicable irrigation rate will be crop dependent but based on green crops (Finch, et al., 2002) it would be about 0.15 m y^{-1}. This is the value used in the assessment. It is further assumed that sufficient water is extracted from the borehole to provide the implied demand.

856. The peak activity concentration in the groundwater to 100,000 years is used to calculate the doses to the exposed group.

Use of Groundwater as Drinking Water

857. The dose due to drinking abstracted groundwater is given by:

$$Dose_{drinkwater} = Q_{water} \cdot C_{R,groundwater}(t) \cdot D_{Rn,ing}$$

where:

- Q_{water} drinking water consumption rate (l y^{-1});
- $C_{Rn,groundwater}(t)$ activity concentration of radionuclide Rn in the groundwater used for irrigation at time t (Bq l^{-1}); and,
- $D_{Rn,ing}$ dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

858. Drinking water consumption rate for adults taken from (NRPB, 2003b) is 600 l y^{-1}.

859. The activity concentrations of radionuclides in irrigation water are determined by the groundwater transport model outlined above (section E.4.3.4).

860. Dose coefficients are presented in Table 200 and Table 201 for all radionuclides except those listed in Table 117, which use the values shown below to account for different assumptions concerning daughter radionuclides in Goldsim.

Table 117 Dose coefficients used in Goldsim to match the modelling of the decay chains

Radionuclide	Ingestion ($Sv\ Bq^{-1}$)	Inhalation ($Sv\ Bq^{-1}$)	External Irradiation from slab ($Sv\ y^{-1}\ Bq^{-1}\ kg$)
Pb-210	$6.91\ 10^{-7}$	$5.69\ 10^{-6}$	$1.64\ 10^{-9}$
Ra-226	$2.8\ 10^{-7}$	$9.53\ 10^{-6}$	$3.02\ 10^{-6}$
Ra-228	$6.90\ 10^{-7}$	$1.60\ 10^{-5}$	$1.62\ 10^{-6}$
Th-232	$2.3\ 10^{-7}$	$1.10\ 10^{-4}$	$1.41\ 10^{-10}$
U-232	$3.30\ 10^{-7}$	$3.70\ 10^{-5}$	$3.98\ 10^{-6}$
Am-242m	$1.90\ 10^{-7}$	$9.19\ 10^{-5}$	$1.81\ 10^{-8}$
Cm-243	$1.50\ 10^{-7}$	$6.90\ 10^{-5}$	$1.57\ 10^{-7}$

Use of Groundwater for Irrigation of Farmland

861. If abstracted water is used for irrigation, then doses can result from:

- ingestion of foodstuff grown or raised on contaminated soil;
- ingestion or inhalation of dust from the soil; and,
- external exposure to the soil.

862. Abstracted groundwater is applied to the top soil compartment at the irrigation rate. As infiltration (rain water) will also enter the top soil compartment (on different days from irrigation water), the annual water flux out of the top soil compartment is the sum of the irrigation rate and the infiltration rate. As farmland is similar to grassland in terms of runoff and evapotranspiration, the infiltration rate for grassland has been used for farmland (MJCA, 2019b).

863. Activity builds up in the top soil over time, as irrigation with contaminated groundwater continues. The behaviour of radionuclide *Rn* in the top soil is represented by the following equation:

$$\left(\frac{dN_{Rn,TopSoil}}{dt}\right)_{GW} = n_{RnN,Water,Sat} \cdot a_{Farmland} \cdot r_{irrigation} - \frac{N_{Rn,TopSoil} \cdot a_{Farmland} \cdot \left(r_{irrigation} + r_{infiltration}\right)}{M_{Soil,TopSoil} \cdot K_{d,RN,Soil} + V_{Water,TopSoil}}$$

864. The first term relates to irrigation of top soil with groundwater. The second term relates to leaching from the top soil. Radioactive decay and ingrowth are also addressed, separately. The parameters are:

- $n_{Rn,Water,Sat}$ number of atoms of radionuclide *Rn* per unit volume of groundwater (m^{-3});
- $a_{Farmland}$ area of farmland (m^2);
- $r_{irrigation}$ irrigation rate ($m\ y^{-1}$);

- $N_{Rn,TopSoil}$ number of atoms of radionuclide Rn in the top soil compartment;
- $r_{Infiltration}$ infiltration rate (m y^{-1}); and,
- $K_{d,Rn,Soil}$ the distribution coefficient for radionuclide *Rn* in material *Soil* (m^3 kg^{-1}).

865. The mass of soil ($M_{Soil,TopSoil}$) and volume of water ($V_{Water,TopSoil}$) in the top soil compartment are given by:

$$V_{Water,TopSoil} = \varepsilon_{TopSoil} \cdot \vartheta_{TopSoil} \cdot a_{Farmland} \cdot d_{TopSoil}$$

$$M_{Soil,TopSoil} = \rho_{Soil} \cdot a_{Farmland} \cdot d_{TopSoil}$$

where:

- $\varepsilon_{TopSoil}$ porosity of top soil (dimensionless);
- $\vartheta_{TopSoil}$ degree of saturation of top soil (dimensionless);
- ρ_{Soil} density of soil (kg m^{-3}); and,
- $d_{TopSoil}$ the depth of top soil (m).

866. Assumptions regarding the top soil compartment, used to calculate the volume of water and the mass of soil, are summarised in Table 118. The area of farmland assumed is arbitrary and does not affect the calculated doses since it cancels out when the activity concentration in the soil is calculated (see later). Soil properties are taken from the ENRMF assessment (Eden NE, 2015a).

Table 118 Dimensions and properties of top soil used for farming.

Parameter	Units	Value
Area of farmland	m^2	7,000
Depth of soil irrigated	m	1
Top soil porosity	dimensionless	0.3
Top soil saturation	dimensionless	0.5

867. The dose from ingesting crops grown on contaminated soil is given by a combination of interception of contaminated irrigation water by plants and root uptake in plants from contaminated soil (Eden NE, 2015a):

$$Dose_{ing,crops} = \sum_{crop} \left\{ Q_{crop} \cdot \left[C_{Rn,water}(t) \cdot \left(\frac{Irrig \cdot Int_{crop} \cdot F_{crop}}{Yield_{crop}} \right) + C_{RN,soil}(t) \cdot UF_{Rn,crop} \right] \right\} \cdot D_{Rn,ing}$$

where:

- Q_{crop} Crop consumption rate (kg y^{-1});
- $C_{Rn,water}$ Concentration of radionuclide *RN* in the irrigation water at time *t* (Bq l^{-1});
- *Irrig* Irrigation rate (m y^{-1});
- Int_{crop} Effective interception factor;
- F_{crop} Fraction remaining after processing;
- $Yield_{crop}$ Crop yield (kg m^{-2} y^{-1});
- $C_{Rn,soil}$ Soil activity concentration of radionuclide *Rn* at time *t* (Bq kg^{-1});
- $UF_{Rn,crop}$ Soil to crop transfer factors for radionuclide *Rn* (Bq kg^{-1} fresh weight of crop per Bq kg^{-1} of soil); and,
- $D_{Rn,ing}$ Dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

868. Habit data are discussed above (see Table 116) and other parameter values are summarised in Table 119. The grain crop processing factor is set to zero on the basis that it is not common for grain crops to be irrigated (Finch, et al., 2002). The irrigation rate is derived from a soil moisture deficit calculated from monthly average rainfall recorded at Wittering (May to August is 215 mm) and a daily water requirement for green vegetables (about 365 mm over the same period).

Table 119 Overview of parameters used for the irrigation scenario (Eden NE, 2015a)

Parameter	Substance	Units	Value
Density	Soil	kg m^3	1,300
Porosity	Soil		0.3
Saturation	Soil		0.5
Irrigation rate	All crops	m y^{-1}	0.3
Infiltration rate [grassland from (Augean, 2011b)]	All crops	mm y^{-1}	202.4
Crop interception factor	All crops		0.33
Crop processing factor	Grain		0
	Green vegetables		0.3
	Root vegetables		1
Yield (crops)	Grain	kg m^{-2} y^{-1}	0.4
	Green vegetables	kg m^{-2} y^{-1}	3.0
	Root vegetables	kg m^{-2} y^{-1}	3.5
	Pasture	kg m^{-2} y^{-1}	1.7
Consumption rate (animal)	Pasture	kg d^{-1}	55
	Soils	kg d^{-1}	0.6
Occupancy outdoors (people)		y y^{-1}	0.25

Parameter	Substance	Units	Value
Shielding factor indoors			0.1
Occupancy dust		h y^{-1}	2,200
Dustload		kg m^{-3}	1 10^{-7}
Breathing rate adult		m^3 h^{-1}	1

869. The activity concentration of radionuclides in water within the soil ($C_{Rn,Water}$) is determined by GoldSim as the activity in water within the soil divided by the volume of water. The activity concentration of radionuclides in soil ($C_{Rn,Soil}$) is determined as the total activity in the soil including water divided by the dry mass of soil.

870. Soil to crop transfer factors are given in Table 203 and dose coefficients for ingestion are given in Table 200 except for Pb-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243 (Table 117).

871. The dose from ingesting animal foodstuffs (e.g. meat and milk) raised on contaminated land is given by (Eden NE, 2015a):

$$Dose_{ing,animal} = \sum_{animal} \{Q_{animal} \cdot [Q_{soil,A} \cdot C_{Rn,soil}(t) + Q_{pasture,A} \cdot C_{Rn,soil}(t) \cdot UF_{Rn,grass}] \cdot TF_{Rn,animal}\} \cdot D_{Rn,ing}$$

where:

- Q_{animal} is the consumption rate of animal foodstuff (kg y^{-1});
- $Q_{soil,A}$ is the soil consumption rate by the animal (kg day^{-1});
- $Q_{pasture,A}$ is the pasture consumption rate by the animal (kg day^{-1});
- $C_{Rn,soil}$ is the activity concentration of radionuclide *Rn* in soil (Bq kg^{-1});
- $UF_{Rn,Grass}$ is the uptake factor of radionuclide *Rn* by crop *Grass* (Bq kg^{-1} fresh weight per Bq kg^{-1} soil);
- $TF_{Rn,Animal}$ is the transfer factor of radionuclide *Rn* in animal produce *Animal* (d kg^{-1}); and,
- $D_{Rn,ing}$ is the dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

872. Parameter values are summarised in Table 119.

873. The sorption distribution coefficients are defined in Table 199 for soil.

874. Dose from inadvertent ingestion of soil is given by (Eden NE, 2015a):

$$Dose_{ing,soil} = Q_{soil,H} \cdot C_{Rn,soil}(t) \cdot D_{Rn,ing}$$

where:

- $Q_{soil,H}$ is the soil consumption rate by humans (kg y^{-1});
- $C_{RN,soil}(t)$ is the activity concentration of radionuclide *Rn* at time *t* (Bq kg^{-1}); and,
- $D_{Rn,ing}$ is the dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

875. The soil consumption rate is given in Table 119.

876. The activity concentration of radionuclides in soil ($C_{Rn,Soil}$) is determined as the total activity in the soil including water divided by the dry mass of soil.

877. Dose coefficients for ingestion are given in Table 200 except for Pb-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243 which are given in Table 117.

878. The dose from external irradiation while living and working on contaminated soil is given by (Eden NE, 2015a):

$$Dose_{irr,soil} = (O_{out} + O_{in} \cdot SF) \cdot C_{Rn,soil}(t) \cdot DF_{Rn,irr,slab}$$

where:

- O_{out} is the fraction of time spent outside, exposed to contaminated soil (y y^{-1});
- O_{in} is the fraction of time spent inside (y y^{-1});
- *SF* is the shielding factor from the ground while indoors;
- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide *Rn* at time *t* in soil (Bq kg^{-1}); and,
- $DF_{Rn,irr,slab}$ is the dose coefficient for irradiation from radionuclide *Rn* (Sv y^{-1} Bq^{-1} kg), based on the receptor being 1 m from the ground and assuming a semi-infinite slab of contamination.

879. Parameter values are summarised in Table 119.

880. The activity concentration of radionuclides in soil ($C_{Rn,Soil}$) is determined as the total activity in the soil including water divided by the dry mass of soil.

881. Dose coefficients for irradiation are given in Table 201, except for Pb-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243, which are given in Table 117.

882. The dose from inhalation of contaminated soil is given by (Eden NE, 2015a):

$$Dose_{inh,soil} = B \cdot O_{dust} \cdot C_{Rn,soil}(t) \cdot Dustload \cdot D_{Rn,inh}$$

where:

- B is the breathing rate ($m^3\ h^{-1}$);
- O_{dust} is the fraction of time spent exposed to dust from the soil ($h\ y^{-1}$);
- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide Rn at time t in soil ($Bq\ kg^{-1}$);
- $Dustload$ is the dust concentration in air ($kg\ m^{-3}$); and,
- $D_{Rn,inh}$ is the dose coefficient for inhalation of radionuclide Rn ($Sv\ Bq^{-1}$)

883. Parameter values are summarised in Table 119.

884. The activity concentration of radionuclides in soil ($C_{Rn,Soil}$) is determined as the total activity in the soil including water divided by the dry mass of soil.

885. Dose coefficients for inhalation are given in Table 200, except for Pb-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243, which are given in Table 117.

Element and Radionuclide Specific Parameters

886. Radionuclide specific dose coefficients and with element specific parameters for plant and animal uptake specified in Table 200, Table 201 and Table 203. Note that the dose coefficients for Po-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243 used in the groundwater model are shown in Table 117.

E.4.3.8. Bathtubbing causing Contamination of Farmland

887. Bathtubbing results in leachate spilling over the top of the landfill liner at the sides of the landfill. The release is assumed to affect subsoil in an area of surrounding land, calculated in GoldSim as specified in Subsection E.4.3.5. The GoldSim model is set up to calculate the affected area based on a surface water head that is 10% of the leachate head in the waste cells.

888. A proportion of the release is expected to accumulate in the root zone of plants and the remainder will be draining to groundwater. Recent work at Imperial College on the transfer of radionuclides from a water table to crops considered a range of elements that are of interest to a bathtubbing event provided the basis for the value of 1% (Shaw, et al., 2004). Shaw *et al.* reported the movement of two very mobile radionuclides, Tc-99 and Cl-36 from a water table at 0.7 m depth to the upper soil layers. For Tc-99 the activity in upper soil layers was two orders of magnitude lower than that at the water table and Shaw et al. reported much lower transport of less mobile radionuclides. The study showed Cl-36 with upper soil activity at about 10% of that in the lowest layers but declining with distance above the water table. A value of 1% was therefore adopted as conservative for most radionuclides and probably realistic for Cl-36 with a water table at a depth of greater than 1 m.

889. Exposure of the public is assumed to occur as a result of the use of the contaminated land to grow vegetables. Members of the exposed group are assumed to be adults and to be exposed as a result of:

- consumption of green vegetables and root vegetables produced on contaminated land;
- consumption of fish from a contaminated water body;
- external irradiation from radionuclides incorporated in contaminated soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

890. The overflow leads to contamination of subsoil adjacent to the site. It is assumed that this topsoil is farmland. The model for calculating the doses from the bathtubbing scenario is very similar to the irrigation model, and most parameters values are the same. The main difference is that in the bathtubbing scenario there is no foliar deposition and retention by crops.

Element and Radionuclide Specific Parameters

891. Radionuclide specific dose coefficients and with element specific parameters for plant and animal uptake specified in Table 200, Table 201 and Table 203. Note that the dose coefficients for Po-210, Ra-226, Ra-228, Th-232, U-232, Am-242m and Cm-243 used in the groundwater model are shown in Table 117.

E.4.3.9. Groundwater doses after the Period of Authorisation

892. The construction of a water abstraction borehole downstream of the site is not expected since the groundwater at this point is affected by the saline water of the estuary. Nevertheless, this scenario considers the exposures resulting from water taken from a hypothetical point near the site boundary. The scenario is not used to limit radiological capacity and is provided for information.

893. Exposure of members of the public is assumed to occur as a result of using water for irrigation and drinking water. Doses can result from ingestion of foodstuffs grown on contaminated soil (including pasture supporting grazing livestock), inhalation of dust from the soil, external exposure to the soil and from drinking contaminated water.

894. The dose criterion used is a dose of 0.02 mSv y-1 (this is equivalent to the risk guidance level of 10^{-6} y^{-1} for exposure of the public post closure, for situations that are expected to occur).

895. GoldSim output has a low value cut-off and reports a lower limit of 1 10^{-13} µSv y^{-1} MBq^{-1}, which can occur for short lived radionuclides (half-life of less than about 5 years) where radioactive decay reduces activity to very low levels or where there is limited radionuclide transport in groundwater during the period of active management. The cut-off produces reported values of zero dose for shorter half-life radionuclides.

896. The results of the dose calculations for water at the point 100 m from the site boundary are given in Table 120 for each age group. The radiological capacity for each radionuclide is shown, and the corresponding dose from disposal of that inventory and

the year when the maximum dose occurs, are also shown. The results for Ra-226 are independent of the Ra-226 placement depth in the site. The scenario has not been used to limit the radiological capacity at the site since it is most unlikely to occur and is only provided for information. For this reason, the doses for Cl-36, Ca-41, Tc-99, I-129 and Np-237 are greater than 20 µSv y^{-1}.

897. The results of the bathtubbing scenario are given in Table 121 for each age group. The radiological capacity for each radionuclide is shown, and the corresponding dose from disposal of that inventory and the year when the maximum dose occurs, are also shown. The results for Ra-226 are independent of the Ra-226 placement depth in the site. The scenario has not been used to limit the radiological capacity at the site. For this reason, the doses for Se-79, Sm-147, Pa-231, U-233, U-234, U-235, U-236, U-238, Pu-239, Pu-240, Pu-242, Pu-244, Am-243 and Cm-248 are greater than 20 µSv y^{-1}.

Table 120 Maximum annual doses from groundwater for all age groups for an abstraction point

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
H-3	6.43 10^{9}	0	0	0		0	Adult
C-14	1.87 10^{8}	3.36 10^{-12}	3.08 10^{-12}	3.26 10^{-12}	2.01 10^{4}	6.29 10^{-4}	Adult
Cl-36	1.56 10^{8}	1.04 10^{-5}	1.44 10^{-5}	2.59 10^{-5}	1495	4.05 10^{3}	Infant
Ca-41	5.77 10^{9}	3.41 10^{-8}	5.81 10^{-8}	3.46 10^{-8}	20655	3.35 10^{2}	Child
Mn-54	1.12 10^{13}	0	0	0		0	Adult
Fe-55	1.86 10^{13}	0	0	0		0	Adult
Co-60	3.58 10^{11}	0	0	0		0	Adult
Ni-59	1.95 10^{11}	0	0	0	100000	0	Adult
Ni-63	2.42 10^{11}	0	0	0	668	0	Adult
Zn-65	8.95 10^{11}	0	0	0		0	Adult
Se-79	8.98 10^{8}	6.46 10^{-10}	2.13 10^{-9}	2.93 10^{-9}	100000	2.64 10^{0}	Infant
Sr-90	3.83 10^{8}	0	0	0		0	Adult
Mo-93	1.44 10^{9}	4.79 10^{-11}	4.66 10^{-11}	5.93 10^{-11}	16635	8.53 10^{-2}	Infant
Zr-93	3.12 10^{11}	0	0	0	100000	0	Adult
Nb-93m	5.06 10^{10}	0	0	0	151	0	Adult
Nb-94	6.09 10^{6}	0	0	0	100000	0	Adult
Tc-99	6.12 10^{8}	1.02 10^{-5}	1.37 10^{-5}	2.51 10^{-5}	1085	1.53 10^{4}	Infant
Ru-106	9.14 10^{11}	0	0	0		0	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
Ag-108m	$2.65\ 10^{8}$	0	0	0		0	Adult
Ag-110m	$6.41\ 10^{12}$	0	0	0		0	Adult
Cd-109	$1.04\ 10^{12}$	0	0	0		0	Adult
Sb-125	$4.17\ 10^{11}$	0	0	0		0	Adult
Sn-119m	$8.43\ 10^{12}$	0	0	0		0	Adult
Sn-123	$2.97\ 10^{12}$	0	0	0		0	Adult
Sn-126	$4.60\ 10^{6}$	0	0	0		0	Adult
Te-127m	$4.07\ 10^{12}$	0	0	0		0	Adult
I-129	$3.01\ 10^{8}$	$2.39\ 10^{-5}$	$2.76\ 10^{-5}$	$1.59\ 10^{-5}$	19785	$8.31\ 10^{3}$	Child
Ba-133	$7.18\ 10^{9}$	0	0	0		0	Adult
Cs-134	$1.01\ 10^{11}$	0	0	0		0	Adult
Cs-135	$1.55\ 10^{9}$	0	0	0		0	Adult
Cs-137	$9.69\ 10^{8}$	0	0	0		0	Adult
Ce-144	$4.81\ 10^{12}$	0	0	0		0	Adult
Pm-147	$2.14\ 10^{13}$	0	0	0		0	Adult
Sm-147	$4.81\ 10^{8}$	0	0	0		0	Adult
Sm-151	$7.23\ 10^{11}$	0	0	0		0	Adult
Eu-152	$8.05\ 10^{9}$	0	0	0		0	Adult
Eu-154	$4.18\ 10^{10}$	0	0	0		0	Adult
Eu-155	$8.81\ 10^{12}$	0	0	0		0	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
Gd-153	$4.83\ 10^{13}$	0	0	0		0	Adult
Pb-210	$4.85\ 10^{8}$	0	0	0		0	Adult
Po-210	$6.17\ 10^{9}$	0	0	0		0	Adult
Ra-226	$3.89\ 10^{6}$	0	0	0		0	Adult
Ra-228	$2.25\ 10^{10}$	0	0	0		0	Adult
Ac-227	$3.04\ 10^{9}$	0	0	0		0	Adult
Th-228	$1.72\ 10^{11}$	0	0	0		0	Adult
Th-229	$2.88\ 10^{7}$	0	0	0		0	Adult
Th-230	$1.98\ 10^{6}$	0	0	0		0	Adult
Th-232	$7.95\ 10^{6}$	0	0	0		0	Adult
Pa-231	$1.36\ 10^{7}$	0	0	0		0	Adult
U-232	$4.04\ 10^{8}$	0	0	0		0	Adult
U-233	$1.02\ 10^{8}$	$3.39\ 10^{-9}$	$3.42\ 10^{-9}$	$3.09\ 10^{-9}$	100000	$3.50\ 10^{-1}$	Child
U-234	$1.45\ 10^{8}$	$9.53\ 10^{-9}$	$9.42\ 10^{-9}$	$8.21\ 10^{-9}$	100000	$1.38\ 10^{0}$	Adult
U-235	$6.93\ 10^{7}$	$7.63\ 10^{-9}$	$7.52\ 10^{-9}$	$6.85\ 10^{-9}$	100000	$5.29\ 10^{-1}$	Adult
U-236	$1.48\ 10^{9}$	$2.57\ 10^{-9}$	$2.54\ 10^{-9}$	$2.38\ 10^{-9}$	100000	$3.81\ 10^{0}$	Adult
U-238	$1.60\ 10^{9}$	$5.38\ 10^{-9}$	$5.52\ 10^{-9}$	$5.31\ 10^{-9}$	100000	$8.86\ 10^{0}$	Child
Np-237	$1.42\ 10^{7}$	$3.88\ 10^{-6}$	$2.61\ 10^{-6}$	$2.50\ 10^{-6}$	81800	$5.51\ 10^{1}$	Adult
Pu-238	$7.56\ 10^{8}$	$3.27\ 10^{-12}$	$2.22\ 10^{-12}$	$1.81\ 10^{-12}$	597	$2.47\ 10^{-3}$	Adult
Pu-239	$1.55\ 10^{8}$	0	0	0		0	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
Pu-240	$1.89\ 10^{8}$	$3.53\ 10^{-13}$	$2.78\ 10^{-13}$	$2.05\ 10^{-13}$	36335	$6.66\ 10^{-5}$	Adult
Pu-241	$9.39\ 10^{9}$	$2.38\ 10^{-11}$	$1.69\ 10^{-11}$	$9.35\ 10^{-12}$	124	$2.24\ 10^{-1}$	Adult
Pu-242	$1.58\ 10^{8}$	0	0	0	100000	0	Adult
Pu-244	$1.26\ 10^{8}$	$8.98\ 10^{-13}$	$6.95\ 10^{-13}$	$5.58\ 10^{-13}$	100000	$1.13\ 10^{-4}$	Adult
Am-241	$3.03\ 10^{8}$	$7.23\ 10^{-10}$	$5.30\ 10^{-10}$	$4.42\ 10^{-10}$	2585	$2.19\ 10^{-1}$	Adult
Am-242m	$1.75\ 10^{7}$	$4.60\ 10^{-12}$	$3.29\ 10^{-12}$	$2.64\ 10^{-12}$	968	$8.04\ 10^{-5}$	Adult
Am-243	$1.46\ 10^{8}$	0	0	0		0	Adult
Cm-242	$1.48\ 10^{11}$	0	0	0		0	Adult
Cm-243	$4.89\ 10^{7}$	0	0	0		0	Adult
Cm-244	$1.16\ 10^{8}$	0	0	0		0	Adult
Cm-245	$1.26\ 10^{7}$	$1.28\ 10^{-8}$	$9.33\ 10^{-9}$	$7.45\ 10^{-9}$	48880	$1.61\ 10^{-1}$	Adult
Cm-246	$1.27\ 10^{7}$	0	0	0		0	Adult
Cm-248	$1.45\ 10^{7}$	0	0	0		0	Adult

Table 121 Maximum annual doses for all age groups from bathtubbing

Radionuclide	Radiological Capacity (MBq)	Adult dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of max (y)	Dose at Radiological Capacity ($\mu Sv\ y^{-1}$)	Age group with highest dose
H-3	$6.43\ 10^{9}$	$2.47\ 10^{-13}$	$1.35\ 10^{-13}$	$9.42\ 10^{-14}$	130	$1.59\ 10^{-3}$	Adult
C-14	$1.87\ 10^{8}$	$2.67\ 10^{-10}$	$1.55\ 10^{-10}$	$1.38\ 10^{-10}$	1330	$5.01\ 10^{-2}$	Adult
Cl-36	$1.56\ 10^{8}$	$2.20\ 10^{-8}$	$1.92\ 10^{-8}$	$2.13\ 10^{-8}$	133	$3.44\ 10^{0}$	Adult
Ca-41	$5.77\ 10^{9}$	$5.51\ 10^{-10}$	$5.95\ 10^{-10}$	$2.16\ 10^{-10}$	213	$3.43\ 10^{0}$	Child
Mn-54	$1.12\ 10^{13}$	0	0	0	0	0	Adult
Fe-55	$1.86\ 10^{13}$	0	0	0	0	0	Adult
Co-60	$3.58\ 10^{11}$	0	0	0	0	0	Adult
Ni-59	$1.95\ 10^{11}$	$1.07\ 10^{-11}$	$7.55\ 10^{-12}$	$1.13\ 10^{-11}$	2680	$2.21\ 10^{0}$	Infant
Ni-63	$2.42\ 10^{11}$	$9.80\ 10^{-13}$	$7.42\ 10^{-13}$	$1.13\ 10^{-12}$	250	$2.74\ 10^{-1}$	Infant
Zn-65	$8.95\ 10^{11}$	0	0	0	0	0	Adult
Se-79	$8.98\ 10^{8}$	$1.56\ 10^{-8}$	$3.23\ 10^{-8}$	$2.16\ 10^{-8}$	2155	$2.90\ 10^{1}$	Child
Sr-90	$3.83\ 10^{8}$	$7.89\ 10^{-10}$	$7.23\ 10^{-10}$	$3.47\ 10^{-10}$	163	$3.02\ 10^{-1}$	Adult
Mo-93	$1.44\ 10^{9}$	$7.05\ 10^{-9}$	$3.89\ 10^{-9}$	$2.25\ 10^{-9}$	250	$1.01\ 10^{1}$	Adult
Zr-93	$3.12\ 10^{11}$	$3.47\ 10^{-11}$	$5.67\ 10^{-12}$	$1.77\ 10^{-11}$	3730	$1.08\ 10^{1}$	Adult
Nb-93m	$5.06\ 10^{10}$	$1.45\ 10^{-16}$	$1.40\ 10^{-16}$	$1.58\ 10^{-16}$	151	$8.00\ 10^{-6}$	Infant
Nb-94	$6.09\ 10^{6}$	$1.95\ 10^{-7}$	$1.46\ 10^{-7}$	$1.09\ 10^{-7}$	7270	$1.19\ 10^{0}$	Adult
Tc-99	$6.12\ 10^{8}$	$2.90\ 10^{-8}$	$2.52\ 10^{-8}$	$3.11\ 10^{-8}$	133	$1.91\ 10^{1}$	Infant
Ru-106	$9.14\ 10^{11}$	0	0	0	0	0	Adult
Ag-108m	$2.65\ 10^{8}$	$2.88\ 10^{-8}$	$2.16\ 10^{-8}$	$1.60\ 10^{-8}$	535	$7.64\ 10^{0}$	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
Ag-110m	$6.41\ 10^{12}$	0	0	0	0	0	Adult
Cd-109	$1.04\ 10^{12}$	0	0	0	0	0	Adult
Sb-125	$4.17\ 10^{11}$	0	0	0	0	0	Adult
Sn-119m	$8.43\ 10^{12}$	0	0	0	0	0	Adult
Sn-123	$2.97\ 10^{12}$	0	0	0	0	0	Adult
Sn-126	$4.60\ 10^{6}$	$3.14\ 10^{-7}$	$2.36\ 10^{-7}$	$1.76\ 10^{-7}$	11780	$1.44\ 10^{0}$	Adult
Te-127m	$4.07\ 10^{12}$	0	0	0	0	0	Adult
I-129	$3.01\ 10^{8}$	$5.62\ 10^{-9}$	$3.35\ 10^{-9}$	$1.62\ 10^{-9}$	199	$1.69\ 10^{0}$	Adult
Ba-133	$7.18\ 10^{9}$	$8.71\ 10^{-12}$	$6.53\ 10^{-12}$	$4.85\ 10^{-12}$	131	$6.25\ 10^{-2}$	Adult
Cs-134	$1.01\ 10^{11}$	0	0	0	0	0	Adult
Cs-135	$1.55\ 10^{9}$	$6.71\ 10^{-10}$	$2.38\ 10^{-10}$	$1.65\ 10^{-10}$	9710	$1.04\ 10^{0}$	Adult
Cs-137	$9.69\ 10^{8}$	$3.73\ 10^{-11}$	$2.72\ 10^{-11}$	$2.01\ 10^{-11}$	172	$3.61\ 10^{-2}$	Adult
Ce-144	$4.81\ 10^{12}$	0	0	0	0	0	Adult
Pm-147	$2.14\ 10^{13}$	0	0	0	0	0	Adult
Sm-147	$4.81\ 10^{8}$	$1.23\ 10^{-9}$	$6.05\ 10^{-10}$	$5.56\ 10^{-8}$	7710	$2.67\ 10^{1}$	Infant
Sm-151	$7.23\ 10^{11}$	$2.13\ 10^{-14}$	$1.09\ 10^{-14}$	$2.53\ 10^{-13}$	250	$1.83\ 10^{-1}$	Infant
Eu-152	$8.05\ 10^{9}$	$4.49\ 10^{-12}$	$3.37\ 10^{-12}$	$2.50\ 10^{-12}$	147	$3.62\ 10^{-2}$	Adult
Eu-154	$4.18\ 10^{10}$	$7.69\ 10^{-14}$	$5.77\ 10^{-14}$	$4.28\ 10^{-14}$	140	$3.22\ 10^{-3}$	Adult
Eu-155	$8.81\ 10^{12}$	0	0	0	0	0	Adult
Gd-153	$4.83\ 10^{13}$	0	0	0	0	0	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of max (y)	Dose at Radiological Capacity ($\mu Sv\ y^{-1}$)	Age group with highest dose
Pb-210	$4.85\ 10^{8}$	$2.68\ 10^{-11}$	$2.59\ 10^{-11}$	$3.16\ 10^{-11}$	160	$1.53\ 10^{-2}$	Infant
Po-210	$6.17\ 10^{9}$	$4.44\ 10^{-12}$	$3.85\ 10^{-12}$	$7.60\ 10^{-12}$	0	$4.69\ 10^{-2}$	Infant
Ra-226	$3.89\ 10^{6}$	$9.66\ 10^{-8}$	$8.80\ 10^{-8}$	$8.88\ 10^{-8}$	2450	$3.76\ 10^{-1}$	Adult
Ra-228	$2.25\ 10^{10}$	$2.59\ 10^{-13}$	$5.38\ 10^{-13}$	$4.93\ 10^{-13}$	136	$1.21\ 10^{-2}$	Child
Ac-227	$3.04\ 10^{9}$	$1.37\ 10^{-11}$	$9.52\ 10^{-12}$	$2.61\ 10^{-10}$	160	$7.94\ 10^{-1}$	Infant
Th-228	$1.72\ 10^{11}$	$4.52\ 10^{-14}$	$5.30\ 10^{-14}$	$8.05\ 10^{-14}$	0	$1.38\ 10^{-2}$	Infant
Th-229	$2.88\ 10^{7}$	$2.75\ 10^{-8}$	$2.06\ 10^{-8}$	$6.31\ 10^{-7}$	5520	$1.82\ 10^{1}$	Infant
Th-230	$1.98\ 10^{6}$	$7.34\ 10^{-7}$	$4.23\ 10^{-7}$	$7.85\ 10^{-7}$	11795	$1.55\ 10^{0}$	Infant
Th-232	$7.95\ 10^{6}$	$8.01\ 10^{-7}$	$9.77\ 10^{-7}$	$1.61\ 10^{-6}$	15150	$1.28\ 10^{1}$	Infant
Pa-231	$1.36\ 10^{7}$	$1.90\ 10^{-7}$	$1.14\ 10^{-7}$	$2.22\ 10^{-6}$	10060	$3.01\ 10^{1}$	Infant
U-232	$4.04\ 10^{8}$	$1.78\ 10^{-8}$	$1.34\ 10^{-8}$	$2.51\ 10^{-8}$	216	$1.01\ 10^{1}$	Infant
U-233	$1.02\ 10^{8}$	$3.36\ 10^{-8}$	$2.46\ 10^{-8}$	$1.02\ 10^{-6}$	2145	$1.05\ 10^{2}$	Infant
U-234	$1.45\ 10^{8}$	$8.65\ 10^{-8}$	$5.81\ 10^{-8}$	$2.00\ 10^{-7}$	2155	$2.90\ 10^{1}$	Infant
U-235	$6.93\ 10^{7}$	$7.40\ 10^{-8}$	$5.18\ 10^{-8}$	$1.19\ 10^{-6}$	2170	$8.22\ 10^{1}$	Infant
U-236	$1.48\ 10^{9}$	$5.74\ 10^{-9}$	$3.56\ 10^{-9}$	$6.67\ 10^{-8}$	2170	$9.87\ 10^{1}$	Infant
U-238	$1.60\ 10^{9}$	$1.08\ 10^{-8}$	$7.48\ 10^{-9}$	$7.01\ 10^{-8}$	2170	$1.12\ 10^{2}$	Infant
Np-237	$1.42\ 10^{7}$	$4.45\ 10^{-8}$	$2.84\ 10^{-8}$	$2.31\ 10^{-7}$	729	$3.28\ 10^{0}$	Infant
Pu-238	$7.56\ 10^{8}$	$8.78\ 10^{-11}$	$5.36\ 10^{-11}$	$5.52\ 10^{-9}$	249	$4.17\ 10^{0}$	Infant
Pu-239	$1.55\ 10^{8}$	$4.62\ 10^{-9}$	$2.91\ 10^{-9}$	$4.12\ 10^{-7}$	4780	$6.41\ 10^{1}$	Infant
Pu-240	$1.89\ 10^{8}$	$3.40\ 10^{-9}$	$2.14\ 10^{-9}$	$3.04\ 10^{-7}$	3415	$5.74\ 10^{1}$	Infant

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Time of max (y)	Dose at Radiological Capacity (µSv y^{-1})	Age group with highest dose
Pu-241	$9.39\ 10^{9}$	$4.12\ 10^{-12}$	$2.76\ 10^{-12}$	$2.00\ 10^{-10}$	149	$1.88\ 10^{0}$	Infant
Pu-242	$1.58\ 10^{8}$	$4.99\ 10^{-9}$	$3.36\ 10^{-9}$	$4.52\ 10^{-7}$	6100	$7.16\ 10^{1}$	Infant
Pu-244	$1.26\ 10^{8}$	$6.37\ 10^{-8}$	$4.73\ 10^{-8}$	$8.46\ 10^{-7}$	6255	$1.06\ 10^{2}$	Infant
Am-241	$3.03\ 10^{8}$	$1.20\ 10^{-10}$	$8.04\ 10^{-11}$	$8.64\ 10^{-9}$	662	$2.61\ 10^{0}$	Infant
Am-242m	$1.75\ 10^{7}$	$2.06\ 10^{-10}$	$1.35\ 10^{-10}$	$1.42\ 10^{-8}$	250	$2.48\ 10^{-1}$	Infant
Am-243	$1.46\ 10^{8}$	$1.23\ 10^{-8}$	$9.04\ 10^{-9}$	$2.54\ 10^{-7}$	6270	$3.71\ 10^{1}$	Infant
Cm-242	$1.48\ 10^{11}$	$4.55\ 10^{-13}$	$3.91\ 10^{-13}$	$5.80\ 10^{-11}$	0	$8.59\ 10^{0}$	Infant
Cm-243	$4.89\ 10^{7}$	$6.47\ 10^{-12}$	$4.42\ 10^{-12}$	$6.66\ 10^{-10}$	170	$3.26\ 10^{-2}$	Infant
Cm-244	$1.16\ 10^{8}$	$9.41\ 10^{-12}$	$6.35\ 10^{-12}$	$1.15\ 10^{-9}$	154	$1.34\ 10^{-1}$	Infant
Cm-245	$1.26\ 10^{7}$	$4.29\ 10^{-9}$	$2.90\ 10^{-9}$	$1.84\ 10^{-7}$	9930	$2.32\ 10^{0}$	Infant
Cm-246	$1.27\ 10^{7}$	$7.25\ 10^{-10}$	$4.33\ 10^{-10}$	$5.46\ 10^{-8}$	6355	$6.92\ 10^{-1}$	Infant
Cm-248	$1.45\ 10^{7}$	$1.93\ 10^{-8}$	$1.15\ 10^{-8}$	$1.38\ 10^{-6}$	54615	$2.01\ 10^{1}$	Infant

E.4.4. Groundwater release to the estuary

898. The Groundwater model developed in GoldSim (Section E.4.3) has been extended to calculate release rates of radionuclides into the estuary by placing the point at which activity concentrations in groundwater are calculated at 280 m from the site boundary. The activity release rate to the estuary is then calculated as the groundwater flow rate at this point multiplied by the activity concentration in groundwater at this point. Table 122 presents the calculated peak activity concentrations released to the estuary. Activity concentrations of daughters released to the estuary were also extracted from GoldSim and used as input, but are not shown in Table 122.

Table 122 Peak activity released to the estuary

Radionuclide	Leachate activity concentration (Bq m^{-3}/MBq)	Flux to estuary (Bq y^{-1}/MBq)
H-3	$2.36\ 10^{-2}$	$1.90\ 10^{-2}$
C-14	$2.21\ 10^{-3}$	$1.55\ 10^{0}$
Cl-36	$4.35\ 10^{-1}$	$6.56\ 10^{2}$
Ca-41	$2.72\ 10^{-2}$	$5.09\ 10^{1}$
Mn-54	$4.97\ 10^{-15}$	0
Fe-55	$7.92\ 10^{-11}$	0
Co-60	$1.74\ 10^{-7}$	0
Ni-59	$7.96\ 10^{-4}$	$1.48\ 10^{0}$
Ni-63	$5.22\ 10^{-4}$	$9.01\ 10^{-15}$
Zn-65	$3.54\ 10^{-15}$	0
Se-79	$1.11\ 10^{-3}$	$2.39\ 10^{0}$
Sr-90	$9.89\ 10^{-4}$	$1.49\ 10^{-14}$
Mo-93	$5.49\ 10^{-3}$	$5.80\ 10^{0}$
Zr-93	$5.44\ 10^{-4}$	$1.20\ 10^{0}$
Nb-93m	$1.10\ 10^{-5}$	0
Nb-94	$1.49\ 10^{-4}$	$7.28\ 10^{-3}$
Tc-99	$5.03\ 10^{-1}$	$7.58\ 10^{2}$
Ru-106	$5.91\ 10^{-15}$	0
Ag-108m	$5.31\ 10^{-4}$	$1.49\ 10^{-9}$
Ag-110m	$6.13\ 10^{-15}$	0
Cd-109	$1.16\ 10^{-14}$	0
Sb-125	$1.26\ 10^{-9}$	0
Sn-119m	$1.87\ 10^{-15}$	0
Sn-123	$1.16\ 10^{-15}$	0
Sn-126	$1.39\ 10^{-4}$	$1.23\ 10^{-1}$

Radionuclide	Leachate activity concentration (Bq m^{-3}/MBq)	Flux to estuary (Bq y^{-1}/MBq)
Te-127m	$3.41\ 10^{-15}$	0
I-129	$3.14\ 10^{-2}$	$5.77\ 10^{1}$
Ba-133	$6.76\ 10^{-3}$	$1.41\ 10^{-4}$
Cs-134	$5.38\ 10^{-13}$	0
Cs-135	$1.86\ 10^{-4}$	$3.69\ 10^{-1}$
Cs-137	$4.60\ 10^{-5}$	0
Ce-144	$1.25\ 10^{-15}$	0
Pm-147	$8.25\ 10^{-11}$	0
Sm-147	$2.40\ 10^{-4}$	$5.41\ 10^{-1}$
Sm-151	$1.50\ 10^{-4}$	$2.13\ 10^{-23}$
Eu-152	$4.17\ 10^{-5}$	0
Eu-154	$7.12\ 10^{-6}$	0
Eu-155	$1.51\ 10^{-7}$	0
Gd-153	$5.77\ 10^{-15}$	0
Pb-210	$1.67\ 10^{-5}$	0
Po-210	$8.00\ 10^{-15}$	0
Ra-226	$8.70\ 10^{-5}$	$1.17\ 10^{-12}$
Ra-228	$6.37\ 10^{-8}$	0
Ac-227	$1.90\ 10^{-5}$	0
Th-228	$7.61\ 10^{-14}$	0
Th-229	$1.17\ 10^{-4}$	$1.88\ 10^{-5}$
Th-230	$1.17\ 10^{-4}$	$3.18\ 10^{-2}$
Th-232	$1.17\ 10^{-4}$	$7.96\ 10^{-2}$
Pa-231	$1.11\ 10^{-4}$	$7.44\ 10^{-3}$
U-232	$6.04\ 10^{-4}$	$2.79\ 10^{-15}$
U-233	$1.11\ 10^{-3}$	$2.30\ 10^{0}$
U-234	$1.11\ 10^{-3}$	$2.37\ 10^{0}$
U-235	$1.12\ 10^{-3}$	$2.51\ 10^{0}$
U-236	$1.12\ 10^{-3}$	$2.51\ 10^{0}$
U-238	$1.12\ 10^{-3}$	$2.51\ 10^{0}$
Np-237	$6.34\ 10^{-3}$	$1.40\ 10^{1}$
Pu-238	$1.86\ 10^{-4}$	$7.77\ 10^{-23}$
Pu-239	$3.01\ 10^{-4}$	$1.08\ 10^{-1}$
Pu-240	$3.00\ 10^{-4}$	$3.24\ 10^{-3}$
Pu-241	$1.61\ 10^{-5}$	0
Pu-242	$3.02\ 10^{-4}$	$5.93\ 10^{-1}$

Radionuclide	Leachate activity concentration (Bq m^{-3}/MBq)	Flux to estuary (Bq y^{-1}/MBq)
Pu-244	$3.02\ 10^{-4}$	$6.79\ 10^{-1}$
Am-241	$7.79\ 10^{-5}$	$1.98\ 10^{-20}$
Am-242m	$6.50\ 10^{-5}$	0
Am-243	$8.54\ 10^{-5}$	$1.73\ 10^{-6}$
Cm-242	0	0
Cm-243	$5.64\ 10^{-6}$	0
Cm-244	$2.35\ 10^{-6}$	0
Cm-245	$2.39\ 10^{-5}$	$1.23\ 10^{-11}$
Cm-246	$2.38\ 10^{-5}$	$2.45\ 10^{-14}$
Cm-248	$2.40\ 10^{-5}$	$3.49\ 10^{-8}$

899. The estuary was modelled in PC CREAM as a local marine compartment, representative of the area of the estuary shown in Figure 20. The length of ‘coastline’ of the compartment was taken to be 12,000 m, equal to the length of both sides of the estuary added together. The width of the estuary was taken to be 400 m, based on measurements on a map, and the depth was taken to be 15 m (Le Guillou, 1978). The change in height with the tide, 3.4 m (https://riverlevels.uk/north-yorkshire-tees-dock-tidal#.XSRfruhKhPa), was used to derive a volumetric exchange rate of 5.96 10^9 m^3 y^{-1}, assuming that the tide rises and falls twice per day. All other local compartment values within PC CREAM were set to default values.

Figure 20 Estuary local compartment

900. The DORIS module within PC CREAM was used to derive activity concentrations in the estuary compartment and the estuary bed. The output from DORIS was used as input to the ASSESSOR module within PC CREAM to calculate doses to a fishing family that collects seafood from the estuary and consumes it. The habits data for this family are presented in Table 123.

Table 123 Habits data for a fishing family exposed to activity in the estuary from groundwater

	Adult	Child	Infant
Fish consumption (kg y^{-1})	100	20	5
Crustacean consumption (kg y^{-1})	20	5	0
Mollusc consumption (kg y^{-1})	20	5	0
Inhalation rate (m^3 y^{-1})	8100	5600	1900

	Adult	Child	Infant
Time spent near the estuary (h y^{-1})	2000	2000	2000
Distance from the estuary (m)	100	100	100

Table 124 Doses to a fishing family from groundwater flow to estuary

Radionuclide	Radiolgical Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Dose at Radiological Capacity (µSv y^{-1})	Limiting age group
H-3	$6.43\ 10^{9}$	$2.88\ 10^{-18}$	$8.82\ 10^{-19}$	$7.74\ 10^{-20}$	$1.85\ 10^{-8}$	Adult
C-14	$1.87\ 10^{8}$	$1.08\ 10^{-10}$	$3.57\ 10^{-11}$	$2.99\ 10^{-12}$	$2.02\ 10^{-2}$	Adult
Cl-36	$1.56\ 10^{8}$	$6.57\ 10^{-13}$	$4.65\ 10^{-13}$	$3.56\ 10^{-13}$	$1.03\ 10^{-4}$	Adult
Ca-41	$5.77\ 10^{9}$	$4.89\ 10^{-13}$	$4.06\ 10^{-13}$	$2.83\ 10^{-13}$	$2.82\ 10^{-3}$	Adult
Mn-54	$1.12\ 10^{13}$	0	0	0	0	
Fe-55	$1.86\ 10^{13}$	0	0	0	0	
Co-60	$3.58\ 10^{11}$	0	0	0	0	
Ni-59	$1.95\ 10^{11}$	$5.68\ 10^{-13}$	$5.32\ 10^{-13}$	$5.12\ 10^{-13}$	$1.11\ 10^{-1}$	Adult
Ni-63	$2.42\ 10^{11}$	$7.60\ 10^{-28}$	$3.45\ 10^{-28}$	$3.13\ 10^{-29}$	$1.84\ 10^{-16}$	Adult
Zn-65	$8.95\ 10^{11}$	0	0	0	0	
Se-79	$8.98\ 10^{8}$	0	0	0	0	
Sr-90	$3.83\ 10^{8}$	0	0	0	0	
Mo-93	$1.44\ 10^{9}$	$3.12\ 10^{-10}$	$1.01\ 10^{-10}$	$2.16\ 10^{-13}$	$4.49\ 10^{-1}$	Adult
Zr-93	$3.12\ 10^{11}$	0	0	0	0	
Nb-93m	$5.06\ 10^{10}$	0	0	0	0	
Nb-94	$6.09\ 10^{6}$	$1.73\ 10^{-12}$	$1.71\ 10^{-12}$	$1.71\ 10^{-12}$	$1.05\ 10^{-5}$	Adult
Tc-99	$6.12\ 10^{8}$	0	0	0	0	
Ru-106	$9.14\ 10^{11}$	0	0	0	0	
Ag-108m	$2.65\ 10^{8}$	$2.00\ 10^{-19}$	$1.22\ 10^{-19}$	$5.42\ 10^{-20}$	$5.32\ 10^{-11}$	Adult
Ag-110m	$6.41\ 10^{12}$	0	0	0	0	
Cd-109	$1.04\ 10^{12}$	0	0	0	0	
Sb-125	$4.17\ 10^{11}$	0	0	0	0	
Sn-119m	$8.43\ 10^{12}$	0	0	0	0	
Sn-123	$2.97\ 10^{12}$	0	0	0	0	
Sn-126	$4.60\ 10^{6}$	0	0	0	0	
Te-127m	$4.07\ 10^{12}$	0	0	0	0	
I-129	$3.01\ 10^{8}$	$5.31\ 10^{-10}$	$2.21\ 10^{-10}$	$1.16\ 10^{-11}$	$1.60\ 10^{-1}$	Adult
Ba-133	$7.18\ 10^{9}$	$1.92\ 10^{-15}$	$1.90\ 10^{-15}$	$1.89\ 10^{-15}$	$1.38\ 10^{-5}$	Adult

Radionuclide	Radiolgical Capacity (MBq)	Adult dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose at Radiological Capacity ($\mu Sv\ y^{-1}$)	Limiting age group
Cs-134	$1.01\ 10^{11}$	0	0	0	0	
Cs-135	$1.55\ 10^{9}$	$1.96\ 10^{-13}$	$4.50\ 10^{-14}$	$1.64\ 10^{-14}$	$3.04\ 10^{-4}$	Adult
Cs-137	$9.69\ 10^{8}$	0	0	0	0	
Ce-144	$4.81\ 10^{12}$	0	0	0	0	
Pm-147	$2.14\ 10^{13}$	0	0	0	0	
Sm-147	$4.81\ 10^{8}$	0	0	0	0	
Sm-151	$7.23\ 10^{11}$	0	0	0	0	
Eu-152	$8.05\ 10^{9}$	0	0	0	0	
Eu-154	$4.18\ 10^{10}$	0	0	0	0	
Eu-155	$8.81\ 10^{12}$	0	0	0	0	
Gd-153	$4.83\ 10^{13}$	0	0	0	0	
Pb-210	$4.85\ 10^{8}$	0	0	0	0	
Po-210	$6.17\ 10^{9}$	0	0	0	0	
Ra-226	$3.89\ 10^{6}$	0	0	0	0	
Ra-228	$2.25\ 10^{10}$	0	0	0	0	
Ac-227	$3.04\ 10^{9}$	0	0	0	0	
Th-228	$1.72\ 10^{11}$	0	0	0	0	
Th-229	$2.88\ 10^{7}$	0	0	0	0	
Th-230	$1.98\ 10^{6}$	0	0	0	0	
Th-232	$7.95\ 10^{6}$	0	0	0	0	
Pa-231	$1.36\ 10^{7}$	$2.83\ 10^{-12}$	$2.84\ 10^{-12}$	$8.09\ 10^{-13}$	$3.85\ 10^{-5}$	Child
U-232	$4.04\ 10^{8}$	0	0	0	0	
U-233	$1.02\ 10^{8}$	0	0	0	0	
U-234	$1.45\ 10^{8}$	0	0	0	0	
U-235	$6.93\ 10^{7}$	0	0	0	0	
U-236	$1.48\ 10^{9}$	0	0	0	0	
U-238	$1.60\ 10^{9}$	0	0	0	0	
Np-237	$1.42\ 10^{7}$	$2.27\ 10^{-9}$	$6.30\ 10^{-10}$	$8.60\ 10^{-11}$	$3.22\ 10^{-2}$	Adult
Pu-238	$7.56\ 10^{8}$	$1.49\ 10^{-11}$	$7.83\ 10^{-12}$	$7.82\ 10^{-13}$	$1.12\ 10^{-2}$	Adult
Pu-239	$1.55\ 10^{8}$	$1.41\ 10^{-11}$	$3.82\ 10^{-12}$	$4.36\ 10^{-14}$	$2.19\ 10^{-3}$	Adult
Pu-240	$1.89\ 10^{8}$	0	0	0	0	
Pu-241	$9.39\ 10^{9}$	0	0	0	0	
Pu-242	$1.58\ 10^{8}$	0	0	0	0	
Pu-244	$1.26\ 10^{8}$	0	0	0	0	
Am-241	$3.03\ 10^{8}$	$4.39\ 10^{-13}$	$1.22\ 10^{-13}$	$1.66\ 10^{-14}$	$1.33\ 10^{-4}$	Adult

Radionuclide	Radiolgical Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Dose at Radiological Capacity (µSv y^{-1})	Limiting age group
Am-242m	$1.75\ 10^{7}$	$2.13\ 10^{-11}$	$1.12\ 10^{-11}$	$1.12\ 10^{-12}$	$3.72\ 10^{-4}$	Adult
Am-243	$1.46\ 10^{8}$	$4.42\ 10^{-12}$	$1.20\ 10^{-12}$	$1.37\ 10^{-14}$	$6.45\ 10^{-4}$	Adult
Cm-242	$1.48\ 10^{11}$	$7.55\ 10^{-14}$	$3.98\ 10^{-14}$	$3.97\ 10^{-15}$	$1.12\ 10^{-2}$	Adult
Cm-243	$4.89\ 10^{7}$	$1.70\ 10^{-14}$	$4.61\ 10^{-15}$	$5.27\ 10^{-17}$	$8.32\ 10^{-7}$	Adult
Cm-244	$1.16\ 10^{8}$	$1.18\ 10^{-15}$	$3.23\ 10^{-16}$	$5.85\ 10^{-18}$	$1.37\ 10^{-7}$	Adult
Cm-245	$1.26\ 10^{7}$	$2.34\ 10^{-12}$	$6.51\ 10^{-13}$	$8.92\ 10^{-14}$	$2.95\ 10^{-5}$	Adult
Cm-246	$1.27\ 10^{7}$	$9.20\ 10^{-13}$	$2.51\ 10^{-13}$	$4.25\ 10^{-15}$	$1.17\ 10^{-5}$	Adult
Cm-248	$1.45\ 10^{7}$	$8.98\ 10^{-14}$	$3.56\ 10^{-14}$	$1.57\ 10^{-14}$	$1.30\ 10^{-6}$	Adult

901. The doses calculated using illustrative inventories are considered further in Appendix D.

E.4.5. Exposure of coastal walker following site erosion

902. Although it is considered unlikely to occur, erosion of the landfill has been assessed using cautious assumptions. It is assumed that erosion starts about 2540 years after closure.

903. The landfill site has been reclaimed from salt marsh and mudflats over many decades through the deposition of wastes, clinker and slag deposits from industries including gas works, lime works, chlorine works, soda works, blast furnaces and salt evaporating pans (Augean, 2014). These materials are not readily eroded in a low energy estuary environment. The deposited materials have created a land mass that is 2.5 m or more above the tidal reach. The landfill restoration profile rises above the surrounding plain and in the existing plan there are two waste cells (in the north west of the site) that overlap with the projected flood level used for planning purposes (see Section 2.3 and Figure 10).

904. The Tees Estuary has been identified as a buried valley with a depth of 20 to 30 m beneath the site.

905. It is possible that local or national policies for maintaining shipping access and management of local flood defence schemes could change and impact the future evolution of the estuary. If dredging activities stopped there would be accumulation of sediments and development of salt marshes and mudflats in the estuary. The sediment deposits and sea level rise could lead to tidal erosion at the Port Clarence site from the seaward side.

906. Radiation exposure of members of the public spending time at the site once erosion of the landfills has started could occur. Two exposure scenarios are considered; exposure through direct irradiation to a casual user who walks close to the exposed

waste (e.g. on the estuary bank); and, exposure from releases into the estuary (see Section E.4.6).

907. The dose criterion used is a dose of 0.02 mSv y^{-1} for the public (this is equivalent to the risk guidance level of 10^{-6} y^{-1} for exposure of the public post closure, for situations that are expected to occur).

Potentially exposed group

908. The intended end use of the site includes public access to scrub and grassland with paths. An assessment is therefore made of the doses to a member of the public who spends time walking over the restored site and it is assumed that this continues once erosion starts to impact the site even though erosion may restrict access to the site. Time spent close to and walking over the eroding materials is calculated assuming a daily walk of 1 hour, passing the exposed face once, assuming a face length of 1 km and walking at 5 km h^{-1} (about 73 h y^{-1}). The walker inadvertently ingests soil, inhales dust and receives an external exposure from exposed waste and it is cautiously assumed that all three age groups walk together. The habit data are summarized in Table 125.

Table 125 Habit data for exposure of coastal walker to eroded waste: applicable after the Period of Authorisation

Parameter	Value	Comment
Inhalation rate – adult (m^3 h^{-1})	1.0	
Inhalation rate – child (m^3 h^{-1})	0.64	
Inhalation rate – infant (m^3 h^{-1})	0.22	
Time on site – public (h y^{-1})	73	Time taken to pass exposed waste.

E.4.5.1. Assessment calculations for coastal walker

909. The coastal walker receives a dose from external irradiation, inhalation and inadvertent ingestion at the time of erosion (t, taken to be 2540 y after closure) as follows:

$$Dose_{walker} = \left(\frac{D_{irr,slab}^{Rn}}{8766}\right) T C_{Rn,waste}(t) + D_{inh}^{Rn} T B M_{inh} C_{Rn,waste}(t) + D_{ing}^{Rn} Q_{soil} C_{Rn,waste}(t)$$

where:

- M_{inh} is the dust loading of suspended eroded material inhaled by the walker (kg m^{-3});
- T is the time that the walker is exposed to the material (57.0 h y^{-1});
- B is the breathing rate (m^3 h^{-1});

- $D_{irr,slab}$, D_{inh} and D_{ing} are the dose coefficients for radionuclide *Rn* (Sv y^{-1} Bq^{-1} kg; Sv Bq^{-1}; and Sv Bq^{-1}, respectively);
- 8766 is the number of hours in a year (h y^{-1});
- $C_{Rn,waste}(t)$ is the activity concentration of radionuclide *Rn* (Bq kg^{-1}) in the waste at time of erosion, *t*:

$$C_{Rn,waste}(t) = \frac{A_{Rn}(t)}{V_{landfill}\rho_{waste}}$$

- $V_{landfill}$ is the volume of the landfill in which the activity is assumed to be concentrated (m^3); and,
- ρ_{waste} is the density of the waste (kg m^{-3}).

E.4.5.2. Dose to beach user from exposure to external radiation and ingestion/inhalation of soil

910. The dose to a beach walker after the start of erosion is given in Table 126. The expected dose if each radionuclide is disposed at the maximum inventory is also shown. The highest doses are 20 µSv y^{-1}, indicating that this is the limiting scenario for these radionuclides. The dose from wastes disposed of at Port Clarence will always be lower than this due to application of the sum of fractions approach.

Table 126 Doses to beach walkers after the start of tidal erosion

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Dose at Radiological Capacity (µSv y^{-1})	Limiting age group
H-3	$6.43\ 10^{9}$	$2.95\ 10^{-75}$	$3.37\ 10^{-75}$	$1.21\ 10^{-74}$	$7.79\ 10^{-65}$	Infant
C-14	$1.87\ 10^{8}$	$9.56\ 10^{-12}$	$1.05\ 10^{-11}$	$3.49\ 10^{-11}$	$6.54\ 10^{-3}$	Infant
Cl-36	$1.56\ 10^{8}$	$8.90\ 10^{-10}$	$8.96\ 10^{-10}$	$1.04\ 10^{-9}$	$1.62\ 10^{-1}$	Infant
Ca-41	$5.77\ 10^{9}$	$6.50\ 10^{-13}$	$2.49\ 10^{-12}$	$1.24\ 10^{-11}$	$7.14\ 10^{-2}$	Infant
Mn-54	$1.12\ 10^{13}$	0	0	0	0	
Fe-55	$1.86\ 10^{13}$	$7.25\ 10^{-292}$	$2.73\ 10^{-291}$	$2.52\ 10^{-290}$	$4.68\ 10^{-277}$	Infant
Co-60	$3.58\ 10^{11}$	$5.22\ 10^{-151}$	$5.22\ 10^{-151}$	$5.22\ 10^{-151}$	$1.87\ 10^{-139}$	Infant
Ni-59	$1.95\ 10^{11}$	$6.44\ 10^{-13}$	$9.43\ 10^{-13}$	$8.36\ 10^{-12}$	$1.63\ 10^{0}$	Infant
Ni-63	$2.42\ 10^{11}$	$4.25\ 10^{-20}$	$5.95\ 10^{-20}$	$4.87\ 10^{-19}$	$1.18\ 10^{-7}$	Infant
Zn-65	$8.95\ 10^{11}$	0	0	0	0	
Se-79	$8.98\ 10^{8}$	$2.15\ 10^{-11}$	$7.97\ 10^{-11}$	$6.77\ 10^{-10}$	$6.09\ 10^{-1}$	Infant
Sr-90	$3.83\ 10^{8}$	$2.59\ 10^{-35}$	$2.64\ 10^{-35}$	$3.16\ 10^{-35}$	$1.21\ 10^{-26}$	Infant
Mo-93	$1.44\ 10^{9}$	$1.47\ 10^{-10}$	$1.54\ 10^{-10}$	$2.48\ 10^{-10}$	$3.56\ 10^{-1}$	Infant
Zr-93	$3.12\ 10^{11}$	$3.13\ 10^{-11}$	$9.89\ 10^{-12}$	$1.98\ 10^{-11}$	$9.75\ 10^{0}$	Adult
Nb-93m	$5.06\ 10^{10}$	$1.61\ 10^{-58}$	$1.64\ 10^{-58}$	$2.45\ 10^{-58}$	$1.24\ 10^{-47}$	Infant
Nb-94	$6.09\ 10^{6}$	$3.28\ 10^{-6}$	$3.28\ 10^{-6}$	$3.28\ 10^{-6}$	$2.00\ 10^{1}$	Infant
Tc-99	$6.12\ 10^{8}$	$6.24\ 10^{-11}$	$6.46\ 10^{-11}$	$1.69\ 10^{-10}$	$1.03\ 10^{-1}$	Infant
Ru-106	$9.14\ 10^{11}$	0	0	0	0	
Ag-108m	$2.65\ 10^{8}$	$5.29\ 10^{-8}$	$5.29\ 10^{-8}$	$5.29\ 10^{-8}$	$1.40\ 10^{1}$	Infant

Radionuclide	Radiological Capacity (MBq)	Adult dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose at Radiological Capacity ($\mu Sv\ y^{-1}$)	Limiting age group
Ag-110m	$6.41\ 10^{12}$	0	0	0	0	
Cd-109	$1.04\ 10^{12}$	0	0	0	0	
Sb-125	$4.17\ 10^{11}$	$6.64\ 10^{-284}$	$6.64\ 10^{-284}$	$6.64\ 10^{-284}$	$2.77\ 10^{-272}$	Infant
Sn-119m	$8.43\ 10^{12}$	0	0	0	0	
Sn-123	$2.97\ 10^{12}$	0	0	0	0	
Sn-126	$4.60\ 10^{6}$	$4.35\ 10^{-6}$	$4.35\ 10^{-6}$	$4.35\ 10^{-6}$	$2.00\ 10^{1}$	Infant
Te-127m	$4.07\ 10^{12}$	0	0	0	0	
I-129	$3.01\ 10^{8}$	$5.09\ 10^{-9}$	$5.75\ 10^{-9}$	$1.01\ 10^{-8}$	$3.03\ 10^{0}$	Infant
Ba-133	$7.18\ 10^{9}$	$1.52\ 10^{-79}$	$1.52\ 10^{-79}$	$1.52\ 10^{-79}$	$1.09\ 10^{-69}$	Infant
Cs-134	$1.01\ 10^{11}$	0	0	0	0	
Cs-135	$1.55\ 10^{9}$	$2.88\ 10^{-11}$	$3.04\ 10^{-11}$	$7.52\ 10^{-11}$	$1.17\ 10^{-1}$	Infant
Cs-137	$9.69\ 10^{8}$	$5.71\ 10^{-32}$	$5.71\ 10^{-32}$	$5.71\ 10^{-32}$	$5.53\ 10^{-23}$	Infant
Ce-144	$4.81\ 10^{12}$	0	0	0	0	
Pm-147	$2.14\ 10^{13}$	0	0	0	0	
Sm-147	$4.81\ 10^{8}$	$1.11\ 10^{-8}$	$8.39\ 10^{-9}$	$9.15\ 10^{-9}$	$5.35\ 10^{0}$	Adult
Sm-151	$7.23\ 10^{11}$	$1.66\ 10^{-20}$	$1.48\ 10^{-20}$	$5.81\ 10^{-20}$	$4.20\ 10^{-8}$	Infant
Eu-152	$8.05\ 10^{9}$	$8.77\ 10^{-63}$	$8.77\ 10^{-63}$	$8.77\ 10^{-63}$	$7.06\ 10^{-53}$	Infant
Eu-154	$4.18\ 10^{10}$	$2.97\ 10^{-95}$	$2.97\ 10^{-95}$	$2.97\ 10^{-95}$	$1.24\ 10^{-84}$	Infant
Eu-155	$8.81\ 10^{12}$	$1.69\ 10^{-168}$	$1.69\ 10^{-168}$	$1.69\ 10^{-168}$	$1.49\ 10^{-155}$	Infant
Gd-153	$4.83\ 10^{13}$	0	0	0	0	
Pb-210	$4.85\ 10^{8}$	$6.59\ 10^{-43}$	$1.21\ 10^{-42}$	$1.11\ 10^{-41}$	$5.38\ 10^{-33}$	Infant
Po-210	$6.17\ 10^{9}$	0	0	0	0	
Ra-226	$3.89\ 10^{6}$	$1.39\ 10^{-6}$	$1.39\ 10^{-6}$	$1.49\ 10^{-6}$	$5.79\ 10^{0}$	Infant
Ra-228	$2.25\ 10^{10}$	$6.39\ 10^{-139}$	$6.40\ 10^{-139}$	$6.54\ 10^{-139}$	$1.47\ 10^{-128}$	Infant
Ac-227	$3.04\ 10^{9}$	$9.53\ 10^{-42}$	$8.78\ 10^{-42}$	$8.51\ 10^{-42}$	$2.90\ 10^{-32}$	Adult
Th-228	$1.72\ 10^{11}$	0	0	0	0	
Th-229	$2.88\ 10^{7}$	$6.95\ 10^{-7}$	$6.47\ 10^{-7}$	$6.17\ 10^{-7}$	$2.00\ 10^{1}$	Adult
Th-230	$1.98\ 10^{6}$	$2.79\ 10^{-6}$	$2.77\ 10^{-6}$	$2.93\ 10^{-6}$	$5.80\ 10^{0}$	Infant
Th-232	$7.95\ 10^{6}$	$2.43\ 10^{-6}$	$2.41\ 10^{-6}$	$2.52\ 10^{-6}$	$2.00\ 10^{1}$	Infant
Pa-231	$1.36\ 10^{7}$	$5.21\ 10^{-7}$	$4.53\ 10^{-7}$	$4.20\ 10^{-7}$	$7.08\ 10^{0}$	Adult
U-232	$4.04\ 10^{8}$	$7.51\ 10^{-19}$	$6.43\ 10^{-19}$	$8.87\ 10^{-19}$	$3.58\ 10^{-10}$	Infant
U-233	$1.02\ 10^{8}$	$1.96\ 10^{-7}$	$1.81\ 10^{-7}$	$1.75\ 10^{-7}$	$2.00\ 10^{1}$	Adult
U-234	$1.45\ 10^{8}$	$1.35\ 10^{-8}$	$1.11\ 10^{-8}$	$1.18\ 10^{-8}$	$1.97\ 10^{0}$	Adult
U-235	$6.93\ 10^{7}$	$2.89\ 10^{-7}$	$2.85\ 10^{-7}$	$2.85\ 10^{-7}$	$2.00\ 10^{1}$	Adult
U-236	$1.48\ 10^{9}$	$1.02\ 10^{-8}$	$8.49\ 10^{-9}$	$1.00\ 10^{-8}$	$1.51\ 10^{1}$	Adult
U-238	$1.60\ 10^{9}$	$9.42\ 10^{-9}$	$7.82\ 10^{-9}$	$9.90\ 10^{-9}$	$1.59\ 10^{1}$	Infant
Np-237	$1.42\ 10^{7}$	$8.65\ 10^{-8}$	$6.61\ 10^{-8}$	$5.76\ 10^{-8}$	$1.23\ 10^{0}$	Adult
Pu-238	$7.56\ 10^{8}$	$3.95\ 10^{-12}$	$3.33\ 10^{-12}$	$3.78\ 10^{-12}$	$2.98\ 10^{-3}$	Adult
Pu-239	$1.55\ 10^{8}$	$1.29\ 10^{-7}$	$8.32\ 10^{-8}$	$5.64\ 10^{-8}$	$2.00\ 10^{1}$	Adult
Pu-240	$1.89\ 10^{8}$	$1.06\ 10^{-7}$	$6.84\ 10^{-8}$	$4.63\ 10^{-8}$	$2.00\ 10^{1}$	Adult
Pu-241	$9.39\ 10^{9}$	$7.44\ 10^{-11}$	$5.32\ 10^{-11}$	$4.14\ 10^{-11}$	$6.99\ 10^{-1}$	Adult
Pu-242	$1.58\ 10^{8}$	$1.26\ 10^{-7}$	$8.90\ 10^{-8}$	$5.73\ 10^{-8}$	$2.00\ 10^{1}$	Adult
Pu-244	$1.26\ 10^{8}$	$1.59\ 10^{-7}$	$1.10\ 10^{-7}$	$7.21\ 10^{-8}$	$2.00\ 10^{1}$	Adult
Am-241	$3.03\ 10^{8}$	$2.18\ 10^{-9}$	$1.56\ 10^{-9}$	$1.21\ 10^{-9}$	$6.59\ 10^{-1}$	Adult
Am-242m	$1.75\ 10^{7}$	$2.96\ 10^{-12}$	$1.93\ 10^{-12}$	$1.35\ 10^{-12}$	$5.17\ 10^{-5}$	Adult
Am-243	$1.46\ 10^{8}$	$1.37\ 10^{-7}$	$1.06\ 10^{-7}$	$8.60\ 10^{-8}$	$2.00\ 10^{1}$	Adult
Cm-242	$1.48\ 10^{11}$	$1.24\ 10^{-18}$	$8.01\ 10^{-19}$	$5.63\ 10^{-19}$	$1.83\ 10^{-7}$	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose at Radiological Capacity ($\mu Sv\ y^{-1}$)	Limiting age group
Cm-243	$4.89\ 10^{7}$	$1.56\ 10^{-10}$	$1.01\ 10^{-10}$	$6.82\ 10^{-11}$	$7.61\ 10^{-3}$	Adult
Cm-244	$1.16\ 10^{8}$	$2.94\ 10^{-10}$	$1.90\ 10^{-10}$	$1.29\ 10^{-10}$	$3.41\ 10^{-2}$	Adult
Cm-245	$1.26\ 10^{7}$	$1.95\ 10^{-7}$	$1.63\ 10^{-7}$	$1.46\ 10^{-7}$	$2.46\ 10^{0}$	Adult
Cm-246	$1.27\ 10^{7}$	$7.86\ 10^{-8}$	$5.18\ 10^{-8}$	$3.78\ 10^{-8}$	$9.96\ 10^{-1}$	Adult
Cm-248	$1.45\ 10^{7}$	$4.13\ 10^{-7}$	$2.74\ 10^{-7}$	$1.97\ 10^{-7}$	$6.00\ 10^{0}$	Adult

911. This scenario limits the radiological capacity of 11 radionuclides and hence doses arising from disposing of the radiological capacity are 20 $\mu Sv\ y^{-1}$ for these 11 radionuclides. Table 16 presents the dose rate per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$) to beach walkers at the time of erosion (2,540 years after site closure) for these 11 radionuclides.

912. The doses to the walker at the time of erosion calculated using illustrative inventories are considered further in Appendix D.

E.4.6. Exposure following site erosion and release of leachate into sea

913. As discussed in Section E.4.5, erosion of the landfill is considered unlikely to occur. This scenario considers exposure from releases of leachate from the eroding site into the marine environment. It is assumed that erosion will occur from the seaward side of the landfill and that contamination will be leached from the landfill materials as they are eroded. It is also assumed that the leached contamination will predominantly enter the sea rather than a confined estuary. Doses relating to eroded landfill material on the beach are addressed in Section E.4.5 and are not addressed here.

914. The dose criterion used is a dose of 0.02 $mSv\ y^{-1}$ for the public (this is equivalent to the risk guidance level of $10^{-6}\ y^{-1}$ for exposure of the public post closure, for situations that are expected to occur).

Potentially exposed group

915. An assessment is made of the doses to a family (member of the public) who spends time on a beach or shore of the sea close to the site at some time in the future. It is assumed that the family fishes and collects seafood from the area, which they consume, receiving an ingestion dose. The family also receives external exposure from beaches and fishing equipment and inhalation dose from sea spray.

916. Habits data for the family are presented in Table 127.

Table 127 Habit data for a fishing family exposed as a result of coastal erosion

Habit data	Adult	Child	Infant
Fish consumption (kg y^{-1})	100	20	5
Crustacean consumption (kg y^{-1})	20	5	0
Molluscs consumption (kg y^{-1})	20	5	0
Occupancy on shore (h y^{-1})	2000	2000	2000
Inhalation rate (m^3 y^{-1})	8100	5600	1900

E.4.6.1. Assessment for coastal erosion dose

917. Dose to the family from coastal erosion was calculated using PC CREAM 08. A local marine compartment was set up for Port Clarence using the default dimensions, volumetric exchange rate, sediment parameters and dispersion rate for a new local compartment. This was deemed appropriate as it is not possible to tell what shape the coastline or estuary may be following erosion. The parameters are given in Table 128.

Table 128 Port Clarence Local Marine Compartment Parameters

Parameter	Value
Volume (m^3)	2 10^8
Depth (m)	10
Coastline length (m)	1000
Volumetric exchange rate (m^3/y)	4 10^9
Suspended sediment load (t m^{-3})	2 10^{-4}
Sedimentation rate (t m^2 y^{-1})	1 10^{-4}
Sediment density (t m^{-3})	2.6
Diffusion rate (m^2 y^{-1})	3.15 10^{-2}

918. The DORIS module in PC CREAM was used to calculate activity concentrations in sea water, seabed sediment and seafood assuming a 1 Bq y^{-1} release to the local compartment. This was used as input to the ASSESSOR module of PC CREAM, in which the dose to the fishing family was calculated using habits data as shown in Table 127, also for a 1 Bq y^{-1} release.

919. It is assumed that erosion starts 2540 years after closure, and that the site is eroded at a rate of 0.1 m^3 y^{-1}. As the facility is approximately 900 m in length, it would take 9,000 y for the whole facility to be eroded. It was assumed that activity is evenly released over this 9,000 y time period.

920. The activity of radionuclides and ingrown daughters 2540 y after closure, assuming 1 MBq initial inventory, was calculated using the Bateman equations. Any radionuclide or daughter for which the activity remaining was less than 10^{-10} Bq was excluded from further assessment. For the remaining radionuclides, the activity was divided by the erosion period of 9,000 y to obtain a yearly release rate for each radionuclide. Doses calculated in PC CREAM (for 1 Bq y^{-1}) were scaled to the calculated yearly release rate to obtain doses from a 1 MBq inventory.

E.4.6.2. Dose to fishing family following erosion of the site and release of leachate to sea

921. The total dose to a fishing family from ingestion of seafood, external radiation and inhalation of seaspray once erosion of the landfill starts in the year 2540 are given in Table 129, respectively. The cut-off of 10^{-13} produces values of zero dose for shorter half-life radionuclides. The highest dose per MBq disposed at Port Clarence following erosion is from Th-230 ($3\ 10^{-9}$ µSv y^{-1} MBq^{-1}), and the dose from wastes disposed of at Port Clarence will always be lower than this due to application of the sum of fractions approach. This scenario does not limit the radiological capacity for any radionuclide.

922. The dose was also assessed assuming that the radiological capacity was disposed at the landfill.

Table 129 Doses to a fishing family after the start of tidal erosion

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Dose at Radiological Capacity (µSv y^{-1})	Limiting age group
H-3	$6.43\ 10^{9}$	0	0	0	0	
C-14	$1.87\ 10^{8}$	$8.46\ 10^{-9}$	$2.80\ 10^{-9}$	$2.35\ 10^{-10}$	$1.58\ 10^{0}$	Adult
Cl-36	$1.56\ 10^{8}$	$1.65\ 10^{-13}$	$1.17\ 10^{-13}$	$8.92\ 10^{-14}$	$2.58\ 10^{-5}$	Adult
Ca-41	$5.77\ 10^{9}$	$1.56\ 10^{-12}$	$1.30\ 10^{-12}$	$9.06\ 10^{-13}$	$9.02\ 10^{-3}$	Adult
Mn-54	$1.12\ 10^{13}$	0	0	0	0	
Fe-55	$1.86\ 10^{13}$	0	0	0	0	
Co-60	$3.58\ 10^{11}$	0	0	0	0	
Ni-59	$1.95\ 10^{11}$	$6.24\ 10^{-11}$	$5.86\ 10^{-11}$	$5.64\ 10^{-11}$	$1.22\ 10^{1}$	Adult
Ni-63	$2.42\ 10^{11}$	$3.17\ 10^{-19}$	$1.43\ 10^{-19}$	$1.30\ 10^{-20}$	$7.66\ 10^{-8}$	Adult
Zn-65	$8.95\ 10^{11}$	0	0	0	0	
Se-79	$8.98\ 10^{8}$	$1.87\ 10^{-8}$	$2.16\ 10^{-8}$	$1.94\ 10^{-9}$	$1.94\ 10^{1}$	Child
Sr-90	$3.83\ 10^{8}$	0	0	0	0	
Mo-93	$1.44\ 10^{9}$	$5.76\ 10^{-9}$	$1.87\ 10^{-9}$	$1.86\ 10^{-11}$	$8.29\ 10^{0}$	Adult
Zr-93	$3.12\ 10^{11}$	$6.42\ 10^{-11}$	$4.96\ 10^{-11}$	$4.71\ 10^{-11}$	$2.00\ 10^{1}$	Adult
Nb-93m	$5.06\ 10^{10}$	0	0	0	0	
Nb-94	$6.09\ 10^{6}$	$3.58\ 10^{-8}$	$3.54\ 10^{-8}$	$3.54\ 10^{-8}$	$2.18\ 10^{-1}$	Adult
Tc-99	$6.12\ 10^{8}$	$6.99\ 10^{-10}$	$3.54\ 10^{-10}$	$2.96\ 10^{-12}$	$4.27\ 10^{-1}$	Adult
Ru-106	$9.14\ 10^{11}$	0	0	0	0	
Ag-108m	$2.65\ 10^{8}$	$3.30\ 10^{-10}$	$2.00\ 10^{-10}$	$8.93\ 10^{-11}$	$8.75\ 10^{-2}$	Adult
Ag-110m	$6.41\ 10^{12}$	0	0	0	0	
Cd-109	$1.04\ 10^{12}$	0	0	0	0	
Sb-125	$4.17\ 10^{11}$	0	0	0	0	
Sn-119m	$8.43\ 10^{12}$	0	0	0	0	
Sn-123	$2.97\ 10^{12}$	0	0	0	0	
Sn-126	$4.60\ 10^{6}$	$2.76\ 10^{-7}$	$1.42\ 10^{-7}$	$2.38\ 10^{-8}$	$1.27\ 10^{0}$	Adult
Te-127m	$4.07\ 10^{12}$	0	0	0	0	
I-129	$3.01\ 10^{8}$	$1.53\ 10^{-9}$	$6.35\ 10^{-10}$	$3.31\ 10^{-11}$	$4.59\ 10^{-1}$	Adult
Ba-133	$7.18\ 10^{9}$	0	0	0	0	
Cs-134	$1.01\ 10^{11}$	0	0	0	0	
Cs-135	$1.55\ 10^{9}$	$8.73\ 10^{-11}$	$2.01\ 10^{-11}$	$7.35\ 10^{-12}$	$1.36\ 10^{-1}$	Adult

Radionuclide	Radiological Capacity (MBq)	Adult dose (µSv y^{-1} MBq^{-1})	Child dose (µSv y^{-1} MBq^{-1})	Infant dose (µSv y^{-1} MBq^{-1})	Dose at Radiological Capacity (µSv y^{-1})	Limiting age group
Cs-137	$9.69\ 10^{8}$	0	0	0	0	
Ce-144	$4.81\ 10^{12}$	0	0	0	0	
Pm-147	$2.14\ 10^{13}$	0	0	0	0	
Sm-147	$4.81\ 10^{8}$	$4.26\ 10^{-10}$	$1.39\ 10^{-10}$	$3.49\ 10^{-12}$	$2.05\ 10^{-1}$	Adult
Sm-151	$7.23\ 10^{11}$	$3.61\ 10^{-21}$	$2.27\ 10^{-21}$	$9.49\ 10^{-22}$	$2.61\ 10^{-9}$	Adult
Eu-152	$8.05\ 10^{9}$	0	0	0	0	
Eu-154	$4.18\ 10^{10}$	0	0	0	0	
Eu-155	$8.81\ 10^{12}$	0	0	0	0	
Gd-153	$4.83\ 10^{13}$	0	0	0	0	
Pb-210	$4.85\ 10^{8}$	0	0	0	0	
Po-210	$6.17\ 10^{9}$	0	0	0	0	
Ra-226	$3.89\ 10^{6}$	$5.14\ 10^{-6}$	$2.74\ 10^{-6}$	$2.51\ 10^{-7}$	$2.00\ 10^{1}$	Adult
Ra-228	$2.25\ 10^{10}$	0	0	0	0	
Ac-227	$3.04\ 10^{9}$	$3.53\ 10^{-43}$	$3.61\ 10^{-43}$	$7.85\ 10^{-44}$	$1.10\ 10^{-33}$	Child
Th-228	$1.72\ 10^{11}$	0	0	0	0	
Th-229	$2.88\ 10^{7}$	$7.96\ 10^{-9}$	$6.83\ 10^{-9}$	$6.34\ 10^{-9}$	$2.29\ 10^{-1}$	Adult
Th-230	$1.98\ 10^{6}$	$1.01\ 10^{-5}$	$5.38\ 10^{-6}$	$4.74\ 10^{-7}$	$2.00\ 10^{1}$	Adult
Th-232	$7.95\ 10^{6}$	$2.97\ 10^{-7}$	$3.92\ 10^{-7}$	$4.69\ 10^{-8}$	$3.12\ 10^{0}$	Child
Pa-231	$1.36\ 10^{7}$	$3.01\ 10^{-8}$	$2.90\ 10^{-8}$	$8.71\ 10^{-9}$	$4.08\ 10^{-1}$	Adult
U-232	$4.04\ 10^{8}$	$4.94\ 10^{-20}$	$2.15\ 10^{-20}$	$2.24\ 10^{-22}$	$2.00\ 10^{-11}$	Adult
U-233	$1.02\ 10^{8}$	$1.96\ 10^{-9}$	$1.10\ 10^{-9}$	$6.21\ 10^{-10}$	$2.00\ 10^{-1}$	Adult
U-234	$1.45\ 10^{8}$	$1.38\ 10^{-7}$	$7.32\ 10^{-8}$	$6.42\ 10^{-9}$	$2.00\ 10^{1}$	Adult
U-235	$6.93\ 10^{7}$	$2.98\ 10^{-9}$	$2.36\ 10^{-9}$	$9.73\ 10^{-10}$	$2.06\ 10^{-1}$	Adult
U-236	$1.48\ 10^{9}$	$8.94\ 10^{-10}$	$3.40\ 10^{-10}$	$1.24\ 10^{-11}$	$1.32\ 10^{0}$	Adult
U-238	$1.60\ 10^{9}$	$1.44\ 10^{-9}$	$7.34\ 10^{-10}$	$2.41\ 10^{-10}$	$2.32\ 10^{0}$	Adult
Np-237	$1.42\ 10^{7}$	$2.67\ 10^{-8}$	$7.41\ 10^{-9}$	$1.01\ 10^{-9}$	$3.79\ 10^{-1}$	Adult
Pu-238	$7.56\ 10^{8}$	$4.55\ 10^{-11}$	$2.42\ 10^{-11}$	$2.12\ 10^{-12}$	$3.44\ 10^{-2}$	Adult
Pu-239	$1.55\ 10^{8}$	$2.00\ 10^{-8}$	$5.43\ 10^{-9}$	$6.12\ 10^{-11}$	$3.11\ 10^{0}$	Adult
Pu-240	$1.89\ 10^{8}$	$1.65\ 10^{-8}$	$4.50\ 10^{-9}$	$8.06\ 10^{-11}$	$3.12\ 10^{0}$	Adult
Pu-241	$9.39\ 10^{9}$	$3.82\ 10^{-12}$	$1.27\ 10^{-12}$	$3.13\ 10^{-13}$	$3.59\ 10^{-2}$	Adult
Pu-242	$1.58\ 10^{8}$	$2.06\ 10^{-8}$	$5.63\ 10^{-9}$	$9.50\ 10^{-11}$	$3.27\ 10^{0}$	Adult
Pu-244	$1.26\ 10^{8}$	$3.47\ 10^{-8}$	$1.58\ 10^{-8}$	$8.86\ 10^{-9}$	$4.37\ 10^{0}$	Adult
Am-241	$3.03\ 10^{8}$	$1.11\ 10^{-10}$	$3.69\ 10^{-11}$	$9.10\ 10^{-12}$	$3.37\ 10^{-2}$	Adult
Am-242m	$1.75\ 10^{7}$	$5.62\ 10^{-11}$	$2.97\ 10^{-11}$	$2.60\ 10^{-12}$	$9.81\ 10^{-4}$	Adult
Am-243	$1.46\ 10^{8}$	$1.05\ 10^{-8}$	$6.13\ 10^{-9}$	$4.54\ 10^{-9}$	$1.53\ 10^{0}$	Adult
Cm-242	$1.48\ 10^{11}$	0	0	0	0	
Cm-243	$4.89\ 10^{7}$	0	0	0	0	
Cm-244	$1.16\ 10^{8}$	0	0	0	0	
Cm-245	$1.26\ 10^{7}$	$1.48\ 10^{-8}$	$5.77\ 10^{-9}$	$2.38\ 10^{-9}$	$1.86\ 10^{-1}$	Adult
Cm-246	$1.27\ 10^{7}$	$6.26\ 10^{-9}$	$1.67\ 10^{-9}$	$3.55\ 10^{-11}$	$7.94\ 10^{-2}$	Adult
Cm-248	$1.45\ 10^{7}$	$3.25\ 10^{-8}$	$8.94\ 10^{-9}$	$7.61\ 10^{-11}$	$4.73\ 10^{-1}$	Adult

923. The doses calculated using illustrative inventories are considered further in Appendix D.

E.5. Human intrusion scenarios {R7}

924. After the end of active management control of the site, it is assumed that use of the site eventually becomes unrestricted and that either intentional or unintentional intrusion through the disposal cell cap may occur, leading to members of potential exposure groups receiving radiation doses as a consequence of access to waste.

925. In reality, it is likely that knowledge about the site would be retained and planning controls would continue to apply for decades. Redevelopment of the site in an absence of knowledge about its contents is not likely for a long time after the end of the period of authorisation. A review of both intentional and unintentional intrusion scenarios, and on-site or near-site occupancy scenarios, identified in generic guidance or in previous publicly available ESCs [(IAEA, 2004), (Augean, 2009), (UK Environment Agencies, 2009), (Environment Agency, 2012)] has identified four potential intrusion scenarios and six potentially exposed groups likely to be of relevance to Port Clarence. The identified cases are believed to represent the most likely and relevant modes of human intrusion (i.e. they possess the potential to directly excavate the disposed wastes or damage the engineered cap).

926. The active management phase is assumed to last for 60 years. After this the following human intrusion scenarios and exposed groups are considered in the ESC:

- Borehole drilling (at 60 years): dose to worker;
- Trial pit excavation (at 60 years): dose to worker;
- Excavation for housing or road (at 150 years):
  - Dose to worker during excavation;
  - Dose to resident on the site;
  - Radon exposure of resident; and
- Small holder excavating on the site (at 60 and 200 years): dose to smallholder.

927. Dose to a laboratory analyst working with borehole or trial pit samples has not been included in the assessment. Doses to this exposed group in the ENRMF ESC were lower than for other exposed groups for all radionuclides, so it was deemed unnecessary to assess this exposed group for Port Clarence.

928. In Table 130 descriptions of these human intrusion cases based on LLWR assessments (Hicks & Baldwin, 2011) are presented.

Table 130 Human intrusion events

Event/scenario	Summary
Borehole drilling	Could be undertaken as part of geotechnical investigations. The cap and profile materials above the waste would reduce the potential for intrusion into the waste, although boreholes will fully penetrate waste, if drilled into waste cell. Laboratory analysis of contaminated soil samples is also considered within the assessment. Those involved in the intrusion (i.e. drill operatives and laboratory analyst) are assumed to be exposed to the hazard.
Trial pit excavation	Could be undertaken as part of geotechnical investigations. Has the potential to disturb waste, if undertaken into a waste cell. Trial pit excavators are assumed to be exposed to the hazard.
Residential occupant (intact cap)	A housing development is positioned over the landfill. Buildings constructed using ‘floating’ foundations will not penetrate the cap.
Excavation for housing/road	Construction activities for housing developments would include shallow excavations and cap disturbance to prepare the site and install roads and services. Foundations for domestic and light buildings, typically 1 or 2 m deep have the potential to penetrate the engineered cap, particularly, if domestic buildings include cellars. There is also the possibility of building directly upon a waste/spoil mix (i.e. the cap has been destroyed as part of the intrusion event). Those involved in the excavation work would be exposed to the hazard, as would (in the long term) site occupants. Both are considered within the (radon) human intrusion assessment. Subsequent occupation of the site is assumed to be residential, not small holding.
Smallholding	Construction/agricultural activities could result in contaminated material left at the surface. A smallholding is more cautious than a farm, as it allows crops to be grown on a more concentrated activity source.

929. The impact assessment undertaken on behalf of the LLWR (LLWR Ltd, 2011b) suggests that house occupancy and a smallholding on site are likely to offer the highest doses to exposed persons, followed by the borehole driller/housing construction worker. Although there are marked differences between the disposal facilities and the waste inventories, these potentially exposed groups are also likely to represent the limiting cases for Port Clarence in this assessment.

930. Exposure to the borehole driller and the excavation worker is considered as a result of external irradiation, inhalation of dust and inadvertent ingestion of dust.

931. Radiation doses to the resident and smallholder are considered to arise as a result of external irradiation, inhalation of dust and radon gas, inadvertent ingestion of dust and the ingestion of home produced food. The assumptions concerning the resident and smallholder scenarios differ in several ways, including: the quantity of excavated waste, habit data and the time when intrusion is assumed to occur.

932. The dose implications of excavation of waste materials that consist of different sized objects are also considered by assessing the dose to a worker or site occupant. Further details are given in Section E.6.

933. A site re-engineering/remediation scenario was included in the SNIFFER methodology to cover the situation where a site operator has no records of radioactive waste disposals or their location and excavates waste during final site restoration works. In the case of Port Clarence, which is a hazardous waste landfill, with a Permit to receive LLW, records would be maintained as a condition of the Permit. Any remediation work would be done with the knowledge that there was radioactive material on the site and it can be assumed that appropriate precautions against exposure would be adopted. Site rules also prevent any disposal of radioactive waste within 2 m of basal liners and within 1 m of the top of the cell. Hence this scenario is not considered in the ESC.

934. The dose guidance level (human intrusion) is 3 mSv y^{-1} to around 20 mSv y^{-1}, depending on the duration of exposure, and this is applied to all intrusion scenarios for both the public and workers. Future removal of a part of a site as part of a major road construction project has been considered in some assessments (IAEA, 2003). However, this is considered to be extremely unlikely, the dose to the road constructor would be covered by the dose to the borehole driller, and the dose to a resident on spoil would be covered by the site occupant. Hence, it is not explicitly considered in the ESC.

935. In Table 131 the conceptual models and relevant exposure pathways considered in this ESC for each of the human intrusion cases are summarised. The radiological impact of each of these intrusion cases has been estimated using the approaches described in Sections E.5.2 to E.5.8.

Table 131 Summary of human intrusion cases and exposure pathways

Event/scenario	Exposure pathway	Description
Borehole drilling: operative	Inhalation of contaminated dust	Dust generated by borehole intrusion into waste includes radioactive material. Operative inhales dust during drilling activities.
	Ingestion of contaminated material	Operative ingests contaminated material during drilling activities.
	External irradiation	Contaminated material is left on the ground during drilling activities. A worker in close proximity to this material is exposed to external irradiation and dust on skin.
Trial pit excavation	Inhalation of contaminated dust	Dust generated by trial pit intrusion into waste includes radioactive material. Operative inhales dust during excavation activities.

Event/scenario	Exposure pathway	Description
	Ingestion of contaminated material	Operative ingests contaminated material during excavation activities.
	External irradiation	Contaminated material is left on the ground during excavation activities. A worker in close proximity to this material is exposed to external irradiation and dust on skin.
Excavation for housing/road: excavator	Inhalation of contaminated dust	Excavations into waste generate dust including radioactive material. Worker inhales dust during excavation activities.
	Ingestion of contaminated material	Operative ingests contaminated material during excavation activities.
	External irradiation	Contaminated material is left on the ground during excavation activities. A worker in close proximity to this material is exposed to external irradiation and dust on skin.
Residential occupant (intact cap)	Gas (including radon) inhalation	The house occupant is exposed to gases emanating from contaminated material beneath the house.
	External irradiation	The house is built above the intact cap. As a result, a site occupant is exposed to external irradiation while indoors and outside. The concrete floor of the house provides some shielding from gamma radiation.
Excavation for housing/road: long-term residential occupant	Inhalation of contaminated dust	Contaminated material is left on the ground at the site after construction of a housing development. Wind action generates contaminated dust and a site occupant is exposed to the dust while outside.
	Ingestion of contaminated material	While outside (e.g. gardening), a site occupant ingests contaminated material (e.g. through hand-to-mouth contact and licking of the lips). Ingestion of contaminated vegetables grown on the site is also considered.
	External irradiation	The house is built on contaminated ground and contaminated material is present in garden soil. As a result, a site occupant is exposed to external irradiation while indoors and outside. The concrete floor of the house provides some shielding from gamma radiation.
	Gas (including Radon) inhalation	The house occupant is exposed to gases emanating from contaminated material beneath the house.
Smallholding	Inhalation of contaminated dust	Contaminated material is left on the ground at the site after site excavation. Wind action generates contaminated dust and a site occupant is exposed to the dust while outside.

Event/scenario	Exposure pathway	Description
	Ingestion of contaminated material	The smallholder ingests contaminated foodstuffs as a result of growing crops and keeping animals on the site. The smallholder also inadvertently ingests contaminated soil while working outside.
	External irradiation	The house is built on contaminated ground and contaminated material is present in garden soil. As a result, a site occupant is exposed to external irradiation while indoors and outside. The concrete floor of the house provides some shielding from gamma radiation.
	Gas (including Radon) inhalation	The house occupant is exposed to gases emanating from contaminated material beneath the house.

E.5.1. Presentation of dose assessments

936. The radiological capacity for individual radionuclides present in the LLW is obtained from the results of the ESC and depends on the radiological characteristics of the radionuclide. The radiological capacity is calculated on the basis that the LLW only contains this one radionuclide. The overall radiological capacity for an individual radionuclide is the minimum of the radiological capacities calculated for each of the scenarios. The results of the assessment are presented as effective doses per MBq disposed (µSv y^{-1} MBq^{-1}).

937. The results of the dose assessments presented in Sections E.5.2 to E.5.8 show the radiological capacity that could be disposed of each radionuclide and the dose (µSv y^{-1}) from disposal of that inventory. The dose calculated for each radionuclide would only be achieved if that radionuclide was the only one disposed of. Actual waste disposal will be controlled using a sum of fractions approach (see paragraph 411).

938. Estimates of radiological impact based on 'illustrative inventories' for waste streams that might be typical of those contributing to the total impact from disposals at the facility have been produced. These estimates are presented in Appendix D.

E.5.2. Borehole drilling – Drill Operative

E.5.2.1. Estimating activity concentration in waste for exposure calculations

939. The initial radioactive inventory evolves with time as radionuclides decay and as they are slowly released from the waste cell (i.e. seepage through the sealing layer and the barrier). Consequently, the activity at time *t*, $A_{Rn}(t)$, is given (after site closure) in SNIFFER (SNIFFER, 2006):

$$A_{Rn}(t) = e^{-\left(\lambda_{Rn}+\lambda^{Rn}_{waste,after}\right)(t-t_{op})} A_{Rn,initial} e^{-\left(\lambda_{Rn}+\lambda^{Rn}_{waste,before}\right)t_{op}}$$

where:

$$\lambda_{waste,before}^{Rn} = \frac{q_{out}}{V_{landfill}(\varphi_{waste}\varepsilon + \rho_{waste}K_{d,waste}^{Rn})}$$

- q_{out} is the volume of water flowing through the liner before closure (m^3 y^{-1});
- $V_{landfill}$ is the volume of the waste (m^3);
- φ_{waste} is the porosity of the waste;
- ε is the degree of saturation of the waste;
- ρ_{waste} is the bulk density of the waste (kg m^{-3});
- $K_{d,waste}^{Rn}$ is the distribution coefficient for radionuclide *Rn* in the waste (m^3 kg^{-1}) ;
- λ_{Rn} is the decay constant of radionuclide *Rn* (y^{-1});
- t_{op} is the time that the landfill is operational (taken to be 0 years);
- $A_{Rn,initial}$ is the initial inventory of radionuclide *Rn;* and,

$$\lambda_{waste,after}^{Rn} = \frac{q_{barrier}}{V_{landfill}(\varphi_{waste}\varepsilon + \rho_{waste}K_{d,waste}^{Rn})}$$

- $q_{barrier}$ is the volume of water flowing out of the landfill into the geological barrier after closure (m^3 y^{-1}).

940. The waste density and porosity are given in Table 119. The time that the landfill is operational is assumed to be 0 years so that no depletion of the inventory in the landfill during the operational period is allowed for. This will produce an overestimate of the inventory in the landfill site at the time of intrusion.

941. Seepage through a geomembrane sealing layer is dominated by flow through defects (holes) in the liner, SNIFFER (SNIFFER, 2006). The flow is given by an empirical formula:

$$q_{out} = c \cdot a_{Defect}^{0.1} \cdot h^{0.9} \cdot K_{Barrier}^{0.74} \cdot 3.16E + 07$$

where:

- c is a constant depending on the contact between the liner and the material below;
- a_{Defect} is the area of the defects (m^2);
- h is the head of leachate (m);
- $K_{barrier}$ is the hydraulic conductivity of the barrier (m s^{-1}); and,

- 3.16E+07 is the number of seconds in a year (s y^{-1}).

942. Assumptions regarding the liner are given in Table 119. During the landfill's operational period, $q_{barrier}$ is set equal to q_{out}.

943. After closure of the landfill, $q_{barrier}$ is set to be:

$$min(q_{inf}, a_{landfill}K_{barrier})$$

where:

- $a_{landfill}$ is the surface area of the landfill (m^2); and,
- q_{inf} is the infiltration volume into the landfill, given by:

$$q_{inf} = P_{eff} \cdot a_{landfill}$$

and,

$$P_{eff} = (P_{total} - AE - runoff)\left[1 - E_0\left(1 - \frac{t}{t_f}\right)\right] for\ t \leq t_f$$

where:

- P_{eff} is the rate of water infiltration through the cap of the landfill (m y^{-1});
- P_{total} is the total precipitation (m y^{-1});
- AE is the amount of precipitation lost to evapotranspiration (m y^{-1});
- $runoff$ is the amount of precipitation lost by runoff (m y^{-1});
- E_0 is the initial cap efficiency;
- t is the time after closure (y); and,
- t_f is the time of cap failure (y).

E.5.2.2. Assessment calculations for Drill Operative

External irradiation, inhalation and ingestion

944. The drill operative receives a dose from external irradiation, inhalation and ingestion (SNIFFER, 2006):

$$Dose_{excavator} = \left(\frac{D_{irr,slab}^{Rn}}{8766}\right)TC_{Rn,waste}(t) + D_{inh}^{Rn}TBM_{inh}C_{Rn,waste}(t)$$

$$+ D_{ing}^{Rn}TM_{ing}C_{Rn,waste}(t)$$

where:

- M_{inh} is the dust loading of contaminated waste inhaled by the excavator (kg m^{-3});
- M_{ing} is the rate of ingestion of dust from the material (kg h^{-1});
- T is the time that the excavator is exposed to the material (h y^{-1});
- B is the breathing rate (m^3 h^{-1});
- $D_{irr,slab}^{Rn}$, D_{inh}^{Rn}, and D_{ing}^{Rn} are the dose coefficients for radionuclide *Rn* (Sv y^{-1} Bq^{-1} kg; Sv Bq^{-1}; and Sv Bq^{-1}, respectively);
- 8766 is the number of hours in a year (h y^{-1});
- $C_{Rn,waste}(t)$ is the activity concentration of radionuclide *Rn* (Bq kg^{-1}) in the waste at time of excavation, *t*:

$$C_{Rn,waste}(t) = \frac{A_{Rn}(t)}{V_{landfill}\rho_{waste}}$$

- $V_{landfill}$ is the volume of the landfill in which the activity is assumed to be concentrated (m^3); and,
- ρ_{waste} is the density of the waste (kg m^{-3}).

Hands and face

945. While the exposure to external irradiation is assumed to arise from proximity to a semi-infinite slab of contaminated material, there is also a possibility of a dose arising from direct contact with contaminated waste dust on the hands and face.

946. For the hands, this is given by:

$$Dose_{skin,hands} = \left(\frac{C_{Rn,waste}(t)d_{hands}\rho_{waste}}{10^4}\right)\left(D_{gamma7}^{Rn} + D_{beta40}^{Rn}\right)W_{skin}T\frac{Area_{hands}}{Area_{body}}$$

where:

- D_{gamma7}^{Rn} is the skin equivalent dose rate for radionuclide *Rn* to the basal layer of the skin epidermis for gamma irradiation (Sv h^{-1} Bq^{-1} cm^2) [see Appendix B of (Augean, 2009)];
- D_{beta40}^{Rn} is the skin equivalent dose rate for radionuclide *Rn* to the basal layer of the skin epidermis for beta irradiation, skin thickness 400 µm (40 mg cm^{-2}), (Sv h^{-1} Bq^{-1} cm^2) [see Appendix B of (Augean, 2009)];
- 10^4 converts Bq m^{-2} to Bq cm^{-2};
- d_{hands} is the thickness of the contaminated layer on the hands (m);

- W_{skin} is the tissue weighting factor for skin;
- $Area_{hands}$ is the area of skin in contact with contaminated material (cm^2); and,
- $Area_{body}$ is the total exposed skin area of the adult body (cm^2).

947. For the face, this is given by:

$$D_{skin,face} = \left(\frac{C_{Rn,waste}(t)d_{face}\rho_{waste}}{10^4}\right)\left(D^{Rn}_{gamma7} + D^{Rn}_{beta40}\right)W_{skin}T\frac{Area_{face}}{Area_{body}}$$

where; the meaning of the symbols is a direct substitution of *face* for *hands*.

948. Note that Borehole driller assessment calculation is cautious as no account is taken of non-contaminated cap material that is also be excavated and becomes mixed with the radioactive material, resulting in dilution.

Table 132 Parameters used for the borehole excavation scenario*

Parameter	Units	Value	Description
M_{inh}	kg m^{-3}	6 10^{-7}**	Dust load of contaminated waste inhaled by the excavator
M_{ing}	kg h^{-1}	1.25 10^{-5}***	Rate of ingestion of dust from excavated material
T	h y^{-1}	16*	Time the excavator is exposed to excavated material (per event)
B	m^3 h^{-1}	1.2	Worker breathing rate
$V_{landfill}$	m^3	See Table 61	Volume of landfill (cells) in which activity is homogeneously distributed
ρ_{waste}	kg m^{-3}	See Table 141	Waste density
d_{hands}	M	1.0 10^{-4}	Thickness of the contaminated layer on the hands
W_{skin}		1 10^{-2}	Tissue weighting factor for skin
$Area_{hands}$	cm^2	2 10^2	Area of skin in contact with contaminated dust
$Area_{body}$	cm^2	3 10^3	Area of skin in contact with contaminated dust
d_{face}	M	5.0 10^{-5}	Thickness of the contaminated layer on the face
$Area_{face}$	cm^2	1 10^2	Area of skin in contact with contaminated dust
$V_{excavate}$	m^3	0.5*	Volume of excavated material

* Values taken from (Augean, 2009), unless otherwise stated.
** Values taken from (Hicks & Baldwin, 2011).
*** Values from (US EPA, 2014).

949. The calculations for a borehole drill operative assume that a single drilling engineer is involved in 5 borehole excavations (Hicks & Baldwin, 2011), i.e. the potential dose arising from 5 intrusion events is calculated.

E.5.2.3. Dose to Borehole Drill Operative on site after 60 years

950. In Table 133 the dose rates to borehole drill operatives (µSv y^{-1} MBq^{-1}) involved in excavating waste at Port Clarence 60 years after capping are presented. The 60 years after capping is immediately at the end of the period of authorisation.

951. The largest dose rates per MBq disposal are for Nb-94, Ag-108m, Th-229, Pa-231 and Cm-248. These radionuclides will correspondingly have the smallest radiological capacities under this scenario. Radiological capacity calculations are presented in Section 7.4.2. No Ra-226 emplacement depth restrictions are assumed in the calculation of the doses to the borehole drill operative.

Table 133 Dose to Borehole Drill Operative excavating at the site

Radionuclide	Radiological capacity (MBq)	Dose to Borehole drill operative (60y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})
H-3	$6.43\ 10^{9}$	$1.77\ 10^{-16}$	$1.14\ 10^{-6}$
C-14	$1.87\ 10^{8}$	$1.77\ 10^{-13}$	$3.31\ 10^{-5}$
Cl-36	$1.56\ 10^{8}$	$1.33\ 10^{-12}$	$2.08\ 10^{-4}$
Ca-41	$5.77\ 10^{9}$	$3.15\ 10^{-14}$	$1.82\ 10^{-4}$
Mn-54	$1.12\ 10^{13}$	$1.55\ 10^{-30}$	$1.73\ 10^{-17}$
Fe-55	$1.86\ 10^{13}$	$1.54\ 10^{-20}$	$2.87\ 10^{-7}$
Co-60	$3.58\ 10^{11}$	$2.46\ 10^{-12}$	$8.81\ 10^{-1}$
Ni-59	$1.95\ 10^{11}$	$2.41\ 10^{-14}$	$4.71\ 10^{-3}$
Ni-63	$2.42\ 10^{11}$	$2.37\ 10^{-14}$	$5.73\ 10^{-3}$
Zn-65	$8.95\ 10^{11}$	$1.35\ 10^{-36}$	$1.21\ 10^{-24}$
Se-79	$8.98\ 10^{8}$	$5.61\ 10^{-13}$	$5.04\ 10^{-4}$
Sr-90	$3.83\ 10^{8}$	$3.93\ 10^{-12}$	$1.50\ 10^{-3}$
Mo-93	$1.44\ 10^{9}$	$7.43\ 10^{-13}$	$1.07\ 10^{-3}$
Zr-93	$3.12\ 10^{11}$	$4.14\ 10^{-13}$	$1.29\ 10^{-1}$
Nb-93m	$5.06\ 10^{10}$	$5.91\ 10^{-15}$	$2.99\ 10^{-4}$
Nb-94	$6.09\ 10^{6}$	$3.91\ 10^{-9}$	$2.38\ 10^{-2}$
Tc-99	$6.12\ 10^{8}$	$3.22\ 10^{-13}$	$1.97\ 10^{-4}$
Ru-106	$9.14\ 10^{11}$	$1.16\ 10^{-27}$	$1.06\ 10^{-15}$
Ag-108m	$2.65\ 10^{8}$	$3.54\ 10^{-9}$	$9.39\ 10^{-1}$
Ag-110m	$6.41\ 10^{12}$	$2.73\ 10^{-35}$	$1.75\ 10^{-22}$
Cd-109	$1.04\ 10^{12}$	$3.21\ 10^{-26}$	$3.34\ 10^{-14}$
Sb-125	$4.17\ 10^{11}$	$2.82\ 10^{-16}$	$1.18\ 10^{-4}$
Sn-119m	$8.43\ 10^{12}$	$4.10\ 10^{-35}$	$3.46\ 10^{-22}$
Sn-123	$2.97\ 10^{12}$	$1.82\ 10^{-62}$	$5.40\ 10^{-50}$
Sn-126	$4.60\ 10^{6}$	$4.80\ 10^{-9}$	$2.21\ 10^{-2}$
Te-127m	$4.07\ 10^{12}$	$1.81\ 10^{-74}$	$7.35\ 10^{-62}$
I-129	$3.01\ 10^{8}$	$2.29\ 10^{-11}$	$6.88\ 10^{-3}$
Ba-133	$7.18\ 10^{9}$	$1.54\ 10^{-11}$	$1.11\ 10^{-1}$

Radionuclide	Radiological capacity (MBq)	Dose to Borehole drill operative (60y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})
Cs-134	$1.01\ 10^{11}$	$6.88\ 10^{-18}$	$6.96\ 10^{-7}$
Cs-135	$1.55\ 10^{9}$	$4.43\ 10^{-13}$	$6.87\ 10^{-4}$
Cs-137	$9.69\ 10^{8}$	$3.48\ 10^{-10}$	$3.37\ 10^{-1}$
Ce-144	$4.81\ 10^{12}$	$2.15\ 10^{-34}$	$1.03\ 10^{-21}$
Pm-147	$2.14\ 10^{13}$	$1.91\ 10^{-20}$	$4.09\ 10^{-7}$
Sm-147	$4.81\ 10^{8}$	$9.46\ 10^{-11}$	$4.54\ 10^{-2}$
Sm-151	$7.23\ 10^{11}$	$3.33\ 10^{-14}$	$2.41\ 10^{-2}$
Eu-152	$8.05\ 10^{9}$	$1.32\ 10^{-10}$	$1.06\ 10^{0}$
Eu-154	$4.18\ 10^{10}$	$2.46\ 10^{-11}$	$1.03\ 10^{0}$
Eu-155	$8.81\ 10^{12}$	$1.19\ 10^{-14}$	$1.05\ 10^{-1}$
Gd-153	$4.83\ 10^{13}$	$3.59\ 10^{-38}$	$1.73\ 10^{-24}$
Pb-210	$4.85\ 10^{8}$	$5.99\ 10^{-11}$	$2.91\ 10^{-2}$
Po-210	$6.17\ 10^{9}$	$4.87\ 10^{-58}$	$3.01\ 10^{-48}$
Ra-226	$3.89\ 10^{6}$	$4.92\ 10^{-9}$	$1.92\ 10^{-2}$
Ra-228	$2.25\ 10^{10}$	$5.22\ 10^{-12}$	$1.17\ 10^{-1}$
Ac-227	$3.04\ 10^{9}$	$8.88\ 10^{-10}$	$2.70\ 10^{0}$
Th-228	$1.72\ 10^{11}$	$1.63\ 10^{-18}$	$2.80\ 10^{-7}$
Th-229	$2.88\ 10^{7}$	$3.04\ 10^{-9}$	$8.76\ 10^{-2}$
Th-230	$1.98\ 10^{6}$	$1.07\ 10^{-9}$	$2.11\ 10^{-3}$
Th-232	$7.95\ 10^{6}$	$4.15\ 10^{-9}$	$3.30\ 10^{-2}$
Pa-231	$1.36\ 10^{7}$	$6.54\ 10^{-9}$	$8.89\ 10^{-2}$
U-232	$4.04\ 10^{8}$	$4.40\ 10^{-10}$	$1.78\ 10^{-1}$
U-233	$1.02\ 10^{8}$	$1.13\ 10^{-10}$	$1.15\ 10^{-2}$
U-234	$1.45\ 10^{8}$	$9.34\ 10^{-11}$	$1.36\ 10^{-2}$
U-235	$6.93\ 10^{7}$	$3.79\ 10^{-10}$	$2.62\ 10^{-2}$
U-236	$1.48\ 10^{9}$	$8.62\ 10^{-11}$	$1.28\ 10^{-1}$
U-238	$1.60\ 10^{9}$	$8.04\ 10^{-11}$	$1.29\ 10^{-1}$
Np-237	$1.42\ 10^{7}$	$5.02\ 10^{-10}$	$7.11\ 10^{-3}$
Pu-238	$7.56\ 10^{8}$	$6.42\ 10^{-10}$	$4.85\ 10^{-1}$
Pu-239	$1.55\ 10^{8}$	$1.12\ 10^{-9}$	$1.75\ 10^{-1}$
Pu-240	$1.89\ 10^{8}$	$1.12\ 10^{-9}$	$2.11\ 10^{-1}$
Pu-241	$9.39\ 10^{9}$	$2.82\ 10^{-11}$	$2.65\ 10^{-1}$
Pu-242	$1.58\ 10^{8}$	$1.03\ 10^{-9}$	$1.64\ 10^{-1}$
Pu-244	$1.26\ 10^{8}$	$1.04\ 10^{-9}$	$1.31\ 10^{-1}$
Am-241	$3.03\ 10^{8}$	$8.34\ 10^{-10}$	$2.52\ 10^{-1}$
Am-242m	$1.75\ 10^{7}$	$1.14\ 10^{-9}$	$2.00\ 10^{-2}$
Am-243	$1.46\ 10^{8}$	$9.55\ 10^{-10}$	$1.39\ 10^{-1}$
Cm-242	$1.48\ 10^{11}$	$3.28\ 10^{-12}$	$4.85\ 10^{-1}$
Cm-243	$4.89\ 10^{7}$	$2.13\ 10^{-10}$	$1.04\ 10^{-2}$
Cm-244	$1.16\ 10^{8}$	$5.65\ 10^{-11}$	$6.57\ 10^{-3}$
Cm-245	$1.26\ 10^{7}$	$1.08\ 10^{-9}$	$1.36\ 10^{-2}$
Cm-246	$1.27\ 10^{7}$	$9.12\ 10^{-10}$	$1.16\ 10^{-2}$
Cm-248	$1.45\ 10^{7}$	$3.38\ 10^{-9}$	$4.91\ 10^{-2}$

952. The doses calculated using illustrative inventories are considered further in Appendix D.

E.5.3. Trial pit excavation

E.5.3.1. Assessment Calculations for Trial Pit Excavator

953. The exposure pathways for the trial pit excavator are the same as for the borehole excavator: for details, see Section E.5.2. The differences between the two scenarios manifest themselves in the duration of intrusion, depth of intrusion and the quantity of material recovered. These parameters are summarised in Table 134. All other parameters remain the same. The calculation is cautious in the same sense as the borehole excavation scenario – see Section E.5.2.

Table 134 Parameters for trial pit excavation

Parameter	Units	Value	Description
T	h y^{-1}	1	Time the excavator is exposed to excavated material (per event)
$V_{excavate}$	m^3	10	Volume of excavated material
$N_{intrusion}$		20	Number of intrusions (assumed to take place in the same landfill area)

Values taken from (Hicks & Baldwin, 2011)

954. This scenario has also been used to consider both the consignment tonnage limit and the specific activity limits applied in the CFA. In the first set of calculations a consignment is assumed to have a specific activity of 200 Bq g^{-1}, weigh 10 t and comprise 10 packages. It is also assumed that excavator is exposed to this single group of packages for 20 hours. The calculations were then repeated for different specific activities in the consignment in order to determine specific activity limits for each radionuclide. Further details are given in Section E.6.5.

E.5.3.2. Dose to Trial Pit Excavator on site after 60 years

955. The largest dose rates per MBq disposal for the trial pit excavator under this scenario are from Nb-94, Ra-226, Pa-231 and Th-232. Note that the specific doses calculated for this scenario are smaller than those calculated for the borehole drill operative scenario (see Table 135). This is because the borehole drilling is of longer duration than the trial pit excavation and the borehole drill operative is therefore exposed to contaminated material for longer. The radiological capacity calculations do not therefore consider this scenario, which results in a lower dose to workers than the borehole drill operative scenario.

956. The calculated doses to a trial pit excavator who is exposed to a single 10 t consignment containing waste at 200 Bq g^{-1} are shown in the last column of Table 135. The largest dose from a consignment containing a maximum specific activity of 200

Bq g^{-1} is 2.5 mSv y^{-1} for Th-232, followed closely by Nb-94, Ra-226 and Pa-231. Hence, a restriction on the activity concentration in a consignment of 200 Bq g^{-1} will protect the trial pit excavator for all radionuclides. The impact of a nuclide specific restriction on the activity concentration in a consignment is addressed in Section E.6.5.

Table 135 Dose to Trial pit excavator at the site

Radionuclide	Radiological capacity (MBq)	Dose to trial pit excavator (60y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})	Dose to Trial pit excavator – 10 t waste at 200 Bq g^{-1} (mSv y^{-1})
H-3	$6.43\ 10^{9}$	$4.43\ 10^{-17}$	$2.85\ 10^{-7}$	$5.64\ 10^{-8}$
C-14	$1.87\ 10^{8}$	$4.42\ 10^{-14}$	$8.28\ 10^{-6}$	$5.62\ 10^{-5}$
Cl-36	$1.56\ 10^{8}$	$3.33\ 10^{-13}$	$5.20\ 10^{-5}$	$4.10\ 10^{-4}$
Ca-41	$5.77\ 10^{9}$	$7.87\ 10^{-15}$	$4.54\ 10^{-5}$	$1.00\ 10^{-5}$
Mn-54	$1.12\ 10^{13}$	$3.86\ 10^{-31}$	$4.33\ 10^{-18}$	$4.70\ 10^{-22}$
Fe-55	$1.86\ 10^{13}$	$3.86\ 10^{-21}$	$7.16\ 10^{-8}$	$4.92\ 10^{-12}$
Co-60	$3.58\ 10^{11}$	$6.15\ 10^{-13}$	$2.20\ 10^{-1}$	$7.49\ 10^{-4}$
Ni-59	$1.95\ 10^{11}$	$6.03\ 10^{-15}$	$1.18\ 10^{-3}$	$7.67\ 10^{-6}$
Ni-63	$2.42\ 10^{11}$	$5.93\ 10^{-15}$	$1.43\ 10^{-3}$	$7.54\ 10^{-6}$
Zn-65	$8.95\ 10^{11}$	$3.38\ 10^{-37}$	$3.02\ 10^{-25}$	$4.12\ 10^{-28}$
Se-79	$8.98\ 10^{8}$	$1.40\ 10^{-13}$	$1.26\ 10^{-4}$	$1.78\ 10^{-4}$
Sr-90	$3.83\ 10^{8}$	$9.82\ 10^{-13}$	$3.76\ 10^{-4}$	$1.22\ 10^{-3}$
Mo-93	$1.44\ 10^{9}$	$1.86\ 10^{-13}$	$2.67\ 10^{-4}$	$2.33\ 10^{-4}$
Zr-93	$3.12\ 10^{11}$	$1.04\ 10^{-13}$	$3.23\ 10^{-2}$	$1.32\ 10^{-4}$
Nb-93m	$5.06\ 10^{10}$	$1.48\ 10^{-15}$	$7.48\ 10^{-5}$	$1.84\ 10^{-6}$
Nb-94	$6.09\ 10^{6}$	$9.79\ 10^{-10}$	$5.96\ 10^{-3}$	$1.19\ 10^{0}$
Tc-99	$6.12\ 10^{8}$	$8.05\ 10^{-14}$	$4.93\ 10^{-5}$	$1.02\ 10^{-4}$
Ru-106	$9.14\ 10^{11}$	$2.90\ 10^{-28}$	$2.65\ 10^{-16}$	$3.53\ 10^{-19}$
Ag-108m	$2.65\ 10^{8}$	$8.84\ 10^{-10}$	$2.35\ 10^{-1}$	$1.08\ 10^{0}$
Ag-110m	$6.41\ 10^{12}$	$6.83\ 10^{-36}$	$4.38\ 10^{-23}$	$8.31\ 10^{-27}$
Cd-109	$1.04\ 10^{12}$	$8.02\ 10^{-27}$	$8.35\ 10^{-15}$	$9.80\ 10^{-18}$
Sb-125	$4.17\ 10^{11}$	$7.06\ 10^{-17}$	$2.94\ 10^{-5}$	$8.59\ 10^{-8}$
Sn-119m	$8.43\ 10^{12}$	$1.03\ 10^{-35}$	$8.65\ 10^{-23}$	$1.25\ 10^{-26}$
Sn-123	$2.97\ 10^{12}$	$4.55\ 10^{-63}$	$1.35\ 10^{-50}$	$5.54\ 10^{-54}$
Sn-126	$4.60\ 10^{6}$	$1.20\ 10^{-9}$	$5.52\ 10^{-3}$	$1.46\ 10^{0}$
Te-127m	$4.07\ 10^{12}$	$4.52\ 10^{-75}$	$1.84\ 10^{-62}$	$5.54\ 10^{-66}$
I-129	$3.01\ 10^{8}$	$5.72\ 10^{-12}$	$1.72\ 10^{-3}$	$7.21\ 10^{-3}$
Ba-133	$7.18\ 10^{9}$	$3.85\ 10^{-12}$	$2.77\ 10^{-2}$	$4.69\ 10^{-3}$
Cs-134	$1.01\ 10^{11}$	$1.72\ 10^{-18}$	$1.74\ 10^{-7}$	$2.10\ 10^{-9}$
Cs-135	$1.55\ 10^{9}$	$1.11\ 10^{-13}$	$1.72\ 10^{-4}$	$1.41\ 10^{-4}$
Cs-137	$9.69\ 10^{8}$	$8.71\ 10^{-11}$	$8.43\ 10^{-2}$	$1.06\ 10^{-1}$
Ce-144	$4.81\ 10^{12}$	$5.37\ 10^{-35}$	$2.59\ 10^{-22}$	$6.56\ 10^{-26}$
Pm-147	$2.14\ 10^{13}$	$4.77\ 10^{-21}$	$1.02\ 10^{-7}$	$6.04\ 10^{-12}$
Sm-147	$4.81\ 10^{8}$	$2.36\ 10^{-11}$	$1.14\ 10^{-2}$	$3.01\ 10^{-2}$
Sm-151	$7.23\ 10^{11}$	$8.33\ 10^{-15}$	$6.03\ 10^{-3}$	$1.06\ 10^{-5}$
Eu-152	$8.05\ 10^{9}$	$3.29\ 10^{-11}$	$2.65\ 10^{-1}$	$4.01\ 10^{-2}$
Eu-154	$4.18\ 10^{10}$	$6.15\ 10^{-12}$	$2.57\ 10^{-1}$	$7.49\ 10^{-3}$
Eu-155	$8.81\ 10^{12}$	$2.97\ 10^{-15}$	$2.62\ 10^{-2}$	$3.62\ 10^{-6}$
Gd-153	$4.83\ 10^{13}$	$8.97\ 10^{-39}$	$4.33\ 10^{-25}$	$1.09\ 10^{-29}$

Radionuclide	Radiological capacity (MBq)	Dose to trial pit excavator (60y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})	Dose to Trial pit excavator – 10 t waste at 200 Bq g^{-1} (mSv y^{-1})
Pb-210	$4.85\ 10^{8}$	$1.50\ 10^{-11}$	$7.27\ 10^{-3}$	$1.91\ 10^{-2}$
Po-210	$6.17\ 10^{9}$	$1.22\ 10^{-58}$	$7.52\ 10^{-49}$	$1.55\ 10^{-49}$
Ra-226	$3.89\ 10^{6}$	$1.23\ 10^{-9}$	$4.79\ 10^{-3}$	$1.51\ 10^{0}$
Ra-228	$2.25\ 10^{10}$	$1.31\ 10^{-12}$	$2.94\ 10^{-2}$	$1.60\ 10^{-3}$
Ac-227	$3.04\ 10^{9}$	$2.22\ 10^{-10}$	$6.75\ 10^{-1}$	$2.81\ 10^{-1}$
Th-228	$1.72\ 10^{11}$	$4.07\ 10^{-19}$	$6.99\ 10^{-8}$	$4.97\ 10^{-10}$
Th-229	$2.88\ 10^{7}$	$7.60\ 10^{-10}$	$2.19\ 10^{-2}$	$9.59\ 10^{-1}$
Th-230	$1.98\ 10^{6}$	$2.67\ 10^{-10}$	$5.28\ 10^{-4}$	$3.38\ 10^{-1}$
Th-232	$7.95\ 10^{6}$	$1.04\ 10^{-9}$	$8.25\ 10^{-3}$	$1.29\ 10^{0}$
Pa-231	$1.36\ 10^{7}$	$1.64\ 10^{-9}$	$2.22\ 10^{-2}$	$2.07\ 10^{0}$
U-232	$4.04\ 10^{8}$	$1.10\ 10^{-10}$	$4.44\ 10^{-2}$	$1.40\ 10^{-1}$
U-233	$1.02\ 10^{8}$	$2.82\ 10^{-11}$	$2.88\ 10^{-3}$	$3.58\ 10^{-2}$
U-234	$1.45\ 10^{8}$	$2.34\ 10^{-11}$	$3.39\ 10^{-3}$	$2.97\ 10^{-2}$
U-235	$6.93\ 10^{7}$	$9.46\ 10^{-11}$	$6.56\ 10^{-3}$	$1.16\ 10^{-1}$
U-236	$1.48\ 10^{9}$	$2.16\ 10^{-11}$	$3.19\ 10^{-2}$	$2.74\ 10^{-2}$
U-238	$1.60\ 10^{9}$	$2.01\ 10^{-11}$	$3.23\ 10^{-2}$	$2.56\ 10^{-2}$
Np-237	$1.42\ 10^{7}$	$1.25\ 10^{-10}$	$1.78\ 10^{-3}$	$1.59\ 10^{-1}$
Pu-238	$7.56\ 10^{8}$	$1.61\ 10^{-10}$	$1.21\ 10^{-1}$	$2.04\ 10^{-1}$
Pu-239	$1.55\ 10^{8}$	$2.81\ 10^{-10}$	$4.37\ 10^{-2}$	$3.58\ 10^{-1}$
Pu-240	$1.89\ 10^{8}$	$2.80\ 10^{-10}$	$5.28\ 10^{-2}$	$3.56\ 10^{-1}$
Pu-241	$9.39\ 10^{9}$	$7.05\ 10^{-12}$	$6.62\ 10^{-2}$	$8.97\ 10^{-3}$
Pu-242	$1.58\ 10^{8}$	$2.58\ 10^{-10}$	$4.09\ 10^{-2}$	$3.29\ 10^{-1}$
Pu-244	$1.26\ 10^{8}$	$2.60\ 10^{-10}$	$3.27\ 10^{-2}$	$3.31\ 10^{-1}$
Am-241	$3.03\ 10^{8}$	$2.08\ 10^{-10}$	$6.31\ 10^{-2}$	$2.65\ 10^{-1}$
Am-242m	$1.75\ 10^{7}$	$2.86\ 10^{-10}$	$5.00\ 10^{-3}$	$3.64\ 10^{-1}$
Am-243	$1.46\ 10^{8}$	$2.39\ 10^{-10}$	$3.48\ 10^{-2}$	$3.03\ 10^{-1}$
Cm-242	$1.48\ 10^{11}$	$8.20\ 10^{-13}$	$1.21\ 10^{-1}$	$1.04\ 10^{-3}$
Cm-243	$4.89\ 10^{7}$	$5.34\ 10^{-11}$	$2.61\ 10^{-3}$	$6.71\ 10^{-2}$
Cm-244	$1.16\ 10^{8}$	$1.41\ 10^{-11}$	$1.64\ 10^{-3}$	$1.80\ 10^{-2}$
Cm-245	$1.26\ 10^{7}$	$2.70\ 10^{-10}$	$3.40\ 10^{-3}$	$3.42\ 10^{-1}$
Cm-246	$1.27\ 10^{7}$	$2.28\ 10^{-10}$	$2.89\ 10^{-3}$	$2.90\ 10^{-1}$
Cm-248	$1.45\ 10^{7}$	$8.45\ 10^{-10}$	$1.23\ 10^{-2}$	$1.08\ 10^{0}$

* Assumes Ra-226 distributed with other LLW.

957. This scenario is one of the scenarios used to determine the proposed radionuclide activity concentration limits for packaged wastes (see Section 7.4.2.3 for further details).

E.5.4. Excavation for housing or road – excavator

E.5.4.1. Assessment calculations for Housing or Road Excavator

958. The exposure pathways for the house/road excavator are the same as for the borehole excavator: for details, see Section E.5.2. The differences between the two scenarios manifest themselves in the duration of intrusion, depth of intrusion, and the quantity of

material recovered. These differences are summarised in Table 136. All other parameters remain the same. The calculation is cautious in the same sense as the borehole excavation scenario since it ignores the uncontaminated cap material that will also be excavated - see Section E.5.2.

Table 136 Parameters for house/road excavation

Parameter	Units	Value	Description
T	h y^{-1}	80	Time the excavator is exposed to excavated material
$V_{excavate}$	m^3	2000	Volume of excavated material

Values taken from (Hicks & Baldwin, 2011).

E.5.4.2. Dose to Excavator for Housing or Road on site after 150 years

959. The largest dose rates per MBq disposal for the person excavating the site for housing etc in 150 years are for Nb-94, Ra-226, Pa-231 and Th-232 (Table 137). These radionuclides will correspondingly have the smallest radiological capacities for this scenario. In most cases the dose rates to the borehole drill operator are greater than to the house or road excavator, the exceptions are Pa-231, Th-230, U-233, U-234 and U-235 where daughter ingrowth at 150 years increases the dose rate compared to that at 60 years. Note that the specific doses and radiological capacities for this scenario would be identical to those calculated for a borehole excavator making 5 intrusions (see Table 137) except for the timing of the intrusion event. This is because the dose (and hence derived quantities such as the radiological capacity) depends upon the duration of exposure and the activity concentration, not the volume of excavated material. In this note, both of these scenarios use exposure times of 80 hours per year to contaminated material, and hence the doses are the same. The excavation for housing or a road (150 years) is assumed to occur later than the borehole drilling scenario (60 years) and radioactive decay reduces the doses expected for most radionuclides. The impact of Radium placement depth within Port Clarence on these intrusion doses and on radon release is discussed in the next section (see paragraph 1013).

Table 137 Dose to Housing site/road excavator at the site

Radionuclide	Radiological capacity (MBq)	Dose to Housing site excavator (150y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})
H-3	$6.43\ 10^{9}$	$1.12\ 10^{-18}$	$7.20\ 10^{-9}$
C-14	$1.87\ 10^{8}$	$1.75\ 10^{-13}$	$3.28\ 10^{-5}$
Cl-36	$1.56\ 10^{8}$	$1.33\ 10^{-12}$	$2.08\ 10^{-4}$
Ca-41	$5.77\ 10^{9}$	$3.14\ 10^{-14}$	$1.81\ 10^{-4}$
Mn-54	$1.12\ 10^{13}$	$3.11\ 10^{-62}$	$3.49\ 10^{-49}$
Fe-55	$1.86\ 10^{13}$	$1.95\ 10^{-30}$	$3.62\ 10^{-17}$
Co-60	$3.58\ 10^{11}$	$1.78\ 10^{-17}$	$6.38\ 10^{-6}$
Ni-59	$1.95\ 10^{11}$	$2.41\ 10^{-14}$	$4.71\ 10^{-3}$

Radionuclide	Radiological capacity (MBq)	Dose to Housing site excavator (150y) ($mSv\ y^{-1}\ MBq^{-1}$)	Dose from radiological capacity ($mSv\ y^{-1}$)
Ni-63	$2.42\ 10^{11}$	$1.27\ 10^{-14}$	$3.08\ 10^{-3}$
Zn-65	$8.95\ 10^{11}$	$3.66\ 10^{-77}$	$3.28\ 10^{-65}$
Se-79	$8.98\ 10^{8}$	$5.61\ 10^{-13}$	$5.04\ 10^{-4}$
Sr-90	$3.83\ 10^{8}$	$4.50\ 10^{-13}$	$1.72\ 10^{-4}$
Mo-93	$1.44\ 10^{9}$	$7.31\ 10^{-13}$	$1.05\ 10^{-3}$
Zr-93	$3.12\ 10^{11}$	$4.14\ 10^{-13}$	$1.29\ 10^{-1}$
Nb-93m	$5.06\ 10^{10}$	$1.24\ 10^{-16}$	$6.25\ 10^{-6}$
Nb-94	$6.09\ 10^{6}$	$3.90\ 10^{-9}$	$2.38\ 10^{-2}$
Tc-99	$6.12\ 10^{8}$	$3.22\ 10^{-13}$	$1.97\ 10^{-4}$
Ru-106	$9.14\ 10^{11}$	$3.81\ 10^{-54}$	$3.48\ 10^{-42}$
Ag-108m	$2.65\ 10^{8}$	$3.05\ 10^{-9}$	$8.09\ 10^{-1}$
Ag-110m	$6.41\ 10^{12}$	$6.71\ 10^{-75}$	$4.30\ 10^{-62}$
Cd-109	$1.04\ 10^{12}$	$1.14\ 10^{-47}$	$1.18\ 10^{-35}$
Sb-125	$4.17\ 10^{11}$	$4.28\ 10^{-26}$	$1.78\ 10^{-14}$
Sn-119m	$8.43\ 10^{12}$	$7.10\ 10^{-69}$	$5.98\ 10^{-56}$
Sn-123	$2.97\ 10^{12}$	$4.66\ 10^{-139}$	$1.38\ 10^{-126}$
Sn-126	$4.60\ 10^{6}$	$4.80\ 10^{-9}$	$2.21\ 10^{-2}$
Te-127m	$4.07\ 10^{12}$	$9.77\ 10^{-168}$	$3.97\ 10^{-155}$
I-129	$3.01\ 10^{8}$	$2.29\ 10^{-11}$	$6.88\ 10^{-3}$
Ba-133	$7.18\ 10^{9}$	$4.10\ 10^{-14}$	$2.94\ 10^{-4}$
Cs-134	$1.01\ 10^{11}$	$5.22\ 10^{-31}$	$5.28\ 10^{-20}$
Cs-135	$1.55\ 10^{9}$	$4.43\ 10^{-13}$	$6.87\ 10^{-4}$
Cs-137	$9.69\ 10^{8}$	$4.40\ 10^{-11}$	$4.27\ 10^{-2}$
Ce-144	$4.81\ 10^{12}$	$3.98\ 10^{-69}$	$1.92\ 10^{-56}$
Pm-147	$2.14\ 10^{13}$	$8.95\ 10^{-31}$	$1.92\ 10^{-17}$
Sm-147	$4.81\ 10^{8}$	$9.46\ 10^{-11}$	$4.54\ 10^{-2}$
Sm-151	$7.23\ 10^{11}$	$1.67\ 10^{-14}$	$1.21\ 10^{-2}$
Eu-152	$8.05\ 10^{9}$	$1.31\ 10^{-12}$	$1.06\ 10^{-2}$
Eu-154	$4.18\ 10^{10}$	$1.73\ 10^{-14}$	$7.24\ 10^{-4}$
Eu-155	$8.81\ 10^{12}$	$2.43\ 10^{-20}$	$2.14\ 10^{-7}$
Gd-153	$4.83\ 10^{13}$	$2.46\ 10^{-79}$	$1.19\ 10^{-65}$
Pb-210	$4.85\ 10^{8}$	$3.61\ 10^{-12}$	$1.75\ 10^{-3}$
Po-210	$6.17\ 10^{9}$	$1.53\ 10^{-129}$	$9.44\ 10^{-120}$
Ra-226	$3.89\ 10^{6}$	$4.74\ 10^{-9}$	$1.84\ 10^{-2}$
Ra-228	$2.25\ 10^{10}$	$1.01\ 10^{-16}$	$2.28\ 10^{-6}$
Ac-227	$3.04\ 10^{9}$	$5.06\ 10^{-11}$	$1.54\ 10^{-1}$
Th-228	$1.72\ 10^{11}$	$1.10\ 10^{-32}$	$1.89\ 10^{-21}$
Th-229	$2.88\ 10^{7}$	$3.02\ 10^{-9}$	$8.68\ 10^{-2}$
Th-230	$1.98\ 10^{6}$	$1.25\ 10^{-9}$	$2.48\ 10^{-3}$
Th-232	$7.95\ 10^{6}$	$4.15\ 10^{-9}$	$3.30\ 10^{-2}$
Pa-231	$1.36\ 10^{7}$	$7.37\ 10^{-9}$	$1.00\ 10^{-1}$
U-232	$4.04\ 10^{8}$	$1.78\ 10^{-10}$	$7.18\ 10^{-2}$
U-233	$1.02\ 10^{8}$	$1.39\ 10^{-10}$	$1.42\ 10^{-2}$
U-234	$1.45\ 10^{8}$	$9.42\ 10^{-11}$	$1.37\ 10^{-2}$
U-235	$6.93\ 10^{7}$	$3.81\ 10^{-10}$	$2.64\ 10^{-2}$
U-236	$1.48\ 10^{9}$	$8.62\ 10^{-11}$	$1.28\ 10^{-1}$
U-238	$1.60\ 10^{9}$	$8.04\ 10^{-11}$	$1.29\ 10^{-1}$

Radionuclide	Radiological capacity (MBq)	Dose to Housing site excavator (150y) (mSv y^{-1} MBq^{-1})	Dose from radiological capacity (mSv y^{-1})
Np-237	1.42 10^{7}	5.02 10^{-10}	7.12 10^{-3}
Pu-238	7.56 10^{8}	3.15 10^{-10}	2.38 10^{-1}
Pu-239	1.55 10^{8}	1.12 10^{-9}	1.74 10^{-1}
Pu-240	1.89 10^{8}	1.11 10^{-9}	2.09 10^{-1}
Pu-241	9.39 10^{9}	2.49 10^{-11}	2.34 10^{-1}
Pu-242	1.58 10^{8}	1.03 10^{-9}	1.64 10^{-1}
Pu-244	1.26 10^{8}	1.05 10^{-9}	1.32 10^{-1}
Am-241	3.03 10^{8}	7.22 10^{-10}	2.18 10^{-1}
Am-242m	1.75 10^{7}	1.01 10^{-9}	1.76 10^{-2}
Am-243	1.46 10^{8}	9.49 10^{-10}	1.39 10^{-1}
Cm-242	1.48 10^{11}	1.61 10^{-12}	2.38 10^{-1}
Cm-243	4.89 10^{7}	2.62 10^{-11}	1.28 10^{-3}
Cm-244	1.16 10^{8}	4.78 10^{-12}	5.55 10^{-4}
Cm-245	1.26 10^{7}	1.07 10^{-9}	1.35 10^{-2}
Cm-246	1.27 10^{7}	9.00 10^{-10}	1.14 10^{-2}
Cm-248	1.45 10^{7}	3.38 10^{-9}	4.91 10^{-2}

* Assumes Ra-226 distributed with other LLW.

E.5.5. Site Resident – no cap damage

E.5.5.1. Assessment calculations for Site Residents (no cap damage)

960. Members of the public living in a house built close to, or on, the site after closure is also considered. The house is assumed to be built 150 years after closure in such a way, e.g. on a concrete raft, that it does not damage the integrity of the cap. The situation where the cap is damaged or a house is built on excavated spoil is considered in Section E.5.7. External irradiation from the buried wastes and inhalation of radioactive gases released through the cap are considered. Habit data are presented in Table 138.

Table 138 Habit data for site resident family

Parameter	Units	Value	Description
B_a	m^3 h^{-1}	1	Inhalation rate - adult
B_c	m^3 h^{-1}	0.64	Inhalation rate - child
B_i	m^3 h^{-1}	0.22	Inhalation rate - infant
$O_{in,a}$	h y^{-1}	7012.8	Indoor occupancy – adult (0.8 indoors)
$O_{in,c}$	h y^{-1}	7366.0	Indoor occupancy – adult (0.84 indoors)
$O_{in,i}$	h y^{-1}	7978.0	Indoor occupancy – adult (0.91 indoors)

961. The calculations consider the release of H-3, C-14 and radon gases. The doses are summed with the doses from external irradiation that could occur through the intact

cap. With the exception of radon exposure, the impact of these exposure pathways is expected to be low. Exposure to gas is only considered while the person is indoors since when outdoors there would be significant dilution in the atmosphere, leading to negligible doses in comparison.

Gas generation – H-3 and C-14

962. The gas pathway is considered in the same way for tritium and C-14. The release rate of radioactive gas is given in paragraph 613 using the release fractions and initial activity values in Table 69.

963. The release rate of gases from a landfill is expected to vary over time. A conservative assumption for the operational period assumed all C-14 and H-3 that was associated with organic material would be released over a ten year period. Gas generation within the landfill has been simulated using the GasSim model (Augean, 2010) which shows a rapid build-up in the rate of release after capping followed by an exponential decline. It was shown that 85% of the gas yield for carbon occurs within 60 years and it is assumed that the remainder is released at a slower rate. We have cautiously assumed this lower rate remains constant until the period of interest i.e. for a further 90 years. The average timescale for carbon-based gas generation has therefore been set to 600 for this scenario (90/0.15). For H-3, the default SNIFFER value of 50 is used.

964. The effective doses arising from inhalation of generated gases are calculated for site residents post-closure (at *t* = 150 years), assuming that a proportion of a resident's time is spent indoors (as detailed in Table 138). The dose is calculated according to:

$$Dose_{gas,indoors} = D_{inh,j}^{Rn} \cdot B_j \cdot O_{in,j} \left[R_{Rn,gas}(t) \cdot \frac{a_H}{a} \cdot \left(\frac{1}{kV} \right) \right]$$

where:

- $D_{inh,j}^{Rn}$ is the inhalation dose coefficient of radionuclide *Rn* for age group j (Sv Bq^{-1});
- B_j is the inhalation rate for age group j (m^3 h^{-1});
- $O_{in,j}$ is the occupancy indoors for age group j (h y^{-1});
- $R_{Rn,gas}(t)$ is the release rate of radioactive gas at time *t* (Bq y^{-1});
- $\frac{a_H}{a}$ is the horizontal area of a dwelling divided by the area over which the radioactive gas is being released (i.e. the facility footprint);
- k is the turnover rate, accounting for gas release from the house by ventilation (y^{-1}); and,
- V is the volume of the house (m^3).

965. The gas dispersion parameters used in this work are summarised in Table 139, the dimensions of the landfill are given in Table 61, the dose coefficients in Table 200 and habit data in Table 138.

Table 139 Gas dispersion parameters

Parameter	Units	Value	Description
a_H	m^2	50	Area of dwelling
k	y^{-1}	2600	Turnover rate
V	m^3	125	House volume

Gas generation – Radon

966. This section considers migration of radon gas from a waste cell into a building constructed on the intact cap.

967. This case considers long-term occupation of the former landfill site, and thus long-term potential exposure to contaminated wastes. The flux of radon through soil ($F_{radon}(t)$) is described by the equation given in paragraph 622. The parameters in Table 72 were used for the building located on an intact cap with the exception of h_2 which was set to the intact cap depth (1.6 m) plus the depth of material above the LLW (1 m) i.e. a total of 2.6 m.

968. The activity concentration of radon gas in the house, $C_{Rn-222,house}$ (Bq m^{-3}) is then calculated according to (SNIFFER, 2006):

$$C_{Rn-222,house} = F_{radon}(t) \cdot \frac{a_H}{AREA} \cdot \frac{1}{(\lambda_{house} \cdot v_{house})}$$

where:

- a_H is the area of the house (m^2);
- $AREA$ is the surface area of that part of the landfill facility containing radioactive waste, 273,468 m^2;
- λ_{house} is the turnover rate of air in the house (y^{-1}); and
- v_{house} is the volume of the house (m^3).

969. The values of the quantities used in this work are given in Table 140, except for the landfill area (see Table 61).

Table 140 Radon parameters

Parameter	Units	Value	Description	Source
λ_{house}	y^{-1}	2600	Air turnover rate in house	(Passive House Institute, 2012)
v_{house}	m^3	125	Volume of house	(HPA, 2007)
a_H	m^2	50	Area of house	(Quintessa Ltd, 2011)

970. The resultant inhalation dose (Sv y^{-1}) to a resident of the house is then given by:

$$Dose = D_{inh,j} \cdot C_{Rn-222,house} \cdot B_{inh,j} \cdot O_{indoor,j}$$

where:

- $D_{inh,j}$ is the inhalation dose coefficient for age group j (Sv Bq^{-1});
- $B_{inh,j}$ is the breathing rate for age group j (m^3 h^{-1}); and,
- $O_{indoor,j}$ is the indoor occupancy for age group j (h y^{-1}).

971. The dose coefficient is presented in Table 73 and habit data in Table 138.

External irradiation

972. The dose to a future site resident from external irradiation is also calculated through the intact cap assuming that a proportion of a resident's time is spent indoors (as set out in Table 138). The dose is calculated according to:

$$Dose_{irr} = D_{irr,slab}^{Rn} \cdot \left(O_{out,j} + O_{in,j} sf\right) \cdot \left(\frac{A_{Rn,waste}(t)}{V_{waste}\rho_{waste}}\right) \cdot e^{-\mu^{Rn} \cdot x}$$

where:

- $D_{irr,slab}^{Rn}$ is the dose conversion factor for irradiation from radionuclide *Rn* (see 0), based on the receptor being 1 m from the ground, and the contamination is taken to be a semi-infinite slab (Sv y^{-1} Bq^{-1} kg);
- $O_{out,j}$ is the outdoor occupancy for age group j;
- $O_{in,j}$ is the indoor occupancy for age group j;
- sf is the shielding factor from the ground when indoors;
- $A_{Rn,waste}(t)$ is the activity of radionuclide *Rn* at time *t*;
- V_{waste} is the volume of waste (m^3);
- ρ_{waste} is the density of waste (kg m^{-3});
- μ^{Rn} is the linear attenuation coefficient for radionuclide *Rn* (see Table 200); and,
- x is the thickness of the cap and cover material (m).

973. The values of these parameters employed in this work are summarised in Table 141 unless stated otherwise. The model uses the linear attenuation coefficient to account for shielding by clean material above the waste mass; the greater the depth of clean material, the greater the shielding. Note that since the linear attenuation coefficient is dependent on the density of the material, the mass attenuation coefficient (μ^{Rn} / the density of the material) is often reported for convenience. The linear attenuation coefficients used in the model are taken from (SNIFFER, 2006) and are the recommended values for soil given by (Hung, 2000).

Table 141 External irradiation parameters

Parameter	Units	Value	Description
$O_{out,a}$		0.25	Outdoor occupancy - adult
$O_{in,a}$		0.75	Indoor occupancy - adult
$O_{out,c}$		0.16	Outdoor occupancy - child
$O_{in,c}$		0.84	Indoor occupancy - child
$O_{out,i}$		0.09	Outdoor occupancy - infant
$O_{in,i}$		0.91	Indoor occupancy - infant
sf		0.1	Shielding factor
V_{waste}	m^3	$4.16\ 10^6$	Waste volume
ρ_{waste}	kg m^{-3}	1530	Waste density
x	m	2.6	Cap plus cover thickness

E.5.5.2. Dose to Site Resident – no cap damage

974. Doses to a site resident from radon are sensitive to the depth of radium placement beneath the surface of the landfill. This is shown in Table 142 (taken from (Eden NE, 2015a) which give radon fluxes, indoor Rn-222 activity concentrations and inhalation doses at 150 years arising from a nominal 1 MBq of Ra-226 in the landfill. These calculations assume that the released radon is in secular equilibrium with the parent radium.

975. As the placement depth decreases the estimated dose increases and the radiological capacity decreases. Hence, an emplacement strategy for Ra-226 wastes will have a significant effect on the radon doses.

Table 142 Radon inhalation doses for a dwelling built on a capped landfill – unit inventory

Depth below ground level	Rn-222 flux into house (Bq y^{-1})	Indoor Rn-222 activity concentration (Bq m^{-3})	Adult inhalation effective dose (µSv y^{-1} MBq^{-1})
6.6	$9.56\ 10^{-10}$	$1.03\ 10^{-18}$	$4.31\ 10^{-17}$
5.6	$1.28\ 10^{-7}$	$1.37\ 10^{-16}$	$5.77\ 10^{-15}$
5	$2.43\ 10^{-6}$	$2.60\ 10^{-15}$	$1.09\ 10^{-13}$
4.6	$1.73\ 10^{-5}$	$1.85\ 10^{-14}$	$7.80\ 10^{-13}$
3.6	$2.35\ 10^{-3}$	$2.53\ 10^{-12}$	$1.06\ 10^{-10}$
2.6*	$3.23\ 10^{-1}$	$3.46\ 10^{-10}$	$1.46\ 10^{-8}$
1.6	$4.45\ 10^{1}$	$4.78\ 10^{-8}$	$2.01\ 10^{-6}$

976. The doses to site residents (150 years after closure and with the cap intact) from gas released from the ENRMF and through external irradiation are presented in Table 143, Table 144 and Table 145 for the adult, child and infant groups respectively. Note that these results include the effects of ingrowth after 150 years upon the calculated doses.

977. The expected dose if each radionuclide is disposed at the maximum inventory is shown in the right hand column. The highest dose is to an infant from Ra-226 (879 µSv y^{-1})

with no emplacement strategy; if an emplacement strategy is applied to wastes containing Ra-226 at > 5 Bq g^{-1} then the dose is below 1 µSv y^{-1} (deep placement is shown at bottom of table). The highest dose to an adult is 109 µSv y^{-1} from Ra-226 with no emplacement strategy for waste containing > 5 Bq g^{-1}. The next highest dose to an adult is from C-14 gas (99 µSv y^{-1}), but the highest dose from waste disposed of at Port Clarence will always be lower than this due to application of the sum of fractions approach. Similar patterns are seen for the dose to child and infant. All other doses are equal to or below 10 µSv y^{-1}.

Table 143 Adult Site Resident Exposure

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from radiological capacity
		Gas	External	Total	
H-3	$6.43\ 10^{9}$	$1.62\ 10^{-10}$	0	$1.62\ 10^{-10}$	$1.04\ 10^{0}$
C-14	$1.87\ 10^{8}$	$2.11\ 10^{-6}$	$2.76\ 10^{-66}$	$2.11\ 10^{-6}$	$3.95\ 10^{2}$
Cl-36	$1.56\ 10^{8}$	0	$1.44\ 10^{-28}$	$1.44\ 10^{-28}$	$2.26\ 10^{-20}$
Ca-41	$5.77\ 10^{9}$	0	0	0	0
Mn-54	$1.12\ 10^{13}$	0	$3.02\ 10^{-71}$	$3.02\ 10^{-71}$	$3.39\ 10^{-58}$
Fe-55	$1.86\ 10^{13}$	0	0	0	0
Co-60	$3.58\ 10^{11}$	0	$6.22\ 10^{-25}$	$6.22\ 10^{-25}$	$2.23\ 10^{-13}$
Ni-59	$1.95\ 10^{11}$	0	0	0	0
Ni-63	$2.42\ 10^{11}$	0	0	0	0
Zn-65	$8.95\ 10^{11}$	0	$3.54\ 10^{-85}$	$3.54\ 10^{-85}$	$3.16\ 10^{-73}$
Se-79	$8.98\ 10^{8}$	0	$3.22\ 10^{-64}$	$3.22\ 10^{-64}$	$2.90\ 10^{-55}$
Sr-90	$3.83\ 10^{8}$	0	$3.30\ 10^{-27}$	$3.30\ 10^{-27}$	$1.27\ 10^{-18}$
Mo-93	$1.44\ 10^{9}$	0	$8.30\ 10^{-9}$	$8.30\ 10^{-9}$	$1.19\ 10^{1}$
Zr-93	$3.12\ 10^{11}$	0	0	0	0
Nb-93m	$5.06\ 10^{10}$	0	$2.38\ 10^{-12}$	$2.38\ 10^{-12}$	$1.21\ 10^{-1}$
Nb-94	$6.09\ 10^{6}$	0	$2.09\ 10^{-18}$	$2.09\ 10^{-18}$	$1.28\ 10^{-11}$
Tc-99	$6.12\ 10^{8}$	0	$6.24\ 10^{-48}$	$6.24\ 10^{-48}$	$3.82\ 10^{-39}$
Ru-106	$9.14\ 10^{11}$	0	$2.70\ 10^{-64}$	$2.70\ 10^{-64}$	$2.47\ 10^{-52}$
Ag-108m	$2.65\ 10^{8}$	0	$1.34\ 10^{-19}$	$1.34\ 10^{-19}$	$3.55\ 10^{-11}$
Ag-110m	$6.41\ 10^{12}$	0	$1.40\ 10^{-83}$	$1.40\ 10^{-83}$	$8.98\ 10^{-71}$
Cd-109	$1.04\ 10^{12}$	0	$2.29\ 10^{-84}$	$2.29\ 10^{-84}$	$2.38\ 10^{-72}$
Sb-125	$4.17\ 10^{11}$	0	$6.28\ 10^{-37}$	$6.28\ 10^{-37}$	$2.62\ 10^{-25}$
Sn-119m	$8.43\ 10^{12}$	0	0	0	0
Sn-123	$2.97\ 10^{12}$	0	$8.95\ 10^{-148}$	$8.95\ 10^{-148}$	$2.66\ 10^{-135}$
Sn-126	$4.60\ 10^{6}$	0	$4.79\ 10^{-19}$	$4.79\ 10^{-19}$	$2.20\ 10^{-12}$
Te-127m	$4.07\ 10^{12}$	0	$9.22\ 10^{-194}$	$9.22\ 10^{-194}$	$3.75\ 10^{-181}$

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose ($\mu Sv\ y^{-1}$) from radiological capacity
		Gas	External	Total	
I-129	$3.01\ 10^{8}$	0	$3.96\ 10^{-138}$	$3.96\ 10^{-138}$	$1.19\ 10^{-129}$
Ba-133	$7.18\ 10^{9}$	0	$1.80\ 10^{-27}$	$1.80\ 10^{-27}$	$1.29\ 10^{-17}$
Cs-134	$1.01\ 10^{11}$	0	$1.14\ 10^{-40}$	$1.14\ 10^{-40}$	$1.15\ 10^{-29}$
Cs-135	$1.55\ 10^{9}$	0	$1.77\ 10^{-56}$	$1.77\ 10^{-56}$	$2.76\ 10^{-47}$
Cs-137	$9.69\ 10^{8}$	0	$5.41\ 10^{-21}$	$5.41\ 10^{-21}$	$5.24\ 10^{-12}$
Ce-144	$4.81\ 10^{12}$	0	$2.71\ 10^{-92}$	$2.71\ 10^{-92}$	$1.30\ 10^{-79}$
Pm-147	$2.14\ 10^{13}$	0	$2.59\ 10^{-64}$	$2.59\ 10^{-64}$	$5.54\ 10^{-51}$
Sm-147	$4.81\ 10^{8}$	0	0	0	0
Sm-151	$7.23\ 10^{11}$	0	0	0	0
Eu-152	$8.05\ 10^{9}$	0	$4.37\ 10^{-21}$	$4.37\ 10^{-21}$	$3.52\ 10^{-11}$
Eu-154	$4.18\ 10^{10}$	0	$7.31\ 10^{-23}$	$7.31\ 10^{-23}$	$3.06\ 10^{-12}$
Eu-155	$8.81\ 10^{12}$	0	$1.97\ 10^{-49}$	$1.97\ 10^{-49}$	$1.74\ 10^{-36}$
Gd-153	$4.83\ 10^{13}$	0	$1.99\ 10^{-110}$	$1.99\ 10^{-110}$	$9.62\ 10^{-97}$
Pb-210	$4.85\ 10^{8}$	0	$1.16\ 10^{-23}$	$1.16\ 10^{-23}$	$5.61\ 10^{-15}$
Po-210	$6.17\ 10^{9}$	0	$7.21\ 10^{-143}$	$7.21\ 10^{-143}$	$4.45\ 10^{-133}$
Ra-226	$3.89\ 10^{6}$	$2.81\ 10^{-5}$	$2.30\ 10^{-16}$	$2.81\ 10^{-5}$	$1.09\ 10^{2}$
Ra-228	$2.25\ 10^{10}$	0	$1.65\ 10^{-22}$	$1.65\ 10^{-22}$	$3.71\ 10^{-12}$
Ac-227	$3.04\ 10^{9}$	0	$3.87\ 10^{-22}$	$3.87\ 10^{-22}$	$1.18\ 10^{-12}$
Th-228	$1.72\ 10^{11}$	0	$1.79\ 10^{-38}$	$1.79\ 10^{-38}$	$3.08\ 10^{-27}$
Th-229	$2.88\ 10^{7}$	0	$1.93\ 10^{-17}$	$1.93\ 10^{-17}$	$5.56\ 10^{-10}$
Th-230	$1.98\ 10^{6}$	0	$1.54\ 10^{-17}$	$1.54\ 10^{-17}$	$3.05\ 10^{-11}$
Th-232	$7.95\ 10^{6}$	0	$4.39\ 10^{-15}$	$4.39\ 10^{-15}$	$3.49\ 10^{-8}$
Pa-231	$1.36\ 10^{7}$	0	$4.52\ 10^{-20}$	$4.52\ 10^{-20}$	$6.14\ 10^{-13}$
U-232	$4.04\ 10^{8}$	0	$1.45\ 10^{-19}$	$1.45\ 10^{-19}$	$5.85\ 10^{-11}$
U-233	$1.02\ 10^{8}$	0	$2.76\ 10^{-19}$	$2.76\ 10^{-19}$	$2.83\ 10^{-11}$
U-234	$1.45\ 10^{8}$	0	$1.31\ 10^{-40}$	$1.31\ 10^{-40}$	$1.91\ 10^{-32}$
U-235	$6.93\ 10^{7}$	0	$7.74\ 10^{-28}$	$7.74\ 10^{-28}$	$5.36\ 10^{-20}$
U-236	$1.48\ 10^{9}$	0	$3.26\ 10^{-23}$	$3.26\ 10^{-23}$	$4.83\ 10^{-14}$
U-238	$1.60\ 10^{9}$	0	$3.62\ 10^{-23}$	$3.62\ 10^{-23}$	$5.80\ 10^{-14}$
Np-237	$1.42\ 10^{7}$	0	$5.37\ 10^{-26}$	$5.37\ 10^{-26}$	$7.61\ 10^{-19}$
Pu-238	$7.56\ 10^{8}$	0	$3.74\ 10^{-47}$	$3.74\ 10^{-47}$	$2.83\ 10^{-38}$
Pu-239	$1.55\ 10^{8}$	0	$7.02\ 10^{-31}$	$7.02\ 10^{-31}$	$1.09\ 10^{-22}$
Pu-240	$1.89\ 10^{8}$	0	$3.46\ 10^{-46}$	$3.46\ 10^{-46}$	$6.53\ 10^{-38}$
Pu-241	$9.39\ 10^{9}$	0	$1.72\ 10^{-43}$	$1.72\ 10^{-43}$	$1.62\ 10^{-33}$
Pu-242	$1.58\ 10^{8}$	0	$8.43\ 10^{-31}$	$8.43\ 10^{-31}$	$1.33\ 10^{-22}$

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from radiological capacity
		Gas	External	Total	
Pu-244	$1.26\ 10^{8}$	0	$8.68\ 10^{-24}$	$8.68\ 10^{-24}$	$1.09\ 10^{-15}$
Am-241	$3.03\ 10^{8}$	0	$2.32\ 10^{-30}$	$2.32\ 10^{-30}$	$7.02\ 10^{-22}$
Am-242m	$1.75\ 10^{7}$	0	$1.09\ 10^{-21}$	$1.09\ 10^{-21}$	$1.90\ 10^{-14}$
Am-243	$1.46\ 10^{8}$	0	$2.74\ 10^{-29}$	$2.74\ 10^{-29}$	$4.00\ 10^{-21}$
Cm-242	$1.48\ 10^{11}$	0	$1.79\ 10^{-49}$	$1.79\ 10^{-49}$	$2.64\ 10^{-38}$
Cm-243	$4.89\ 10^{7}$	0	$2.90\ 10^{-30}$	$2.90\ 10^{-30}$	$1.42\ 10^{-22}$
Cm-244	$1.16\ 10^{8}$	0	$2.36\ 10^{-56}$	$2.36\ 10^{-56}$	$2.74\ 10^{-48}$
Cm-245	$1.26\ 10^{7}$	0	$5.89\ 10^{-34}$	$5.89\ 10^{-34}$	$7.41\ 10^{-27}$
Cm-246	$1.27\ 10^{7}$	0	$8.55\ 10^{-69}$	$8.55\ 10^{-69}$	$1.08\ 10^{-61}$
Cm-248	$1.45\ 10^{7}$	0	$1.13\ 10^{-29}$	$1.13\ 10^{-29}$	$1.64\ 10^{-22}$
Ra-226$	$3.89\ 10^{6}$	$4.68\ 10^{-11}$	$3.74\ 10^{-30}$	$4.68\ 10^{-11}$	$1.82\ 10^{-4}$

Note: $ Ra-226 buried 5m or greater below the restored site level

Table 144 Child Site Resident Exposure

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from radiological capacity
		Gas	External	Total	
H-3	$6.43\ 10^{9}$	$1.70\ 10^{-10}$	0	$1.70\ 10^{-10}$	$1.09\ 10^{0}$
C-14	$1.87\ 10^{8}$	$1.93\ 10^{-6}$	$2.07\ 10^{-66}$	$1.93\ 10^{-6}$	$3.61\ 10^{2}$
Cl-36	$1.56\ 10^{8}$	0	$1.08\ 10^{-28}$	$1.08\ 10^{-28}$	$1.69\ 10^{-20}$
Ca-41	$5.77\ 10^{9}$	0	0	0	0
Mn-54	$1.12\ 10^{13}$	0	$2.27\ 10^{-71}$	$2.27\ 10^{-71}$	$2.54\ 10^{-58}$
Fe-55	$1.86\ 10^{13}$	0	0	0	0
Co-60	$3.58\ 10^{11}$	0	$4.67\ 10^{-25}$	$4.67\ 10^{-25}$	$1.67\ 10^{-13}$
Ni-59	$1.95\ 10^{11}$	0	0	0	0
Ni-63	$2.42\ 10^{11}$	0	0	0	0
Zn-65	$8.95\ 10^{11}$	0	$2.65\ 10^{-85}$	$2.65\ 10^{-85}$	$2.37\ 10^{-73}$
Se-79	$8.98\ 10^{8}$	0	$2.42\ 10^{-64}$	$2.42\ 10^{-64}$	$2.17\ 10^{-55}$
Sr-90	$3.83\ 10^{8}$	0	$2.48\ 10^{-27}$	$2.48\ 10^{-27}$	$9.49\ 10^{-19}$
Mo-93	$1.44\ 10^{9}$	0	$6.23\ 10^{-9}$	$6.23\ 10^{-9}$	$8.95\ 10^{0}$
Zr-93	$3.12\ 10^{11}$	0	0	0	0
Nb-93m	$5.06\ 10^{10}$	0	$1.79\ 10^{-12}$	$1.79\ 10^{-12}$	$9.04\ 10^{-2}$
Nb-94	$6.09\ 10^{6}$	0	$1.57\ 10^{-18}$	$1.57\ 10^{-18}$	$9.57\ 10^{-12}$

Radionuclide	Radiological capacity (MBq)	Dose (μSv y^{-1} MBq^{-1})			Dose (μSv y^{-1}) from radiological capacity
		Gas	External	Total	
Tc-99	$6.12\ 10^{8}$	0	$4.68\ 10^{-48}$	$4.68\ 10^{-48}$	$2.86\ 10^{-39}$
Ru-106	$9.14\ 10^{11}$	0	$2.03\ 10^{-64}$	$2.03\ 10^{-64}$	$1.85\ 10^{-52}$
Ag-108m	$2.65\ 10^{8}$	0	$1.00\ 10^{-19}$	$1.00\ 10^{-19}$	$2.67\ 10^{-11}$
Ag-110m	$6.41\ 10^{12}$	0	$1.05\ 10^{-83}$	$1.05\ 10^{-83}$	$6.74\ 10^{-71}$
Cd-109	$1.04\ 10^{12}$	0	$1.72\ 10^{-84}$	$1.72\ 10^{-84}$	$1.79\ 10^{-72}$
Sb-125	$4.17\ 10^{11}$	0	$4.71\ 10^{-37}$	$4.71\ 10^{-37}$	$1.96\ 10^{-25}$
Sn-119m	$8.43\ 10^{12}$	0	0	0	0
Sn-123	$2.97\ 10^{12}$	0	$6.71\ 10^{-148}$	$6.71\ 10^{-148}$	$1.99\ 10^{-135}$
Sn-126	$4.60\ 10^{6}$	0	$3.59\ 10^{-19}$	$3.59\ 10^{-19}$	$1.65\ 10^{-12}$
Te-127m	$4.07\ 10^{12}$	0	$6.92\ 10^{-194}$	$6.92\ 10^{-194}$	$2.81\ 10^{-181}$
I-129	$3.01\ 10^{8}$	0	$2.97\ 10^{-138}$	$2.97\ 10^{-138}$	$8.94\ 10^{-130}$
Ba-133	$7.18\ 10^{9}$	0	$1.35\ 10^{-27}$	$1.35\ 10^{-27}$	$9.68\ 10^{-18}$
Cs-134	$1.01\ 10^{11}$	0	$8.56\ 10^{-41}$	$8.56\ 10^{-41}$	$8.65\ 10^{-30}$
Cs-135	$1.55\ 10^{9}$	0	$1.33\ 10^{-56}$	$1.33\ 10^{-56}$	$2.07\ 10^{-47}$
Cs-137	$9.69\ 10^{8}$	0	$4.06\ 10^{-21}$	$4.06\ 10^{-21}$	$3.93\ 10^{-12}$
Ce-144	$4.81\ 10^{12}$	0	$2.03\ 10^{-92}$	$2.03\ 10^{-92}$	$9.78\ 10^{-80}$
Pm-147	$2.14\ 10^{13}$	0	$1.94\ 10^{-64}$	$1.94\ 10^{-64}$	$4.15\ 10^{-51}$
Sm-147	$4.81\ 10^{8}$	0	0	0	0
Sm-151	$7.23\ 10^{11}$	0	0	0	0
Eu-152	$8.05\ 10^{9}$	0	$3.28\ 10^{-21}$	$3.28\ 10^{-21}$	$2.64\ 10^{-11}$
Eu-154	$4.18\ 10^{10}$	0	$5.48\ 10^{-23}$	$5.48\ 10^{-23}$	$2.29\ 10^{-12}$
Eu-155	$8.81\ 10^{12}$	0	$1.48\ 10^{-49}$	$1.48\ 10^{-49}$	$1.30\ 10^{-36}$
Gd-153	$4.83\ 10^{13}$	0	$1.50\ 10^{-110}$	$1.50\ 10^{-110}$	$7.22\ 10^{-97}$
Pb-210	$4.85\ 10^{8}$	0	$8.68\ 10^{-24}$	$8.68\ 10^{-24}$	$4.21\ 10^{-15}$
Po-210	$6.17\ 10^{9}$	0	$5.41\ 10^{-143}$	$5.41\ 10^{-143}$	$3.34\ 10^{-133}$
Ra-226	$3.89\ 10^{6}$	$4.02\ 10^{-5}$	$1.72\ 10^{-16}$	$4.02\ 10^{-5}$	$1.57\ 10^{2}$
Ra-228	$2.25\ 10^{10}$	0	$1.24\ 10^{-22}$	$1.24\ 10^{-22}$	$2.78\ 10^{-12}$
Ac-227	$3.04\ 10^{9}$	0	$2.90\ 10^{-22}$	$2.90\ 10^{-22}$	$8.82\ 10^{-13}$
Th-228	$1.72\ 10^{11}$	0	$1.34\ 10^{-38}$	$1.34\ 10^{-38}$	$2.31\ 10^{-27}$
Th-229	$2.88\ 10^{7}$	0	$1.45\ 10^{-17}$	$1.45\ 10^{-17}$	$4.17\ 10^{-10}$
Th-230	$1.98\ 10^{6}$	0	$1.16\ 10^{-17}$	$1.16\ 10^{-17}$	$2.29\ 10^{-11}$
Th-232	$7.95\ 10^{6}$	0	$3.29\ 10^{-15}$	$3.29\ 10^{-15}$	$2.62\ 10^{-8}$
Pa-231	$1.36\ 10^{7}$	0	$3.39\ 10^{-20}$	$3.39\ 10^{-20}$	$4.61\ 10^{-13}$
U-232	$4.04\ 10^{8}$	0	$1.09\ 10^{-19}$	$1.09\ 10^{-19}$	$4.39\ 10^{-11}$
U-233	$1.02\ 10^{8}$	0	$2.07\ 10^{-19}$	$2.07\ 10^{-19}$	$2.12\ 10^{-11}$

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from radiological capacity
		Gas	External	Total	
U-234	1.45 10^{8}	0	9.85 10^{-41}	9.85 10^{-41}	1.43 10^{-32}
U-235	6.93 10^{7}	0	5.80 10^{-28}	5.80 10^{-28}	4.02 10^{-20}
U-236	1.48 10^{9}	0	2.45 10^{-23}	2.45 10^{-23}	3.62 10^{-14}
U-238	1.60 10^{9}	0	2.71 10^{-23}	2.71 10^{-23}	4.35 10^{-14}
Np-237	1.42 10^{7}	0	4.03 10^{-26}	4.03 10^{-26}	5.71 10^{-19}
Pu-238	7.56 10^{8}	0	2.81 10^{-47}	2.81 10^{-47}	2.12 10^{-38}
Pu-239	1.55 10^{8}	0	5.27 10^{-31}	5.27 10^{-31}	8.18 10^{-23}
Pu-240	1.89 10^{8}	0	2.59 10^{-46}	2.59 10^{-46}	4.90 10^{-38}
Pu-241	9.39 10^{9}	0	1.29 10^{-43}	1.29 10^{-43}	1.21 10^{-33}
Pu-242	1.58 10^{8}	0	6.32 10^{-31}	6.32 10^{-31}	1.00 10^{-22}
Pu-244	1.26 10^{8}	0	6.51 10^{-24}	6.51 10^{-24}	8.19 10^{-16}
Am-241	3.03 10^{8}	0	1.74 10^{-30}	1.74 10^{-30}	5.26 10^{-22}
Am-242m	1.75 10^{7}	0	8.14 10^{-22}	8.14 10^{-22}	1.42 10^{-14}
Am-243	1.46 10^{8}	0	2.06 10^{-29}	2.06 10^{-29}	3.00 10^{-21}
Cm-242	1.48 10^{11}	0	1.34 10^{-49}	1.34 10^{-49}	1.98 10^{-38}
Cm-243	4.89 10^{7}	0	2.18 10^{-30}	2.18 10^{-30}	1.07 10^{-22}
Cm-244	1.16 10^{8}	0	1.77 10^{-56}	1.77 10^{-56}	2.06 10^{-48}
Cm-245	1.26 10^{7}	0	4.42 10^{-34}	4.42 10^{-34}	5.56 10^{-27}
Cm-246	1.27 10^{7}	0	6.41 10^{-69}	6.41 10^{-69}	8.13 10^{-62}
Cm-248	1.45 10^{7}	0	8.46 10^{-30}	8.46 10^{-30}	1.23 10^{-22}
Ra-226$	3.89 10^{6}	6.71 10^{-11}	2.80 10^{-30}	6.71 10^{-11}	2.61 10^{-4}

Note: $ Ra-226 buried 5m or greater below the restored site level

Table 145 Infant Site Resident Exposure

Radionuclide	Radiological capacity (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from radiological capacity
		Gas	External	Total	
H-3	6.43 10^{9}	1.67 10^{-10}	0	1.67 10^{-10}	1.07 10^{0}
C-14	1.87 10^{8}	1.65 10^{-6}	1.54 10^{-66}	1.65 10^{-6}	3.09 10^{2}
Cl-36	1.56 10^{8}	0	8.03 10^{-29}	8.03 10^{-29}	1.26 10^{-20}
Ca-41	5.77 10^{9}	0	0	0	0
Mn-54	1.12 10^{13}	0	1.68 10^{-71}	1.68 10^{-71}	1.89 10^{-58}

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose ($\mu Sv\ y^{-1}$) from radiological capacity
		Gas	External	Total	
Fe-55	$1.86\ 10^{13}$	0	0	0	0
Co-60	$3.58\ 10^{11}$	0	$3.46\ 10^{-25}$	$3.46\ 10^{-25}$	$1.24\ 10^{-13}$
Ni-59	$1.95\ 10^{11}$	0	0	0	0
Ni-63	$2.42\ 10^{11}$	0	0	0	0
Zn-65	$8.95\ 10^{11}$	0	$1.97\ 10^{-85}$	$1.97\ 10^{-85}$	$1.76\ 10^{-73}$
Se-79	$8.98\ 10^{8}$	0	$1.79\ 10^{-64}$	$1.79\ 10^{-64}$	$1.61\ 10^{-55}$
Sr-90	$3.83\ 10^{8}$	0	$1.84\ 10^{-27}$	$1.84\ 10^{-27}$	$7.04\ 10^{-19}$
Mo-93	$1.44\ 10^{9}$	0	$4.62\ 10^{-9}$	$4.62\ 10^{-9}$	$6.65\ 10^{0}$
Zr-93	$3.12\ 10^{11}$	0	0	0	0
Nb-93m	$5.06\ 10^{10}$	0	$1.33\ 10^{-12}$	$1.33\ 10^{-12}$	$6.71\ 10^{-2}$
Nb-94	$6.09\ 10^{6}$	0	$1.17\ 10^{-18}$	$1.17\ 10^{-18}$	$7.10\ 10^{-12}$
Tc-99	$6.12\ 10^{8}$	0	$3.47\ 10^{-48}$	$3.47\ 10^{-48}$	$2.13\ 10^{-39}$
Ru-106	$9.14\ 10^{11}$	0	$1.50\ 10^{-64}$	$1.50\ 10^{-64}$	$1.37\ 10^{-52}$
Ag-108m	$2.65\ 10^{8}$	0	$7.46\ 10^{-20}$	$7.46\ 10^{-20}$	$1.98\ 10^{-11}$
Ag-110m	$6.41\ 10^{12}$	0	$7.80\ 10^{-84}$	$7.80\ 10^{-84}$	$5.00\ 10^{-71}$
Cd-109	$1.04\ 10^{12}$	0	$1.28\ 10^{-84}$	$1.28\ 10^{-84}$	$1.33\ 10^{-72}$
Sb-125	$4.17\ 10^{11}$	0	$3.50\ 10^{-37}$	$3.50\ 10^{-37}$	$1.46\ 10^{-25}$
Sn-119m	$8.43\ 10^{12}$	0	0	0	0
Sn-123	$2.97\ 10^{12}$	0	$4.98\ 10^{-148}$	$4.98\ 10^{-148}$	$1.48\ 10^{-135}$
Sn-126	$4.60\ 10^{6}$	0	$2.67\ 10^{-19}$	$2.67\ 10^{-19}$	$1.23\ 10^{-12}$
Te-127m	$4.07\ 10^{12}$	0	$5.13\ 10^{-194}$	$5.13\ 10^{-194}$	$2.09\ 10^{-181}$
I-129	$3.01\ 10^{8}$	0	$2.20\ 10^{-138}$	$2.20\ 10^{-138}$	$6.63\ 10^{-130}$
Ba-133	$7.18\ 10^{9}$	0	$1.00\ 10^{-27}$	$1.00\ 10^{-27}$	$7.19\ 10^{-18}$
Cs-134	$1.01\ 10^{11}$	0	$6.35\ 10^{-41}$	$6.35\ 10^{-41}$	$6.42\ 10^{-30}$
Cs-135	$1.55\ 10^{9}$	0	$9.88\ 10^{-57}$	$9.88\ 10^{-57}$	$1.53\ 10^{-47}$
Cs-137	$9.69\ 10^{8}$	0	$3.01\ 10^{-21}$	$3.01\ 10^{-21}$	$2.92\ 10^{-12}$
Ce-144	$4.81\ 10^{12}$	0	$1.51\ 10^{-92}$	$1.51\ 10^{-92}$	$7.26\ 10^{-80}$
Pm-147	$2.14\ 10^{13}$	0	$1.44\ 10^{-64}$	$1.44\ 10^{-64}$	$3.08\ 10^{-51}$
Sm-147	$4.81\ 10^{8}$	0	0	0	0
Sm-151	$7.23\ 10^{11}$	0	0	0	0
Eu-152	$8.05\ 10^{9}$	0	$2.43\ 10^{-21}$	$2.43\ 10^{-21}$	$1.96\ 10^{-11}$
Eu-154	$4.18\ 10^{10}$	0	$4.07\ 10^{-23}$	$4.07\ 10^{-23}$	$1.70\ 10^{-12}$
Eu-155	$8.81\ 10^{12}$	0	$1.10\ 10^{-49}$	$1.10\ 10^{-49}$	$9.66\ 10^{-37}$
Gd-153	$4.83\ 10^{13}$	0	$1.11\ 10^{-110}$	$1.11\ 10^{-110}$	$5.36\ 10^{-97}$
Pb-210	$4.85\ 10^{8}$	0	$6.45\ 10^{-24}$	$6.45\ 10^{-24}$	$3.13\ 10^{-15}$

Radionuclide	Radiological capacity (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose ($\mu Sv\ y^{-1}$) from radiological capacity
		Gas	External	Total	
Po-210	$6.17\ 10^{9}$	0	$4.01\ 10^{-143}$	$4.01\ 10^{-143}$	$2.48\ 10^{-133}$
Ra-226	$3.89\ 10^{6}$	$5.62\ 10^{-5}$	$1.28\ 10^{-16}$	$5.62\ 10^{-5}$	$2.19\ 10^{2}$
Ra-228	$2.25\ 10^{10}$	0	$9.18\ 10^{-23}$	$9.18\ 10^{-23}$	$2.07\ 10^{-12}$
Ac-227	$3.04\ 10^{9}$	0	$2.15\ 10^{-22}$	$2.15\ 10^{-22}$	$6.55\ 10^{-13}$
Th-228	$1.72\ 10^{11}$	0	$9.98\ 10^{-39}$	$9.98\ 10^{-39}$	$1.71\ 10^{-27}$
Th-229	$2.88\ 10^{7}$	0	$1.08\ 10^{-17}$	$1.08\ 10^{-17}$	$3.10\ 10^{-10}$
Th-230	$1.98\ 10^{6}$	0	$8.59\ 10^{-18}$	$8.59\ 10^{-18}$	$1.70\ 10^{-11}$
Th-232	$7.95\ 10^{6}$	0	$2.44\ 10^{-15}$	$2.44\ 10^{-15}$	$1.94\ 10^{-8}$
Pa-231	$1.36\ 10^{7}$	0	$2.52\ 10^{-20}$	$2.52\ 10^{-20}$	$3.42\ 10^{-13}$
U-232	$4.04\ 10^{8}$	0	$8.07\ 10^{-20}$	$8.07\ 10^{-20}$	$3.26\ 10^{-11}$
U-233	$1.02\ 10^{8}$	0	$1.54\ 10^{-19}$	$1.54\ 10^{-19}$	$1.57\ 10^{-11}$
U-234	$1.45\ 10^{8}$	0	$7.31\ 10^{-41}$	$7.31\ 10^{-41}$	$1.06\ 10^{-32}$
U-235	$6.93\ 10^{7}$	0	$4.31\ 10^{-28}$	$4.31\ 10^{-28}$	$2.98\ 10^{-20}$
U-236	$1.48\ 10^{9}$	0	$1.81\ 10^{-23}$	$1.81\ 10^{-23}$	$2.69\ 10^{-14}$
U-238	$1.60\ 10^{9}$	0	$2.01\ 10^{-23}$	$2.01\ 10^{-23}$	$3.23\ 10^{-14}$
Np-237	$1.42\ 10^{7}$	0	$2.99\ 10^{-26}$	$2.99\ 10^{-26}$	$4.24\ 10^{-19}$
Pu-238	$7.56\ 10^{8}$	0	$2.08\ 10^{-47}$	$2.08\ 10^{-47}$	$1.58\ 10^{-38}$
Pu-239	$1.55\ 10^{8}$	0	$3.91\ 10^{-31}$	$3.91\ 10^{-31}$	$6.07\ 10^{-23}$
Pu-240	$1.89\ 10^{8}$	0	$1.92\ 10^{-46}$	$1.92\ 10^{-46}$	$3.64\ 10^{-38}$
Pu-241	$9.39\ 10^{9}$	0	$9.57\ 10^{-44}$	$9.57\ 10^{-44}$	$8.99\ 10^{-34}$
Pu-242	$1.58\ 10^{8}$	0	$4.69\ 10^{-31}$	$4.69\ 10^{-31}$	$7.43\ 10^{-23}$
Pu-244	$1.26\ 10^{8}$	0	$4.83\ 10^{-24}$	$4.83\ 10^{-24}$	$6.08\ 10^{-16}$
Am-241	$3.03\ 10^{8}$	0	$1.29\ 10^{-30}$	$1.29\ 10^{-30}$	$3.91\ 10^{-22}$
Am-242m	$1.75\ 10^{7}$	0	$6.04\ 10^{-22}$	$6.04\ 10^{-22}$	$1.06\ 10^{-14}$
Am-243	$1.46\ 10^{8}$	0	$1.53\ 10^{-29}$	$1.53\ 10^{-29}$	$2.23\ 10^{-21}$
Cm-242	$1.48\ 10^{11}$	0	$9.94\ 10^{-50}$	$9.94\ 10^{-50}$	$1.47\ 10^{-38}$
Cm-243	$4.89\ 10^{7}$	0	$1.62\ 10^{-30}$	$1.62\ 10^{-30}$	$7.90\ 10^{-23}$
Cm-244	$1.16\ 10^{8}$	0	$1.31\ 10^{-56}$	$1.31\ 10^{-56}$	$1.53\ 10^{-48}$
Cm-245	$1.26\ 10^{7}$	0	$3.28\ 10^{-34}$	$3.28\ 10^{-34}$	$4.12\ 10^{-27}$
Cm-246	$1.27\ 10^{7}$	0	$4.76\ 10^{-69}$	$4.76\ 10^{-69}$	$6.03\ 10^{-62}$
Cm-248	$1.45\ 10^{7}$	0	$6.28\ 10^{-30}$	$6.28\ 10^{-30}$	$9.12\ 10^{-23}$
Ra-226$	$3.89\ 10^{6}$	$9.36\ 10^{-11}$	$2.08\ 10^{-30}$	$9.36\ 10^{-11}$	$3.64\ 10^{-4}$

Note: $ Ra-226 buried 5m or greater below the restored site level

E.5.6. Excavation for housing – Residential Occupant

978. Construction activities for housing developments would include shallow excavations and cap disturbance to prepare the site and install roads and services. Foundations for domestic and light buildings, typically 1 or 2 m deep, may penetrate the 1.3 m thick capping layer but will not reach the LLW since it is not placed within the top 1 m of the cell. At sites where the load bearing capacity of underlying ground is low, such as made-ground, land in-fill or soft clay, foundations are likely to be cast as a raft (thick concrete slab with steel reinforcement).

979. In this assessment we assume that the ground has sufficient load bearing capacity for conventional foundations and that construction that might intersect waste at depths greater than 1-2 m below the surface does occur, for example excavation for cellars, an underground car park or underground tanks (for petrol or farm slurry). Excavated material could be used as backfill and in landscaping. Those involved in the excavation would be exposed to the hazard and, in the long term, site occupants could be exposed to contaminated materials that remain in the surface environment.

980. Contaminated material may be left at the surface, although it is more likely that such materials would be disposed of given the nature of material in the landfill. The non-radioactive waste disposed of at Port Clarence largely comprises treated residues (grey coloured) and asbestos. This material is not biodegradable and will essentially remain the same over geological timescales. This would discourage extensive excavation and it is therefore unlikely that contaminated soil will be left on the surface of the site.

981. The radioactive and non-radioactive waste includes numerous other materials some of which are unlikely to degrade with time and this would discourage extensive excavation. It is therefore unlikely that extensive excavation will take place and highly unlikely contaminated soil will be left on the surface of the site. Furthermore, the ability of such material to support plant growth is inconceivable without significant dilution of the waste by clean soil.

982. Occupancy of a smallholding is addressed in Section E.5.8.

E.5.6.1. Estimating activity concentration in waste for exposure calculations

Dilution factors

983. The excavated spoil will include a mixture of radioactive waste, hazardous waste, soil and cover material, resulting in ‘dilution’ of the radioactive waste with other material. Characteristics that have been used to determine the dilution factor applied to radioactive waste in excavated spoil in other studies include:

- Depth and area of landfill displaced (volume excavated);
- Capping layer depth and waste emplacement cover depth (depth to contaminated waste);
- Proportion of radioactive waste in the landfilled materials;
- Mixing with clean soil is described in different ways:

- loading of clean soil with excavated spoil;
- depth of waste spread on a given land area;
- depth of clean soil cover or depth of mixing with clean soil; and,

- Fraction of inhabited/utilised area that is contaminated.

984. The term “dilution factor” is not applied consistently in the studies reviewed and may have incorporated one or more of the factors listed above. It can be used to determine a spoil activity concentration based on the following equation:

$$C_{spoil} = \frac{INV_y}{V_L \cdot \rho_{waste}} \cdot DIL$$

where:

- C_{spoil} is the spoil activity concentration (Bq kg^{-1});
- INV_y is the inventory in the landfill in year y (Bq);
- V_L is the landfill volume (m^3);
- P_{waste} is the waste density (kg m^{-3}); and,
- DIL is the dilution factor.

985. The type of construction will determine the depth and area of displaced material. We have assumed the excavation will be 5 m deep (Hicks & Baldwin, 2011), producing a mixed spoil comprising 1.3 m capping materials, 1.0 m cover and 2.7 m waste. The mixed spoil therefore comprises 54% waste. It is assumed for the ESC that radioactive waste input to the landfill will be on average limited to approximately 20% of total inputs to Port Clarence, the rest comprising other wastes and emplacement cover material.

986. A factor of 0.2 is therefore used for larger excavations (a housing development or small holding) where an average composition is more likely to be displaced and hence the excavated spoil is assumed to contain 10.8% radioactive waste. For relatively small excavations it is conceivable that the displaced waste material will comprise only radioactive waste and this was covered in the assessment of doses to the trial pit excavation worker.

987. It is clear that clean soil will need to be mixed with the excavated spoil in order to provide a growing medium that will sustain plant growth. A review of dilution factors used in other ESCs is given in ENRMF ESC (Eden NE, 2015a). A growing medium was therefore assumed to contain a maximum of 10% spoil.

988. This assessment considers two potentially exposed groups:

- A smallholder (200 years after closure) living over the site who requires 1 to 3 hectares of land and produces meat, milk and a mixture of crops.
- A housing development (150 years after closure) with residents growing their own vegetables.

989. The assumptions for the excavations for a smallholding and for a housing estate are taken from the ENRMF ESC (Eden NE, 2015a):

- For the smallholder, excavations to 5 m (100 m^2) have removed 500 m^3 of spoil for a new slurry tank. It is assumed that excavated waste contains 20% radioactive material and, following mixing with clean soil (at a rate of 10% spoil), the diluted spoil would be spread over an area of 1.6 ha which supports food production as detailed below. Combining the spoil dilution (1.3 m capping layer, 1.0 m cover, 2.7 m waste) during excavation, site average radioactive waste content and mixing with clean soil (0.1), an overall dilution factor of 0.0108 is applied (DIL). This is conservative as it does not use assumptions concerning a patchy distribution/partially contaminated area. It is assumed that excavated waste is spread directly under the house and in this case the dilution factor omits the clean soil factor (DIL = 0.108).
- For the housing development, the excavated area is 400 m^2, removing 2000 m^3 of spoil. It is assumed that the excavated waste contains 20% radioactive material (site average) and is mixed with clean soil (at a rate of 10% spoil) for the garden. Combining the spoil dilution, site average radioactive waste content and mixing with clean soil, an overall dilution factor of 0.0108 is applied (DIL). This is conservative as it does not use assumptions concerning a patchy distribution/partially contaminated area. It is assumed that excavated waste is spread directly under the house and in this case the dilution factor omits the clean soil factor (DIL = 0.108).

990. In both cases, it is assumed that up to 1 m of the cap is removed in order to level the site for the house.

991. A factor, limiting the area assumed to be contaminated to a fraction of that available, has not been applied in this assessment. This is an uncertain factor and could have a far greater impact than any of the factors applied above, in particular where land is used either for a smallholding or is farmed commercially. Available assessments and example calculations have used factors as low as 1.0 10^{-4}.

992. The area of land assumed to be used for the smallholding (1.6ha) is based on the crop yields in SNIFFER, critical group consumption rates (NDAWG, 2013) and sufficient crops to feed 3 adults (adult ingestion rates are greater than child and infant). The land also supports 2 cows using 0.57 forage ha, and 2 followers (at a rate of 1 ha for every 3 ha to cows) (Nix, 2010). On this basis the pasture required amounts to about 1.5 ha with a further 0.1 ha for growing crops.

993. The long-term occupants of the housing estate are an adult, child and infant living at a residential site built on top of Port Clarence facility. While it is reasonable for a residential occupant to grow some crops (assumed to be green vegetables and root vegetables) in a garden or allotment, it is assumed for the purposes of this assessment that they will not keep livestock or cultivate grain.

Activity concentration in soil

994. Following excavation, radioactively contaminated waste and the covering layer are mixed, forming a partially-contaminated soil layer. The activity concentration of radionuclide *Rn* in the soil, $C_{Rn,soil,excavate}$ (Bq kg^{-1}) after the excavation event is given by:

$$C_{Rn,soil,excavate} = \frac{A_{Rn}(t)\,.\,Dil}{V_{landfill} \cdot \rho_{landfill}}$$

995. Where Dil is a dilution factor given by the ratio of the volume of contaminated landfill waste to the volume of other material that is mixed in to form the soil multiplied by any further mixing with uncontaminated surface soil. A value of 0.0108 is used for LLW in the garden as discussed above (see paragraph 989) and a factor of 0.108 for exposure inside the house.

E.5.6.2. Assessment calculations for Residential Occupant

996. Doses can result from:

- ingestion of foodstuff grown on contaminated soil;
- ingestion or inhalation of dust from the soil; and,
- external irradiation from contaminated soil.

Ingestion of crops

997. Dose from ingesting crops grown on contaminated soil is given by (Augean, 2009):

$$Dose_{ing,crops} = \sum_{crop} \{Q_{crop} \cdot [C_{Rn,soil}(t) \cdot UF_{Rn,crop}]\} \cdot D_{Rn,ing}$$

where:

- Q_{crop} is the crop consumption rate (kg y-1);
- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide Rn at time t (Bq kg^{-1});
- $UF_{Rn,crop}$ is the soil to crop transfer factors for radionuclide Rn (Bq kg^{-1} fresh weight of crop per Bq kg^{-1} of soil); and,
- $D_{Rn,ing}$ is the dose coefficient for ingestion of radionuclide Rn (Sv Bq^{-1}).

998. Parameter values are summarised in Table 146, dose coefficients for ingestion are given in Table 200 and soil to crop transfer factors are given in Table 203.

Table 146 Parameters used in the long-term occupant scenario*

Parameter	Substance	Units	Value
Consumption rate (adult)	Green vegetables**	$kg\ y^{-1}$	17.5 / 40
	Root vegetables**	$kg\ y^{-1}$	30 / 65
	Soil	$kg\ y^{-1}$	0.03
Consumption rate (child)	Green vegetables**	$kg\ y^{-1}$	7.5 / 17.5
	Root vegetables**	$kg\ y^{-1}$	25 / 42.5
	Soil	$kg\ y^{-1}$	0.018
Consumption rate (infant)	Green vegetables**	$kg\ y^{-1}$	2.5 / 7.5
	Root vegetables**	$kg\ y^{-1}$	7.5 / 22.5
	Soil	$kg\ y^{-1}$	0.044
Occupancy Indoors - adult		$y\ y^{-1}$	0.80
Occupancy outdoors** - adult		$y\ y^{-1}$	0.20
Occupancy Indoors – child		$y\ y^{-1}$	0.84
Occupancy outdoors** - child		$y\ y^{-1}$	0.16
Occupancy Indoors – infant		$y\ y^{-1}$	0.91
Occupancy outdoors** - infant		$y\ y^{-1}$	0.09
Shielding factor indoors**			0.1
Dustload		$kg\ m^{-3}$	$1\ 10^{-7}$
Breathing rate adult		$m^3\ h^{-1}$	1
Breathing rate child		$m^3\ h^{-1}$	0.64
Breathing rate infant		$m^3\ h^{-1}$	0.22
Dilution factor	Soil in garden		0.0108
Dilution factor	Soil under house		0.108

*Values from (Augean, 2009), unless otherwise stated
**Taken from NRPB/HPA W36 (Oatway & Mobbs, 2003), average / 97.5th percentiles

External irradiation

999. Dose from external irradiation while living and working on contaminated soil is given by (Augean, 2009):

$$Dose_{irr,soil} = \left(Clean\ \cdot O_{out,j} + O_{in,j} \cdot SF\right) \cdot C_{Rn,soil}(t) \cdot D_{Rn,irr,slab}$$

where:

- $O_{out,j}$ is the fraction of time spent outside by age group j, exposed to contaminated soil ($y\ y^{-1}$);
- $O_{in,j}$ is the fraction of time spent inside by age group j ($y\ y^{-1}$);
- $Clean$ is the dilution with clean soil in garden;
- SF is the shielding factor from the ground while indoors;
- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide *Rn* at time *t* ($Bq\ kg^{-1}$) in spoil; and,

- $D_{Rn,irr,slab}$ is the dose conversion factor for irradiation from radionuclide *Rn* (Sv y^{-1} Bq^{-1} kg), based on the receptor being 1 m from the ground and assuming a semi-infinite slab of contamination.

1000. Parameter values are summarised in Table 146. Note that the foodstuff consumption rates taken from NRPB/HPA W36 (Oatway & Mobbs, 2003) correspond to a residential occupant who cultivates a quantity of root and green vegetables that supplements, but does not form the bulk, of their vegetable intake. A higher rate of consumption would be more appropriate to a smallholder or subsistence cultivator of crops.

1001. Dose conversion factors for irradiation are given in Table 201.

Inhalation of contaminated soil

1002. Dose from inhalation of contaminated soil is given by (Augean, 2009):

$$Dose_{inh,soil} = B \cdot O_{dust} \cdot C_{Rn,soil}(t) \cdot Dustload \cdot D_{Rn,inh}$$

where:

- B is the breathing rate (m^3 y^{-1});
- O_{dust} is the fraction of time spent exposed to dust from the soil (y y^{-1});
- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide *Rn* at time *t* (Bq kg^{-1});
- $Dustload$ is the dust concentration in air (kg m^{-3}); and,
- $D_{Rn,inh}$ is the dose coefficient for inhalation of radionuclide *Rn* (Sv Bq^{-1}).

Parameter values are summarised in Table 146 and dose coefficients for inhalation are given in Table 200.

Inhalation of gases

1003. The assessment calculations presented for the residential housing scenario include a contribution based on gas migration from underlying waste (see Section E.5.5) and in the case of radon from excavated waste remaining directly under the house. The average timescales for release of gas for H-3 and C-14 used were 50y and 600y, respectively.

1004. The radon model for spoil uses the original model from which the version in SNIFFER is derived (see Section E.5.8.2). The soil depth is assumed to be 0.10 m for the resident.

E.5.6.3. Dose to Residential Occupant on site after 150 years

1005. In Table 147 the dose rates to adult, child and infant residents respectively on the site following construction of houses 150 years after site capping are presented. The largest contributions to dose arise from Ra-226 with no emplacement strategy for wastes containing > 5 Bq g^{-1}, Se-79, Tc-99, Ag-110m and I-129. The impact of Radium placement depth within Port Clarence on radon release is discussed in paragraph 1013.

Table 147 Doses to site residents after 150 y

Radionuclide	Radiological capacity (MBq)	Dose to adult ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Adult dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to child ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to infant ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose from radiological capacity ($\mu Sv\ y^{-1}$)
H-3	$6.43\ 10^{9}$	$1.66\ 10^{-10}$	$1.07\ 10^{0}$	$1.72\ 10^{-10}$	$1.11\ 10^{0}$	$1.69\ 10^{-10}$	$1.09\ 10^{0}$
C-14	$1.87\ 10^{8}$	$2.12\ 10^{-6}$	$3.97\ 10^{2}$	$1.94\ 10^{-6}$	$3.63\ 10^{2}$	$1.66\ 10^{-6}$	$3.10\ 10^{2}$
Cl-36	$1.56\ 10^{8}$	$8.29\ 10^{-7}$	$1.30\ 10^{2}$	$9.67\ 10^{-7}$	$1.51\ 10^{2}$	$1.60\ 10^{-6}$	$2.51\ 10^{2}$
Ca-41	$5.77\ 10^{9}$	$1.69\ 10^{-8}$	$9.76\ 10^{1}$	$2.44\ 10^{-8}$	$1.41\ 10^{2}$	$1.33\ 10^{-8}$	$7.65\ 10^{1}$
Mn-54	$1.12\ 10^{13}$	$1.12\ 10^{-59}$	$1.26\ 10^{-46}$	$9.96\ 10^{-60}$	$1.12\ 10^{-46}$	$7.84\ 10^{-60}$	$8.78\ 10^{-47}$
Fe-55	$1.86\ 10^{13}$	$1.03\ 10^{-27}$	$1.92\ 10^{-14}$	$2.04\ 10^{-27}$	$3.78\ 10^{-14}$	$6.77\ 10^{-27}$	$1.26\ 10^{-13}$
Co-60	$3.58\ 10^{11}$	$5.96\ 10^{-15}$	$2.13\ 10^{-3}$	$5.23\ 10^{-15}$	$1.87\ 10^{-3}$	$3.93\ 10^{-15}$	$1.41\ 10^{-3}$
Ni-59	$1.95\ 10^{11}$	$3.40\ 10^{-10}$	$6.63\ 10^{1}$	$3.39\ 10^{-10}$	$6.62\ 10^{1}$	$5.44\ 10^{-10}$	$1.06\ 10^{2}$
Ni-63	$2.42\ 10^{11}$	$2.87\ 10^{-10}$	$6.93\ 10^{1}$	$3.06\ 10^{-10}$	$7.40\ 10^{1}$	$4.76\ 10^{-10}$	$1.15\ 10^{2}$
Zn-65	$8.95\ 10^{11}$	$3.38\ 10^{-74}$	$3.02\ 10^{-62}$	$2.62\ 10^{-74}$	$2.34\ 10^{-62}$	$2.47\ 10^{-74}$	$2.21\ 10^{-62}$
Se-79	$8.98\ 10^{8}$	$5.17\ 10^{-7}$	$4.64\ 10^{2}$	$1.43\ 10^{-6}$	$1.28\ 10^{3}$	$1.43\ 10^{-6}$	$1.28\ 10^{3}$
Sr-90	$3.83\ 10^{8}$	$1.77\ 10^{-7}$	$6.79\ 10^{1}$	$1.70\ 10^{-7}$	$6.52\ 10^{1}$	$1.05\ 10^{-7}$	$4.01\ 10^{1}$
Mo-93	$1.44\ 10^{9}$	$2.11\ 10^{-7}$	$3.04\ 10^{2}$	$1.49\ 10^{-7}$	$2.15\ 10^{2}$	$1.26\ 10^{-7}$	$1.82\ 10^{2}$
Zr-93	$3.12\ 10^{11}$	$1.05\ 10^{-9}$	$3.26\ 10^{2}$	$3.14\ 10^{-10}$	$9.80\ 10^{1}$	$2.50\ 10^{-10}$	$7.80\ 10^{1}$
Nb-93m	$5.06\ 10^{10}$	$3.72\ 10^{-13}$	$1.88\ 10^{-2}$	$4.69\ 10^{-13}$	$2.37\ 10^{-2}$	$8.58\ 10^{-13}$	$4.34\ 10^{-2}$
Nb-94	$6.09\ 10^{6}$	$1.30\ 10^{-6}$	$7.89\ 10^{0}$	$1.13\ 10^{-6}$	$6.88\ 10^{0}$	$8.41\ 10^{-7}$	$5.12\ 10^{0}$
Tc-99	$6.12\ 10^{8}$	$1.14\ 10^{-6}$	$6.97\ 10^{2}$	$1.32\ 10^{-6}$	$8.09\ 10^{2}$	$2.44\ 10^{-6}$	$1.49\ 10^{3}$
Ru-106	$9.14\ 10^{11}$	$1.33\ 10^{-51}$	$1.22\ 10^{-39}$	$1.19\ 10^{-51}$	$1.09\ 10^{-39}$	$9.93\ 10^{-52}$	$9.08\ 10^{-40}$
Ag-108m	$2.65\ 10^{8}$	$1.01\ 10^{-6}$	$2.68\ 10^{2}$	$8.79\ 10^{-7}$	$2.33\ 10^{2}$	$6.53\ 10^{-7}$	$1.73\ 10^{2}$
Ag-110m	$6.41\ 10^{12}$	$2.22\ 10^{-72}$	$1.43\ 10^{-59}$	$1.94\ 10^{-72}$	$1.24\ 10^{-59}$	$1.44\ 10^{-72}$	$9.22\ 10^{-60}$
Cd-109	$1.04\ 10^{12}$	$7.22\ 10^{-43}$	$7.51\ 10^{-31}$	$7.75\ 10^{-43}$	$8.06\ 10^{-31}$	$1.08\ 10^{-42}$	$1.13\ 10^{-30}$
Sb-125	$4.17\ 10^{11}$	$1.43\ 10^{-23}$	$5.95\ 10^{-12}$	$1.24\ 10^{-23}$	$5.19\ 10^{-12}$	$9.34\ 10^{-24}$	$3.90\ 10^{-12}$

Radionuclide	Radiological capacity (MBq)	Dose to adult ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Adult dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to child ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to infant ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose from radiological capacity ($\mu Sv\ y^{-1}$)
Sn-119m	$8.43\ 10^{12}$	$3.48\ 10^{-65}$	$2.94\ 10^{-52}$	$4.30\ 10^{-65}$	$3.63\ 10^{-52}$	$7.08\ 10^{-65}$	$5.97\ 10^{-52}$
Sn-123	$2.97\ 10^{12}$	$9.87\ 10^{-136}$	$2.93\ 10^{-123}$	$1.18\ 10^{-135}$	$3.50\ 10^{-123}$	$1.94\ 10^{-135}$	$5.76\ 10^{-123}$
Sn-126	$4.60\ 10^{6}$	$1.68\ 10^{-6}$	$7.73\ 10^{0}$	$1.49\ 10^{-6}$	$6.86\ 10^{0}$	$1.19\ 10^{-6}$	$5.49\ 10^{0}$
Te-127m	$4.07\ 10^{12}$	$5.01\ 10^{-165}$	$2.04\ 10^{-152}$	$5.06\ 10^{-165}$	$2.06\ 10^{-152}$	$1.01\ 10^{-164}$	$4.09\ 10^{-152}$
I-129	$3.01\ 10^{8}$	$1.97\ 10^{-6}$	$5.92\ 10^{2}$	$1.94\ 10^{-6}$	$5.84\ 10^{2}$	$1.14\ 10^{-6}$	$3.42\ 10^{2}$
Ba-133	$7.18\ 10^{9}$	$1.36\ 10^{-11}$	$9.79\ 10^{-2}$	$1.19\ 10^{-11}$	$8.57\ 10^{-2}$	$8.87\ 10^{-12}$	$6.37\ 10^{-2}$
Cs-134	$1.01\ 10^{11}$	$1.87\ 10^{-28}$	$1.89\ 10^{-17}$	$1.56\ 10^{-28}$	$1.58\ 10^{-17}$	$1.15\ 10^{-28}$	$1.16\ 10^{-17}$
Cs-135	$1.55\ 10^{9}$	$1.08\ 10^{-8}$	$1.68\ 10^{1}$	$5.25\ 10^{-9}$	$8.15\ 10^{0}$	$3.69\ 10^{-9}$	$5.72\ 10^{0}$
Cs-137	$9.69\ 10^{8}$	$1.68\ 10^{-8}$	$1.63\ 10^{1}$	$1.37\ 10^{-8}$	$1.32\ 10^{1}$	$1.00\ 10^{-8}$	$9.71\ 10^{0}$
Ce-144	$4.81\ 10^{12}$	$1.38\ 10^{-66}$	$6.62\ 10^{-54}$	$1.24\ 10^{-66}$	$5.95\ 10^{-54}$	$1.37\ 10^{-66}$	$6.57\ 10^{-54}$
Pm-147	$2.14\ 10^{13}$	$9.80\ 10^{-28}$	$2.10\ 10^{-14}$	$1.21\ 10^{-27}$	$2.59\ 10^{-14}$	$2.66\ 10^{-27}$	$5.70\ 10^{-14}$
Sm-147	$4.81\ 10^{8}$	$2.35\ 10^{-8}$	$1.13\ 10^{1}$	$1.67\ 10^{-8}$	$8.01\ 10^{0}$	$2.54\ 10^{-8}$	$1.22\ 10^{1}$
Sm-151	$7.23\ 10^{11}$	$1.31\ 10^{-11}$	$9.46\ 10^{0}$	$1.50\ 10^{-11}$	$1.08\ 10^{1}$	$3.57\ 10^{-11}$	$2.58\ 10^{1}$
Eu-152	$8.05\ 10^{9}$	$4.35\ 10^{-10}$	$3.51\ 10^{0}$	$3.79\ 10^{-10}$	$3.05\ 10^{0}$	$2.82\ 10^{-10}$	$2.27\ 10^{0}$
Eu-154	$4.18\ 10^{10}$	$5.74\ 10^{-12}$	$2.40\ 10^{-1}$	$5.00\ 10^{-12}$	$2.09\ 10^{-1}$	$3.72\ 10^{-12}$	$1.56\ 10^{-1}$
Eu-155	$8.81\ 10^{12}$	$8.08\ 10^{-18}$	$7.12\ 10^{-5}$	$7.06\ 10^{-18}$	$6.22\ 10^{-5}$	$5.35\ 10^{-18}$	$4.71\ 10^{-5}$
Gd-153	$4.83\ 10^{13}$	$8.17\ 10^{-77}$	$3.94\ 10^{-63}$	$7.12\ 10^{-77}$	$3.43\ 10^{-63}$	$5.32\ 10^{-77}$	$2.57\ 10^{-63}$
Pb-210	$4.85\ 10^{8}$	$3.21\ 10^{-8}$	$1.56\ 10^{1}$	$4.37\ 10^{-8}$	$2.12\ 10^{1}$	$6.70\ 10^{-8}$	$3.25\ 10^{1}$
Po-210	$6.17\ 10^{9}$	$7.09\ 10^{-127}$	$4.38\ 10^{-117}$	$8.96\ 10^{-127}$	$5.53\ 10^{-117}$	$5.02\ 10^{-126}$	$3.10\ 10^{-116}$
Ra-226$	$3.89\ 10^{6}$	$4.41\ 10^{-5}$	$1.71\ 10^{2}$	$6.19\ 10^{-5}$	$2.41\ 10^{2}$	$8.35\ 10^{-5}$	$3.25\ 10^{2}$
Ra-228	$2.25\ 10^{10}$	$1.15\ 10^{-13}$	$2.58\ 10^{-3}$	$2.75\ 10^{-13}$	$6.20\ 10^{-3}$	$2.21\ 10^{-13}$	$4.97\ 10^{-3}$
Ac-227	$3.04\ 10^{9}$	$5.98\ 10^{-9}$	$1.82\ 10^{1}$	$4.78\ 10^{-9}$	$1.45\ 10^{1}$	$6.15\ 10^{-9}$	$1.87\ 10^{1}$
Th-228	$1.72\ 10^{11}$	$3.40\ 10^{-30}$	$5.84\ 10^{-19}$	$2.99\ 10^{-30}$	$5.13\ 10^{-19}$	$2.42\ 10^{-30}$	$4.15\ 10^{-19}$
Th-229	$2.88\ 10^{7}$	$3.89\ 10^{-7}$	$1.12\ 10^{1}$	$3.24\ 10^{-7}$	$9.32\ 10^{0}$	$3.87\ 10^{-7}$	$1.11\ 10^{1}$
Th-230	$1.98\ 10^{6}$	$1.14\ 10^{-6}$	$2.26\ 10^{0}$	$1.49\ 10^{-6}$	$2.94\ 10^{0}$	$1.88\ 10^{-6}$	$3.72\ 10^{0}$
Th-232	$7.95\ 10^{6}$	$1.02\ 10^{-6}$	$8.13\ 10^{0}$	$1.11\ 10^{-6}$	$8.85\ 10^{0}$	$1.26\ 10^{-6}$	$1.00\ 10^{1}$
Pa-231	$1.36\ 10^{7}$	$5.86\ 10^{-6}$	$7.95\ 10^{1}$	$4.37\ 10^{-6}$	$5.93\ 10^{1}$	$3.48\ 10^{-6}$	$4.72\ 10^{1}$

Radionuclide	Radiological capacity (MBq)	Dose to adult (µSv y^{-1} MBq^{-1})	Adult dose from radiological capacity (µSv y^{-1})	Dose to child (µSv y^{-1} MBq^{-1})	Child dose from radiological capacity (µSv y^{-1})	Dose to infant (µSv y^{-1} MBq^{-1})	Infant dose from radiological capacity (µSv y^{-1})
U-232	$4.04\ 10^{8}$	$3.06\ 10^{-8}$	$1.24\ 10^{1}$	$3.25\ 10^{-8}$	$1.31\ 10^{1}$	$5.49\ 10^{-8}$	$2.21\ 10^{1}$
U-233	$1.02\ 10^{8}$	$2.10\ 10^{-8}$	$2.15\ 10^{0}$	$1.69\ 10^{-8}$	$1.73\ 10^{0}$	$2.41\ 10^{-8}$	$2.47\ 10^{0}$
U-234	$1.45\ 10^{8}$	$1.49\ 10^{-8}$	$2.16\ 10^{0}$	$1.17\ 10^{-8}$	$1.70\ 10^{0}$	$1.73\ 10^{-8}$	$2.51\ 10^{0}$
U-235	$6.93\ 10^{7}$	$1.27\ 10^{-7}$	$8.81\ 10^{0}$	$1.08\ 10^{-7}$	$7.45\ 10^{0}$	$8.87\ 10^{-8}$	$6.15\ 10^{0}$
U-236	$1.48\ 10^{9}$	$1.40\ 10^{-8}$	$2.08\ 10^{1}$	$1.10\ 10^{-8}$	$1.62\ 10^{1}$	$1.71\ 10^{-8}$	$2.54\ 10^{1}$
U-238	$1.60\ 10^{9}$	$1.41\ 10^{-8}$	$2.26\ 10^{1}$	$1.15\ 10^{-8}$	$1.85\ 10^{1}$	$1.90\ 10^{-8}$	$3.04\ 10^{1}$
Np-237	$1.42\ 10^{7}$	$1.22\ 10^{-7}$	$1.73\ 10^{0}$	$6.14\ 10^{-8}$	$8.71\ 10^{-1}$	$6.14\ 10^{-8}$	$8.71\ 10^{-1}$
Pu-238	$7.56\ 10^{8}$	$2.43\ 10^{-8}$	$1.84\ 10^{1}$	$1.29\ 10^{-8}$	$9.72\ 10^{0}$	$1.57\ 10^{-8}$	$1.18\ 10^{1}$
Pu-239	$1.55\ 10^{8}$	$8.63\ 10^{-8}$	$1.34\ 10^{1}$	$4.66\ 10^{-8}$	$7.24\ 10^{0}$	$5.36\ 10^{-8}$	$8.33\ 10^{0}$
Pu-240	$1.89\ 10^{8}$	$8.53\ 10^{-8}$	$1.61\ 10^{1}$	$4.60\ 10^{-8}$	$8.70\ 10^{0}$	$5.30\ 10^{-8}$	$1.00\ 10^{1}$
Pu-241	$9.39\ 10^{9}$	$2.37\ 10^{-9}$	$2.23\ 10^{1}$	$1.34\ 10^{-9}$	$1.26\ 10^{1}$	$1.51\ 10^{-9}$	$1.42\ 10^{1}$
Pu-242	$1.58\ 10^{8}$	$8.12\ 10^{-8}$	$1.29\ 10^{1}$	$4.57\ 10^{-8}$	$7.24\ 10^{0}$	$5.12\ 10^{-8}$	$8.11\ 10^{0}$
Pu-244	$1.26\ 10^{8}$	$8.28\ 10^{-8}$	$1.04\ 10^{1}$	$4.67\ 10^{-8}$	$5.88\ 10^{0}$	$5.42\ 10^{-8}$	$6.81\ 10^{0}$
Am-241	$3.03\ 10^{8}$	$6.87\ 10^{-8}$	$2.08\ 10^{1}$	$3.89\ 10^{-8}$	$1.18\ 10^{1}$	$4.37\ 10^{-8}$	$1.32\ 10^{1}$
Am-242m	$1.75\ 10^{7}$	$8.46\ 10^{-8}$	$1.48\ 10^{0}$	$4.55\ 10^{-8}$	$7.95\ 10^{-1}$	$5.30\ 10^{-8}$	$9.26\ 10^{-1}$
Am-243	$1.46\ 10^{8}$	$9.97\ 10^{-8}$	$1.46\ 10^{1}$	$6.05\ 10^{-8}$	$8.83\ 10^{0}$	$6.38\ 10^{-8}$	$9.31\ 10^{0}$
Cm-242	$1.48\ 10^{11}$	$1.24\ 10^{-10}$	$1.84\ 10^{1}$	$6.57\ 10^{-11}$	$9.72\ 10^{0}$	$8.00\ 10^{-11}$	$1.18\ 10^{1}$
Cm-243	$4.89\ 10^{7}$	$3.99\ 10^{-9}$	$1.95\ 10^{-1}$	$2.88\ 10^{-9}$	$1.41\ 10^{-1}$	$2.77\ 10^{-9}$	$1.36\ 10^{-1}$
Cm-244	$1.16\ 10^{8}$	$3.92\ 10^{-10}$	$4.56\ 10^{-2}$	$2.17\ 10^{-10}$	$2.52\ 10^{-2}$	$2.76\ 10^{-10}$	$3.20\ 10^{-2}$
Cm-245	$1.26\ 10^{7}$	$1.31\ 10^{-7}$	$1.64\ 10^{0}$	$8.52\ 10^{-8}$	$1.07\ 10^{0}$	$8.10\ 10^{-8}$	$1.02\ 10^{0}$
Cm-246	$1.27\ 10^{7}$	$8.28\ 10^{-8}$	$1.05\ 10^{0}$	$4.34\ 10^{-8}$	$5.50\ 10^{-1}$	$5.07\ 10^{-8}$	$6.42\ 10^{-1}$
Cm-248	$1.45\ 10^{7}$	$3.10\ 10^{-7}$	$4.51\ 10^{0}$	$1.67\ 10^{-7}$	$2.43\ 10^{0}$	$1.95\ 10^{-7}$	$2.83\ 10^{0}$
Ra-226$^{\$}$	$3.89\ 10^{6}$	$4.68\ 10^{-11}$	$1.82\ 10^{-4}$	$6.71\ 10^{-11}$	$2.61\ 10^{-4}$	$9.36\ 10^{-11}$	$3.64\ 10^{-4}$

Note: $ Assumes Ra-226 distributed below 5 m depth.

1006. The doses calculated using illustrative inventories are considered further in Appendix D.

E.5.7. Excavation for Housing – radon exposure from a house on spoil

1007. This case considers building a house on a spoil/waste mix.

1008. This corresponds to a case in which the cap has either completely degraded, or has been destroyed in the intrusion event; thus the house has been built directly upon contaminated soil.

1009. This case considers long-term occupation of the former landfill site, and thus long-term potential exposure to contaminated wastes.

E.5.7.1. Assessment calculation for radon exposure

1010. The radon model for spoil uses the original model from which the version in SNIFFER is derived. The flux of radon, $F_{radon}(t)$ (Bq m^{-2} y^{-1}), from bare waste is calculated according to (NEA, 1987):

$$F_{radon}(t) = \lambda_{Rn-222} \cdot C_{Ra-226} \cdot e^{-\lambda_{Ra-226}t} \cdot Dil \,.\, \rho_{soil} \cdot \tau \cdot h_{soil} \cdot \varepsilon$$

where:

- C_{Ra-226} is the initial Ra-226 concentration in the waste (Bq kg^{-1});
- t is the time at which the flux is evaluated;
- Dil is the fraction of waste in soil;
- ρ_{waste} is the bulk density of the waste (kg m^{-3}) see Table 72;
- τ is the emanation factor, the fraction of the radon atoms produced which escape from the solid phase of the waste into the pore spaces;
- ε is the self-confinement factor; and,
- h_{soil} is the thickness of the soil (m);

1011. The self-confinement factor is calculated from:

$$\varepsilon = \frac{H_{soil}}{h_{soil}} \tanh \frac{h_{soil}}{H_{soil}}$$

where:

- H_{soil} is the effective diffusion relaxation length for the soil.

1012. The effective relaxation length for soil is 0.2 m and the thickness of soil for a house built on spoil is assumed to be 0.1 m.

E.5.7.2. Dose from radon when building on a waste/spoil mix

1013. In Table 148, the results of assessment calculations for radon gas and a dilution factor of 0.108 are presented for waste containing 5 Bq g^{-1} of Ra-226.

Table 148 Radon inhalation doses and radiological capacities for a dwelling built on a waste/spoil mix buried with other LLW at any depth

Case	Indoor Rn-222 activity concentration (Bq m^{-3} MBq^{-1})	Inhalation effective dose (mSv y^{-1} for 5 Bq g^{-1})	Radiological capacity based on radon dose and 3 mSv/y (TBq)
Adult	$2.23\ 10^{-6}$	$2.99\ 10^{0}$	$3.19\ 10^{1}$
Child	$2.23\ 10^{-6}$	$4.02\ 10^{0}$	$2.37\ 10^{1}$
Infant	$2.23\ 10^{-6}$	$5.61\ 10^{0}$	$1.70\ 10^{1}$

1014. The calculations imply that the average activity concentration of Ra-226 in wastes that are excavated in this scenario, and that will meet the 3 mSv dose criterion, is about 2.7 Bq g^{-1}, based on the dose to an infant. An upper level of 5 Bq g^{-1} would ensure an average activity concentration in waste that was around this level. This restriction only applies to the activity concentration of Ra-226 in wastes that could be excavated in this scenario so that a dwelling is built on a spoil/waste mixture containing the radium bearing waste. This scenario does not impose restrictions on the Ra-226 activity concentration of wastes that are buried deeper than the excavation depth. Waste emplacement strategies within waste cells can be employed to obviate the constraints imposed by this scenario. If it is cautiously assumed that the maximum depth of any human intrusion event leading to a dwelling built on spoil is 5 m, then ensuring that waste containing Ra-226 above 5 Bq g^{-1} is placed at depths greater than this will prevent it becoming mixed with spoil.

1015. The possibility of radon migration through the remaining cell-filling material must also be considered. Conceptually, this is the same calculation as considered in Section E.3.5 except modelling migration of radon through cell-filling material (i.e. soil, soil-like waste and other non-radium bearing wastes) instead of considering radon migration through an intact cap.

1016. Scoping calculations suggest, therefore, that consideration of waste emplacement strategies (i.e. placing radium bearing wastes at depths of greater than 5 m below the restored surface of the waste cells) may allow constraints upon the site's radium capacity to be set at a higher level.

1017. If wastes containing significant activity concentrations of Ra-226 were placed at depths of greater than 5 m, then this would result in radon migrating through cover material. As discussed earlier as cover depth increases the dose from radon declines. Radium will be placed at various depths from 5 m below the restored surface. The minimum depth which would apply to Radium wastes (and to any LLW) would be 2.3 m since LLW is not placed within the top 1 m of a cell and the cap is 1.3 m thick. A value of 5 Bq g^{-1}, corresponding to the activity concentration specified in the NORM exemption level (see paragraph 145), has been used to limit disposals in the upper layers of waste cells.

1018. The indoor Rn-222 activity concentration can be compared with the HPA radon action level of 200 Bq m^{-3} and the target level of 100 Bq m^{-3} for new dwellings (http://www.ukradon.org/information/level). The geometric mean radon level in the unitary authority of Middlesbrough is 15 Bq m^{-3}, with a highest recorded value, prior to 2002, of 41 Bq m^{-3} (NRPB, 2002). These figures should be compared to a geometric mean radon level in England of 50 Bq m^{-3}, and a geometric mean radon level in Cleveland, the postcode area in which Port Clarence is located, of 20 Bq m^{-3}. If the quantities of radium emplaced in Port Clarence are equal to the radiological capacity given in Table 148 above, then this would result in an indoor Rn-222 activity concentration of approximately 70 Bq m^{-3} (below the HPA action level).

E.5.8. Excavation for a smallholding

E.5.8.1. Assessment calculations for the Smallholder

1019. Occupancy of a smallholding is considered as this is more cautious than a larger farm because it assumes more crops are grown on a relatively small area.

1020. The smallholding case is conceptually similar to the long-term residential occupant described in Section E.5.6 but includes additional exposure pathways: it is assumed that the smallholder may grow green and root vegetables, farm some livestock (e.g. cows) and that they consume both the meat and milk from this livestock. In consequence, the mathematical model for the smallholder is based on that of the residential occupant, and the following equation that calculates the dose arising from ingesting animal foodstuff (e.g. meat and milk) raised on contaminated land is given by Galson (Augean, 2009):

$$Dose_{ing,animal} = \sum_{animal} \{Q_{animal} \cdot [q_{soil} \cdot C_{Rn,soil}(t) + q_{pasture} \cdot C_{Rn,soil}(t) \cdot UF_{Rn,grass}] \cdot TF_{Rn,animal}\} \cdot D_{Rn,ing}$$

where:

- Q_{animal} is the consumption rate of animal foodstuff (kg y^{-1});
- q_{soil} is the soil consumption rate by the animal (kg d^{-1});
- $q_{pasture}$ is the pasture consumption rate by the animal (kg d^{-1});
- $UF_{Rn,grass}$ is the soil to grass transfer factor for radionuclide *Rn* (Bq kg^{-1} fresh weight of crop per Bq kg^{-1} of soil);
- $TF_{Rn,animal}$ is the animal product transfer factor for radionuclide *Rn* (d kg^{-1});

- $C_{Rn,soil}(t)$ is the activity concentration of radionuclide *Rn* at time *t* (Bq kg^{-1}); and,
- $D_{Rn,ing}$ is the dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1}).

1021. The smallholding calculation is carried out at 200 years after closure. Note that the overall dilution factor applied to LLW for soil used for the crops and livestock is 0.0108 as discussed above (see paragraph 988). The house is assumed to be built on an intact part of the cap. External exposure inside the house is dominated by the contribution from the surrounding soil (with SF) rather than by the direct radiation through the floor. Soil to crop transfer factors are given in Table 203 and dose coefficients for ingestion are given in Table 200. Relevant parameters for the smallholding scenario are given in Table 149 and animal produce transfer factors are given in Table 204.

Table 149 Parameters for smallholding scenario

Parameter	Substance	Value
Consumption rate (animal)	Pasture	55 kg d^{-1}
	Soils	0.6 kg d^{-1}
Consumption rates – adult$	Green vegetables	35 kg y^{-1}/ 80 kg y^{-1}
	Root vegetables	60 kg y^{-1}/ 130 kg y^{-1}
	Meat	23 kg y^{-1}/ 70 kg y^{-1}
	Milk	122.5 kg y^{-1}/ 240 kg y^{-1}
Consumption rates – child$	Green vegetables	15 kg y^{-1}/ 35 kg y^{-1}
	Root vegetables	50 kg y^{-1}/ 90 kg y^{-1}
	Meat	19 kg y^{-1}/ 40 kg y^{-1}
	Milk	127 kg y^{-1}/ 240 kg y^{-1}
Consumption rates – infant$	Green vegetables	5 kg y^{-1}/ 15 kg y^{-1}
	Root vegetables	15 kg y^{-1}/ 45 kg y^{-1}
	Meat	3.8 kg y^{-1}/ 13 kg y^{-1}
	Milk	148 kg y^{-1}/ 320 kg y^{-1}
Occupancy Indoors – adult		0.75 y y^{-1}
Occupancy outdoors - adult		0.25 y y^{-1}
Occupancy Indoors – child		0.84 y y^{-1}
Occupancy outdoors - child		0.16 y y^{-1}
Occupancy Indoors - infant		0.91 y y^{-1}
Occupancy outdoors - infant		0.09 y y^{-1}
Shielding factor indoors		0.1
Occupancy dust		2191.5 h y^{-1}
Dustload		1 10^{-7} kg m^{-3}
Breathing rate adult		1 m^3 h^{-1}
Breathing rate child		0.64 m^3 h^{-1}
Breathing rate infant		0.22 m^3 h^{-1}
Dilution factor	Soil on land	0.0108
Dilution factor	Soil under house	0.108

Values except dilution factors taken from (Augean, 2009)
$Rates are average and 97.5th percentile

1022. The assessment calculations presented for the smallholding scenario also include a gas contribution based on gas migration from underlying waste (see Section E.5.5) and in the case of radon from excavated waste remaining directly under the house. The average gas release rates for H-3 and C-14 used were 50 and 900, respectively.

1023. The radon model for spoil uses the original model from which the version in SNIFFER is derived (see Section E.5.7.2).

E.5.8.2. Assessment calculation for radon exposure

1024. The flux of radon, $F_{radon}(t)$ (Bq m^{-2} y^{-1}), from bare waste is calculated according to (NEA, 1987):

$$F_{radon}(t) = \lambda_{Rn-222} \cdot C_{Ra-226} \cdot e^{-\lambda_{Ra-226}t} \cdot Dil \,.\, \rho_{soil} \cdot \tau \cdot h_{soil} \cdot \varepsilon$$

where:

- C_{Ra-226} is the initial Ra-226 concentration in the waste (Bq kg^{-1});
- t is the time at which the flux is evaluated;
- Dil is the fraction of waste in soil;
- ρ_{waste} is the bulk density of the waste (kg m^{-3}) see Table 107;
- τ is the emanation factor, the fraction of the radon atoms produced which escape from the solid phase of the waste into the pore spaces;
- ε is the self-confinement factor; and,
- h_{soil} is the thickness of the soil (m);

1025. The self-confinement factor is calculated from:

$$\varepsilon = \frac{H_{soil}}{h_{soil}} \tanh \frac{h_{soil}}{H_{soil}}$$

where:

- H_{soil} is the effective diffusion relaxation length for the soil.

1026. The effective relaxation length for soil is 0.2 m and the thickness of soil is assumed to be 0.1 m.

E.5.8.3. Dose to Smallholder on site after 200 years

1027. In Table 150 the dose rates to an adult, child and infant smallholder respectively on the site following construction of a slurry pit 200 years after site capping are presented.

The largest dose rates per MBq arise from Ra-226 assuming no emplacement strategy for wastes containing > 5 Bq g^{-1}, Cl-36, Se-79 and Tc-99 to infants.

Note that the radiological capacity is conservatively based on the dose to a smallholder assuming that intrusion occurs 60 years after closure. The sensitivity of the dose and hence the capacity to the time of intrusion is discussed in Section E.8.1.1.

Table 150 Doses to smallholder family after 200 years

Radionuclide	Radiological capacity (MBq)	Dose to adult ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Adult dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to child ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to infant ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose from radiological capacity ($\mu Sv\ y^{-1}$)
H-3	$6.43\ 10^{9}$	$1.04\ 10^{-11}$	$6.69\ 10^{-2}$	$1.09\ 10^{-11}$	$7.00\ 10^{-2}$	$1.12\ 10^{-11}$	$7.22\ 10^{-2}$
C-14	$1.87\ 10^{8}$	$1.47\ 10^{-6}$	$2.75\ 10^{2}$	$1.34\ 10^{-6}$	$2.52\ 10^{2}$	$1.17\ 10^{-6}$	$2.20\ 10^{2}$
Cl-36	$1.56\ 10^{8}$	$3.83\ 10^{-6}$	$5.99\ 10^{2}$	$6.12\ 10^{-6}$	$9.58\ 10^{2}$	$1.92\ 10^{-5}$	$3.00\ 10^{3}$
Ca-41	$5.77\ 10^{9}$	$3.76\ 10^{-8}$	$2.17\ 10^{2}$	$6.21\ 10^{-8}$	$3.58\ 10^{2}$	$4.60\ 10^{-8}$	$2.66\ 10^{2}$
Mn-54	$1.12\ 10^{13}$	$3.41\ 10^{-77}$	$3.82\ 10^{-64}$	$2.73\ 10^{-77}$	$3.06\ 10^{-64}$	$2.22\ 10^{-77}$	$2.49\ 10^{-64}$
Fe-55	$1.86\ 10^{13}$	$5.35\ 10^{-32}$	$9.93\ 10^{-19}$	$1.03\ 10^{-31}$	$1.92\ 10^{-18}$	$9.14\ 10^{-32}$	$1.70\ 10^{-18}$
Co-60	$3.58\ 10^{11}$	$9.71\ 10^{-18}$	$3.47\ 10^{-6}$	$7.46\ 10^{-18}$	$2.67\ 10^{-6}$	$5.66\ 10^{-18}$	$2.02\ 10^{-6}$
Ni-59	$1.95\ 10^{11}$	$1.25\ 10^{-9}$	$2.44\ 10^{2}$	$1.87\ 10^{-9}$	$3.64\ 10^{2}$	$5.92\ 10^{-9}$	$1.16\ 10^{3}$
Ni-63	$2.42\ 10^{11}$	$7.46\ 10^{-10}$	$1.80\ 10^{2}$	$1.19\ 10^{-9}$	$2.88\ 10^{2}$	$3.67\ 10^{-9}$	$8.87\ 10^{2}$
Zn-65	$8.95\ 10^{11}$	$1.74\ 10^{-96}$	$1.56\ 10^{-84}$	$1.30\ 10^{-96}$	$1.16\ 10^{-84}$	$1.34\ 10^{-96}$	$1.20\ 10^{-84}$
Se-79	$8.98\ 10^{8}$	$1.09\ 10^{-6}$	$9.81\ 10^{2}$	$3.34\ 10^{-6}$	$3.00\ 10^{3}$	$3.10\ 10^{-6}$	$2.78\ 10^{3}$
Sr-90	$3.83\ 10^{8}$	$1.67\ 10^{-7}$	$6.39\ 10^{1}$	$2.30\ 10^{-7}$	$8.82\ 10^{1}$	$2.69\ 10^{-7}$	$1.03\ 10^{2}$
Mo-93	$1.44\ 10^{9}$	$9.54\ 10^{-7}$	$1.37\ 10^{3}$	$1.08\ 10^{-6}$	$1.56\ 10^{3}$	$2.04\ 10^{-6}$	$2.93\ 10^{3}$
Zr-93	$3.12\ 10^{11}$	$2.03\ 10^{-9}$	$6.31\ 10^{2}$	$6.59\ 10^{-10}$	$2.05\ 10^{2}$	$4.44\ 10^{-10}$	$1.38\ 10^{2}$
Nb-93m	$5.06\ 10^{10}$	$8.34\ 10^{-14}$	$4.22\ 10^{-3}$	$1.14\ 10^{-13}$	$5.77\ 10^{-3}$	$1.86\ 10^{-13}$	$9.40\ 10^{-3}$
Nb-94	$6.09\ 10^{6}$	$1.50\ 10^{-6}$	$9.16\ 10^{0}$	$1.13\ 10^{-6}$	$6.89\ 10^{0}$	$8.44\ 10^{-7}$	$5.14\ 10^{0}$
Tc-99	$6.12\ 10^{8}$	$2.28\ 10^{-6}$	$1.40\ 10^{3}$	$2.87\ 10^{-6}$	$1.76\ 10^{3}$	$4.90\ 10^{-6}$	$3.00\ 10^{3}$
Ru-106	$9.14\ 10^{11}$	$4.71\ 10^{-66}$	$4.30\ 10^{-54}$	$4.49\ 10^{-66}$	$4.10\ 10^{-54}$	$4.30\ 10^{-66}$	$3.93\ 10^{-54}$
Ag-108m	$2.65\ 10^{8}$	$1.08\ 10^{-6}$	$2.86\ 10^{2}$	$8.10\ 10^{-7}$	$2.15\ 10^{2}$	$6.04\ 10^{-7}$	$1.60\ 10^{2}$
Ag-110m	$6.41\ 10^{12}$	$2.55\ 10^{-94}$	$1.64\ 10^{-81}$	$1.92\ 10^{-94}$	$1.23\ 10^{-81}$	$1.43\ 10^{-94}$	$9.13\ 10^{-82}$
Cd-109	$1.04\ 10^{12}$	$1.93\ 10^{-54}$	$2.00\ 10^{-42}$	$2.30\ 10^{-54}$	$2.39\ 10^{-42}$	$2.85\ 10^{-54}$	$2.96\ 10^{-42}$
Sb-125	$4.17\ 10^{11}$	$5.83\ 10^{-29}$	$2.43\ 10^{-17}$	$4.41\ 10^{-29}$	$1.84\ 10^{-17}$	$3.33\ 10^{-29}$	$1.39\ 10^{-17}$
Sn-119m	$8.43\ 10^{12}$	$1.29\ 10^{-83}$	$1.09\ 10^{-70}$	$1.82\ 10^{-83}$	$1.53\ 10^{-70}$	$3.55\ 10^{-83}$	$3.00\ 10^{-70}$
Sn-123	$2.97\ 10^{12}$	$5.62\ 10^{-178}$	$1.67\ 10^{-165}$	$7.66\ 10^{-178}$	$2.28\ 10^{-165}$	$1.53\ 10^{-177}$	$4.55\ 10^{-165}$

Radionuclide	Radiological capacity (MBq)	Dose to adult (µSv y^{-1} MBq^{-1})	Adult dose from radiological capacity (µSv y^{-1})	Dose to child (µSv y^{-1} MBq^{-1})	Child dose from radiological capacity (µSv y^{-1})	Dose to infant (µSv y^{-1} MBq^{-1})	Infant dose from radiological capacity (µSv y^{-1})
Sn-126	4.60 10^{6}	2.04 10^{-6}	9.39 10^{0}	1.65 10^{-6}	7.59 10^{0}	1.51 10^{-6}	6.94 10^{0}
Te-127m	4.07 10^{12}	5.66 10^{-214}	2.30 10^{-201}	8.30 10^{-214}	3.38 10^{-201}	1.74 10^{-213}	7.08 10^{-201}
I-129	3.01 10^{8}	7.70 10^{-6}	2.32 10^{3}	9.97 10^{-6}	3.00 10^{3}	9.52 10^{-6}	2.86 10^{3}
Ba-133	7.18 10^{9}	5.90 10^{-13}	4.23 10^{-3}	4.49 10^{-13}	3.22 10^{-3}	3.34 10^{-13}	2.40 10^{-3}
Cs-134	1.01 10^{11}	1.37 10^{-35}	1.39 10^{-24}	9.52 10^{-36}	9.63 10^{-25}	7.22 10^{-36}	7.30 10^{-25}
Cs-135	1.55 10^{9}	5.10 10^{-8}	7.91 10^{1}	3.10 10^{-8}	4.81 10^{1}	3.04 10^{-8}	4.72 10^{1}
Cs-137	9.69 10^{8}	8.71 10^{-9}	8.44 10^{0}	5.86 10^{-9}	5.68 10^{0}	4.59 10^{-9}	4.44 10^{0}
Ce-144	4.81 10^{12}	8.48 10^{-86}	4.08 10^{-73}	6.94 10^{-86}	3.34 10^{-73}	8.10 10^{-86}	3.90 10^{-73}
Pm-147	2.14 10^{13}	4.01 10^{-33}	8.59 10^{-20}	5.50 10^{-33}	1.18 10^{-19}	8.89 10^{-33}	1.90 10^{-19}
Sm-147	4.81 10^{8}	4.18 10^{-8}	2.01 10^{1}	3.28 10^{-8}	1.58 10^{1}	4.05 10^{-8}	1.94 10^{1}
Sm-151	7.23 10^{11}	1.67 10^{-11}	1.21 10^{1}	2.10 10^{-11}	1.52 10^{1}	3.91 10^{-11}	2.82 10^{1}
Eu-152	8.05 10^{9}	3.91 10^{-11}	3.15 10^{-1}	2.94 10^{-11}	2.36 10^{-1}	2.18 10^{-11}	1.76 10^{-1}
Eu-154	4.18 10^{10}	1.18 10^{-13}	4.94 10^{-3}	8.87 10^{-14}	3.71 10^{-3}	6.61 10^{-14}	2.76 10^{-3}
Eu-155	8.81 10^{12}	6.50 10^{-21}	5.73 10^{-8}	4.92 10^{-21}	4.34 10^{-8}	3.77 10^{-21}	3.32 10^{-8}
Gd-153	4.83 10^{13}	1.29 10^{-99}	6.20 10^{-86}	9.65 10^{-100}	4.66 10^{-86}	7.23 10^{-100}	3.49 10^{-86}
Pb-210	4.85 10^{8}	1.36 10^{-8}	6.60 10^{0}	2.03 10^{-8}	9.86 10^{0}	2.85 10^{-8}	1.38 10^{1}
Po-210	6.17 10^{9}	8.10 10^{-166}	5.00 10^{-156}	1.14 10^{-165}	7.02 10^{-156}	2.92 10^{-165}	1.80 10^{-155}
Ra-226	3.89 10^{6}	5.93 10^{-5}	2.30 10^{2}	8.79 10^{-5}	3.42 10^{2}	1.23 10^{-4}	4.78 10^{2}
Ra-228	2.25 10^{10}	5.14 10^{-16}	1.16 10^{-5}	1.49 10^{-15}	3.34 10^{-5}	1.30 10^{-15}	2.93 10^{-5}
Ac-227	3.04 10^{9}	1.66 10^{-9}	5.04 10^{0}	1.39 10^{-9}	4.22 10^{0}	1.63 10^{-9}	4.96 10^{0}
Th-228	1.72 10^{11}	5.40 10^{-38}	9.27 10^{-27}	4.23 10^{-38}	7.26 10^{-27}	3.44 10^{-38}	5.91 10^{-27}
Th-229	2.88 10^{7}	5.73 10^{-7}	1.65 10^{1}	5.05 10^{-7}	1.45 10^{1}	5.14 10^{-7}	1.48 10^{1}
Th-230	1.98 10^{6}	2.99 10^{-6}	5.92 10^{0}	4.46 10^{-6}	8.81 10^{0}	6.20 10^{-6}	1.23 10^{1}
Th-232	7.95 10^{6}	1.42 10^{-6}	1.13 10^{1}	1.84 10^{-6}	1.46 10^{1}	1.66 10^{-6}	1.32 10^{1}
Pa-231	1.36 10^{7}	1.12 10^{-5}	1.51 10^{2}	8.96 10^{-6}	1.22 10^{2}	6.34 10^{-6}	8.60 10^{1}
U-232	4.04 10^{8}	3.37 10^{-8}	1.36 10^{1}	4.58 10^{-8}	1.85 10^{1}	7.86 10^{-8}	3.17 10^{1}
U-233	1.02 10^{8}	3.86 10^{-8}	3.95 10^{0}	3.74 10^{-8}	3.82 10^{0}	5.32 10^{-8}	5.44 10^{0}

Radionuclide	Radiological capacity (MBq)	Dose to adult ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Adult dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to child ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Child dose from radiological capacity ($\mu Sv\ y^{-1}$)	Dose to infant ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Infant dose from radiological capacity ($\mu Sv\ y^{-1}$)
U-234	$1.45\ 10^{8}$	$2.68\ 10^{-8}$	$3.89\ 10^{0}$	$2.64\ 10^{-8}$	$3.84\ 10^{0}$	$4.04\ 10^{-8}$	$5.86\ 10^{0}$
U-235	$6.93\ 10^{7}$	$1.81\ 10^{-7}$	$1.25\ 10^{1}$	$1.44\ 10^{-7}$	$1.00\ 10^{1}$	$1.26\ 10^{-7}$	$8.76\ 10^{0}$
U-236	$1.48\ 10^{9}$	$2.53\ 10^{-8}$	$3.75\ 10^{1}$	$2.48\ 10^{-8}$	$3.68\ 10^{1}$	$4.02\ 10^{-8}$	$5.96\ 10^{1}$
U-238	$1.60\ 10^{9}$	$2.57\ 10^{-8}$	$4.13\ 10^{1}$	$2.64\ 10^{-8}$	$4.24\ 10^{1}$	$4.47\ 10^{-8}$	$7.17\ 10^{1}$
Np-237	$1.42\ 10^{7}$	$2.15\ 10^{-7}$	$3.05\ 10^{0}$	$1.08\ 10^{-7}$	$1.53\ 10^{0}$	$9.87\ 10^{-8}$	$1.40\ 10^{0}$
Pu-238	$7.56\ 10^{8}$	$2.19\ 10^{-8}$	$1.66\ 10^{1}$	$1.32\ 10^{-8}$	$1.00\ 10^{1}$	$1.38\ 10^{-8}$	$1.04\ 10^{1}$
Pu-239	$1.55\ 10^{8}$	$1.15\ 10^{-7}$	$1.79\ 10^{1}$	$7.13\ 10^{-8}$	$1.11\ 10^{1}$	$7.01\ 10^{-8}$	$1.09\ 10^{1}$
Pu-240	$1.89\ 10^{8}$	$1.14\ 10^{-7}$	$2.15\ 10^{1}$	$7.02\ 10^{-8}$	$1.33\ 10^{1}$	$6.90\ 10^{-8}$	$1.30\ 10^{1}$
Pu-241	$9.39\ 10^{9}$	$3.17\ 10^{-9}$	$2.98\ 10^{1}$	$1.96\ 10^{-9}$	$1.84\ 10^{1}$	$1.89\ 10^{-9}$	$1.77\ 10^{1}$
Pu-242	$1.58\ 10^{8}$	$1.09\ 10^{-7}$	$1.73\ 10^{1}$	$6.97\ 10^{-8}$	$1.10\ 10^{1}$	$6.71\ 10^{-8}$	$1.06\ 10^{1}$
Pu-244	$1.26\ 10^{8}$	$1.12\ 10^{-7}$	$1.41\ 10^{1}$	$7.17\ 10^{-8}$	$9.02\ 10^{0}$	$7.14\ 10^{-8}$	$8.98\ 10^{0}$
Am-241	$3.03\ 10^{8}$	$9.20\ 10^{-8}$	$2.78\ 10^{1}$	$5.67\ 10^{-8}$	$1.72\ 10^{1}$	$5.47\ 10^{-8}$	$1.66\ 10^{1}$
Am-242m	$1.75\ 10^{7}$	$1.04\ 10^{-7}$	$1.81\ 10^{0}$	$6.28\ 10^{-8}$	$1.10\ 10^{0}$	$6.24\ 10^{-8}$	$1.09\ 10^{0}$
Am-243	$1.46\ 10^{8}$	$1.40\ 10^{-7}$	$2.05\ 10^{1}$	$8.88\ 10^{-8}$	$1.30\ 10^{1}$	$8.34\ 10^{-8}$	$1.22\ 10^{1}$
Cm-242	$1.48\ 10^{11}$	$1.12\ 10^{-10}$	$1.66\ 10^{1}$	$6.76\ 10^{-11}$	$1.00\ 10^{1}$	$7.06\ 10^{-11}$	$1.04\ 10^{1}$
Cm-243	$4.89\ 10^{7}$	$1.66\ 10^{-9}$	$8.11\ 10^{-2}$	$1.11\ 10^{-9}$	$5.42\ 10^{-2}$	$1.05\ 10^{-9}$	$5.15\ 10^{-2}$
Cm-244	$1.16\ 10^{8}$	$3.48\ 10^{-10}$	$4.04\ 10^{-2}$	$2.16\ 10^{-10}$	$2.51\ 10^{-2}$	$2.17\ 10^{-10}$	$2.52\ 10^{-2}$
Cm-245	$1.26\ 10^{7}$	$1.75\ 10^{-7}$	$2.20\ 10^{0}$	$1.13\ 10^{-7}$	$1.42\ 10^{0}$	$9.96\ 10^{-8}$	$1.25\ 10^{0}$
Cm-246	$1.27\ 10^{7}$	$1.19\ 10^{-7}$	$1.51\ 10^{0}$	$6.88\ 10^{-8}$	$8.72\ 10^{-1}$	$6.88\ 10^{-8}$	$8.72\ 10^{-1}$
Cm-248	$1.45\ 10^{7}$	$4.49\ 10^{-7}$	$6.52\ 10^{0}$	$2.68\ 10^{-7}$	$3.90\ 10^{0}$	$2.67\ 10^{-7}$	$3.88\ 10^{0}$
Ra-226$	$3.89\ 10^{6}$	$4.58\ 10^{-11}$	$1.78\ 10^{-4}$	$6.56\ 10^{-11}$	$2.55\ 10^{-4}$	$9.16\ 10^{-11}$	$3.56\ 10^{-4}$

Note: $ Assumes Ra-226 distributed below 5 m depth.

1028. The critical group consumption rate is applied to the two foodstuffs with the greatest contribution to dose rate. This varies by radionuclide as shown below in Table 151 for adult consumption. There are several cases where animal products result in larger dose rates (e.g. Cl-36, Cs-134 and Cs-137).

Table 151 Contributing foodstuff doses in the diet of an adult smallholder

Radionuclide	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)			
	Root vegetables	Green vegetables	Meat	Milk
H-3	$2.58\ 10^{-13}$	$6.94\ 10^{-14}$	$7.29\ 10^{-14}$	$2.62\ 10^{-13}$
C-14	$5.76\ 10^{-9}$	$3.36\ 10^{-9}$	$4.92\ 10^{-8}$	$1.41\ 10^{-8}$
Cl-36	$4.73\ 10^{-7}$	$2.76\ 10^{-7}$	$1.31\ 10^{-6}$	$1.77\ 10^{-6}$
Ca-41	$2.09\ 10^{-8}$	$1.29\ 10^{-8}$	$4.16\ 10^{-10}$	$3.32\ 10^{-9}$
Mn-54	$2.87\ 10^{-78}$	$1.76\ 10^{-78}$	$8.67\ 10^{-81}$	$2.77\ 10^{-81}$
Fe-55	$2.20\ 10^{-33}$	$3.95\ 10^{-34}$	$4.91\ 10^{-32}$	$1.29\ 10^{-34}$
Co-60	$8.51\ 10^{-20}$	$5.24\ 10^{-20}$	$4.67\ 10^{-21}$	$7.46\ 10^{-22}$
Ni-59	$4.16\ 10^{-10}$	$1.12\ 10^{-10}$	$2.09\ 10^{-11}$	$6.97\ 10^{-10}$
Ni-63	$2.49\ 10^{-10}$	$6.69\ 10^{-11}$	$1.25\ 10^{-11}$	$4.16\ 10^{-10}$
Zn-65	$8.44\ 10^{-99}$	$1.24\ 10^{-96}$	$7.38\ 10^{-98}$	$1.29\ 10^{-98}$
Se-79	$6.39\ 10^{-7}$	$3.93\ 10^{-7}$	$4.40\ 10^{-8}$	$1.51\ 10^{-8}$
Sr-90	$2.28\ 10^{-9}$	$1.01\ 10^{-7}$	$1.29\ 10^{-8}$	$5.03\ 10^{-8}$
Mo-93	$2.11\ 10^{-7}$	$9.07\ 10^{-8}$	$3.48\ 10^{-8}$	$6.17\ 10^{-7}$
Zr-93	$1.21\ 10^{-9}$	$7.47\ 10^{-10}$	$3.76\ 10^{-14}$	$1.10\ 10^{-13}$
Nb-93m	$4.90\ 10^{-14}$	$3.02\ 10^{-14}$	$2.99\ 10^{-19}$	$2.18\ 10^{-18}$
Nb-94	$3.72\ 10^{-9}$	$2.29\ 10^{-9}$	$2.27\ 10^{-14}$	$1.65\ 10^{-13}$
Tc-99	$1.41\ 10^{-6}$	$8.68\ 10^{-7}$	$1.37\ 10^{-9}$	$1.68\ 10^{-9}$
Ru-106	$2.17\ 10^{-67}$	$2.34\ 10^{-68}$	$1.64\ 10^{-66}$	$1.89\ 10^{-70}$
Ag-108m	$4.73\ 10^{-10}$	$2.65\ 10^{-11}$	$1.71\ 10^{-11}$	$2.97\ 10^{-10}$
Ag-110m	$7.65\ 10^{-98}$	$4.28\ 10^{-99}$	$2.77\ 10^{-99}$	$4.81\ 10^{-98}$
Cd-109	$1.42\ 10^{-54}$	$1.40\ 10^{-55}$	$3.43\ 10^{-55}$	$1.96\ 10^{-56}$
Sb-125	$4.32\ 10^{-31}$	$2.66\ 10^{-31}$	$3.51\ 10^{-34}$	$1.17\ 10^{-33}$
Sn-119m	$7.06\ 10^{-84}$	$4.34\ 10^{-84}$	$2.75\ 10^{-85}$	$7.71\ 10^{-85}$
Sn-123	$2.90\ 10^{-178}$	$1.79\ 10^{-178}$	$1.13\ 10^{-179}$	$3.17\ 10^{-179}$
Sn-126	$1.12\ 10^{-7}$	$6.88\ 10^{-8}$	$4.36\ 10^{-9}$	$1.22\ 10^{-8}$
Te-127m	$7.73\ 10^{-218}$	$1.88\ 10^{-217}$	$4.63\ 10^{-214}$	$1.02\ 10^{-214}$
I-129	$1.12\ 10^{-6}$	$6.53\ 10^{-7}$	$3.19\ 10^{-6}$	$2.73\ 10^{-6}$
Ba-133	$3.13\ 10^{-15}$	$1.93\ 10^{-15}$	$9.42\ 10^{-17}$	$2.81\ 10^{-16}$
Cs-134	$4.06\ 10^{-37}$	$2.37\ 10^{-37}$	$1.77\ 10^{-36}$	$9.62\ 10^{-37}$
Cs-135	$6.11\ 10^{-9}$	$3.56\ 10^{-9}$	$2.67\ 10^{-8}$	$1.45\ 10^{-8}$
Cs-137	$4.01\ 10^{-10}$	$2.34\ 10^{-10}$	$1.75\ 10^{-9}$	$9.51\ 10^{-10}$
Ce-144	$4.51\ 10^{-87}$	$4.63\ 10^{-87}$	$4.53\ 10^{-89}$	$3.62\ 10^{-88}$
Pm-147	$1.92\ 10^{-33}$	$5.16\ 10^{-34}$	$1.32\ 10^{-33}$	$9.21\ 10^{-36}$
Sm-147	$2.16\ 10^{-8}$	$1.33\ 10^{-8}$	$6.92\ 10^{-10}$	$1.45\ 10^{-10}$
Sm-151	$9.27\ 10^{-12}$	$5.70\ 10^{-12}$	$2.97\ 10^{-13}$	$6.20\ 10^{-14}$
Eu-152	$3.31\ 10^{-14}$	$2.04\ 10^{-14}$	$7.03\ 10^{-16}$	$3.98\ 10^{-16}$
Eu-154	$1.30\ 10^{-16}$	$8.03\ 10^{-17}$	$2.77\ 10^{-18}$	$1.57\ 10^{-18}$
Eu-155	$4.79\ 10^{-23}$	$2.95\ 10^{-23}$	$1.02\ 10^{-24}$	$5.75\ 10^{-25}$

Radionuclide	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)			
	Root vegetables	Green vegetables	Meat	Milk
Gd-153	$1.20\ 10^{-102}$	$1.23\ 10^{-102}$	$1.20\ 10^{-104}$	$9.62\ 10^{-104}$
Pb-210	$8.10\ 10^{-9}$	$4.98\ 10^{-9}$	$6.59\ 10^{-11}$	$2.63\ 10^{-10}$
Po-210	$3.10\ 10^{-167}$	$1.81\ 10^{-167}$	$5.52\ 10^{-166}$	$1.29\ 10^{-166}$
Ra-226*	$1.76\ 10^{-5}$	$1.08\ 10^{-5}$	$1.96\ 10^{-7}$	$1.51\ 10^{-6}$
Ra-228	$2.49\ 10^{-16}$	$1.53\ 10^{-16}$	$2.77\ 10^{-18}$	$2.13\ 10^{-17}$
Ac-227	$4.58\ 10^{-10}$	$2.82\ 10^{-10}$	$8.50\ 10^{-12}$	$1.13\ 10^{-13}$
Th-228	$5.13\ 10^{-40}$	$1.38\ 10^{-40}$	$9.36\ 10^{-40}$	$3.03\ 10^{-42}$
Th-229	$6.64\ 10^{-8}$	$1.79\ 10^{-8}$	$1.21\ 10^{-7}$	$3.92\ 10^{-10}$
Th-230	$1.61\ 10^{-6}$	$9.84\ 10^{-7}$	$5.99\ 10^{-8}$	$1.36\ 10^{-7}$
Th-232	$1.17\ 10^{-7}$	$3.16\ 10^{-8}$	$2.14\ 10^{-7}$	$6.94\ 10^{-10}$
Pa-231	$6.50\ 10^{-6}$	$4.00\ 10^{-6}$	$8.75\ 10^{-9}$	$2.12\ 10^{-9}$
U-232	$1.40\ 10^{-8}$	$8.59\ 10^{-9}$	$4.85\ 10^{-10}$	$3.45\ 10^{-9}$
U-233	$1.25\ 10^{-8}$	$7.26\ 10^{-9}$	$2.70\ 10^{-9}$	$2.78\ 10^{-9}$
U-234	$1.08\ 10^{-8}$	$6.66\ 10^{-9}$	$4.51\ 10^{-10}$	$2.67\ 10^{-9}$
U-235	$3.70\ 10^{-8}$	$2.27\ 10^{-8}$	$3.79\ 10^{-10}$	$2.59\ 10^{-9}$
U-236	$1.04\ 10^{-8}$	$6.38\ 10^{-9}$	$3.60\ 10^{-10}$	$2.56\ 10^{-9}$
U-238	$1.07\ 10^{-8}$	$6.57\ 10^{-9}$	$3.71\ 10^{-10}$	$2.64\ 10^{-9}$
Np-237	$2.45\ 10^{-8}$	$1.50\ 10^{-7}$	$3.79\ 10^{-9}$	$1.03\ 10^{-10}$
Pu-238	$1.04\ 10^{-8}$	$6.45\ 10^{-10}$	$1.22\ 10^{-11}$	$7.85\ 10^{-12}$
Pu-239	$5.48\ 10^{-8}$	$3.37\ 10^{-9}$	$6.35\ 10^{-11}$	$3.72\ 10^{-11}$
Pu-240	$5.40\ 10^{-8}$	$3.32\ 10^{-9}$	$6.26\ 10^{-11}$	$3.67\ 10^{-11}$
Pu-241	$1.10\ 10^{-9}$	$6.80\ 10^{-10}$	$6.84\ 10^{-11}$	$1.37\ 10^{-12}$
Pu-242	$5.29\ 10^{-8}$	$3.26\ 10^{-9}$	$6.13\ 10^{-11}$	$3.59\ 10^{-11}$
Pu-244	$5.43\ 10^{-8}$	$3.34\ 10^{-9}$	$6.30\ 10^{-11}$	$3.69\ 10^{-11}$
Am-241	$3.20\ 10^{-8}$	$1.97\ 10^{-8}$	$1.98\ 10^{-9}$	$3.96\ 10^{-11}$
Am-242m	$4.34\ 10^{-8}$	$1.37\ 10^{-8}$	$1.26\ 10^{-9}$	$4.06\ 10^{-11}$
Am-243	$4.38\ 10^{-8}$	$2.68\ 10^{-8}$	$2.69\ 10^{-9}$	$5.40\ 10^{-11}$
Cm-242	$5.34\ 10^{-11}$	$3.28\ 10^{-12}$	$6.18\ 10^{-14}$	$3.62\ 10^{-14}$
Cm-243	$3.49\ 10^{-10}$	$1.78\ 10^{-10}$	$1.83\ 10^{-12}$	$3.95\ 10^{-13}$
Cm-244	$1.62\ 10^{-10}$	$1.69\ 10^{-11}$	$2.51\ 10^{-13}$	$1.17\ 10^{-13}$
Cm-245	$4.66\ 10^{-8}$	$2.81\ 10^{-8}$	$2.83\ 10^{-10}$	$5.70\ 10^{-11}$
Cm-246	$4.50\ 10^{-8}$	$2.77\ 10^{-8}$	$2.79\ 10^{-10}$	$5.57\ 10^{-11}$
Cm-248	$1.70\ 10^{-7}$	$1.04\ 10^{-7}$	$1.05\ 10^{-9}$	$2.10\ 10^{-10}$

* Emplaced at any depth with other LLW

1029. The doses calculated using illustrative inventories are considered further in Appendix D.

E.6. Heterogeneity of disposed waste

1030. The waste that is expected to be sent to Port Clarence for disposal may not be uniformly contaminated and therefore the radioactivity may be heterogeneously distributed throughout the package or consignment. A series of scenarios has been considered to look at the potential dose that could arise from different types of waste that may be sent to the site for disposal. These assessments are independent of whether disposal occurs to the hazardous or non-hazardous landfill. In this section the disposal of large items, discrete (smaller) items and particles are considered (see Table 152). The assessment calculates the dose received if the scenario were to occur.

Table 152 Summary of radiological assessment scenarios for different waste forms

Scenario	Exposed group
Exposure to heterogeneously contaminated large objects during or following excavation	Worker and Member of public
Exposure to discrete items following erosion	Member of public
Exposure to discrete items following excavation	Member of public
Exposure to particles following erosion	Member of public
Exposure to particles following excavation	Member of public

1031. The range of materials that has been assessed covers large contaminated items, such as concrete blocks with a heterogeneous activity distribution profile, down to small particles. For such heterogeneous wastes, the overall specific activity (activity concentration) may be less than a given value, e.g. 200 Bq g^{-1}, but the activity concentration within certain fractions of the waste may exceed this value, e.g. certain fractions may exceed 200 Bq g^{-1}.

1032. Radioactive particles are small discrete items that could be as small as a grain of sand but contain a high level of activity and could be incorporated in a particular radioactive waste stream or package. The possibility that future intrusion events could lead to unintentional recovery of, and exposure to, these particles is assessed.

E.6.1. Large contaminated items

1033. This section considers the implications of disposing of large contaminated items, such as concrete blocks, with a heterogeneous activity distribution profile. The approach taken is the same as that used for the ENRMF ESC (Eden NE, 2015a).

1034. Concrete slabs or blocks from decommissioning buildings and rubble from demolition of buildings used for the storage or conditioning of radioactive wastes may become contaminated. Such contamination may be restricted to the surface layers of the concrete, but the depth of penetration will depend on the nature of the waste or conditioning process (e.g. wet or dry facilities), the period of time the facility was in use, the building material (and any surface treatment such as painting or other sealants) and the chemical properties of the radionuclide fingerprint.

1035. Characterisation of wastes is always subject to some uncertainty. Wastes can be sampled to obtain an overall averaged activity concentration. To determine activity distributions within heterogeneously contaminated wastes they can be sub-sampled or, for large items, cores can be extracted and the depth of contamination, or depth profiles of contamination, can be determined. However, this can be a laborious and expensive undertaking, and considerable uncertainty may remain if there is spatial as well as penetrative heterogeneity in the activity distribution. Best practice is to remove the contaminated surface layer of the building before demolition and dispose of it separately from the rest of the building material, so avoiding significant inhomogeneity in the waste.

1036. To consider the potential effects of a range of assumptions regarding the distribution of activity within wastes, this assessment considers heterogeneous large items and demolition rubble. The characteristics of the large items and rubble are typical of decommissioning wastes. This scenario is not used to constrain landfill capacity, nor the average activity concentration in the waste.

1037. A number of different typical wastes are considered, including: a hypothetical concrete block contaminated with Cs-137; concrete blocks from decommissioning (with different radionuclide fingerprints); and, rubble and crushed concrete from building demolition (with different radionuclide fingerprints).

Scenario selection

1038. There are four principal scenarios by which activity from disposed waste may reach the accessible environment.

- Dissolution in leachate and transport though groundwater.
- Excavation of wastes and subsequent use for cultivation.
- Drilling through waste and handling retrieved material.
- Exposure of waste and subsequent occupancy.

1039. Dissolution in leachate is addressed in Section E.4.3 and the conservative assumptions in that assessment, regarding leaching through the mass of the waste with no retardation due to waste packaging, will also bound the disposal of heterogeneous wastes. The leachate/groundwater scenario is thus not considered further here.

1040. Excavation of waste and subsequent use of the material for cultivation requires a number of assumptions. The waste must provide a suitable growing medium or physical soil improver. The waste must be of sufficient volume and surface area to provide a credible option for cultivation, or must be mixed in a volume of soil or other material to provide a suitable medium and sufficient volume for cultivation. Where waste is mixed to provide a growing medium it will be the averaged activity concentration that is of relevance, rather than the activity distribution profile within the waste matrix itself (see Sections E.5.6 and E.5.8). Hence the use of the waste for cultivation is not considered further here.

1041. Drilling through waste or exposure of waste (through natural processes of erosion or through deliberate human activity) could lead to higher dose impacts for surface contaminated items compared to uniformly contaminated items. These two scenarios (site investigator and site occupant) are considered further.

1042. A series of boreholes could be drilled across the site in order to characterise the area. One or more such boreholes may penetrate the contaminated items and be retrieved for laboratory analysis. The driller may also handle the retrieved core. Such handling can lead to both an organ dose (skin on the hand) and a whole-body effective dose. In addition, dust from the core may be inhaled and inadvertent ingestion may occur. The principal considerations in determining dose are time spent handling or in proximity to the core and, for determining the whole-body effective dose, the averaged distance from the core. It is assumed that drilling will occur only after the end of the period of authorisation. The assessment assumes that a geotechnical worker examines an intact core for 2 hours.

1043. Natural erosion of the landfill surface will depend on processes described earlier in the ESC (Section 2.9).

1044. Following exposure of the waste, occupancy of the area may lead to external exposure and inhalation of contaminated dust may occur. Inadvertent ingestion is considered less likely in this scenario but is included for completeness.

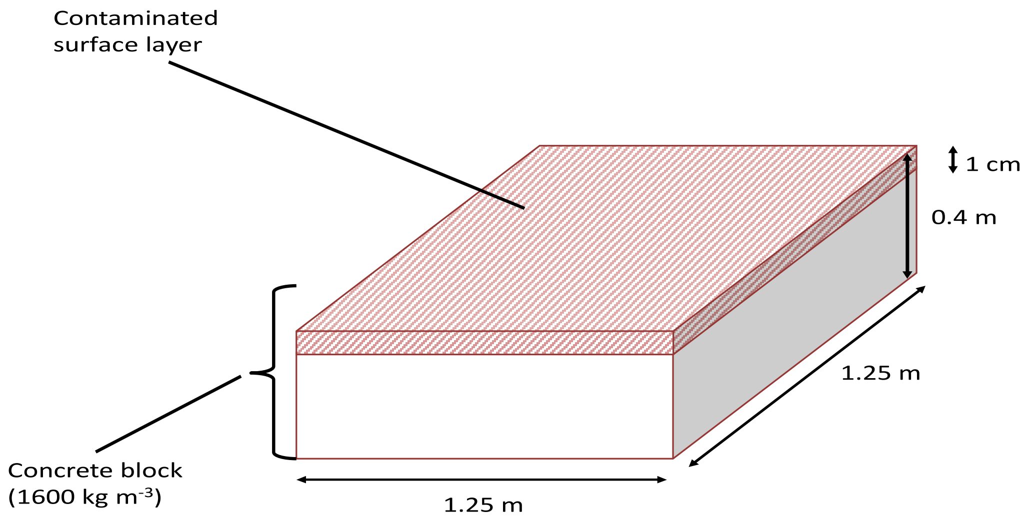
E.6.1.1. Waste characteristics

1045. Six large item waste streams are characterised and detailed below:

- concrete slabs from decommissioning a Fuel Element Debris (FED) storage silo;
- activated concrete shielding blocks;
- building rubble 1: concrete and rubble from building demolition;
- building rubble 2: crushed concrete, soil and rubble from building demolition;
- reinforced concrete from dismantling a research facility; and,
- a hypothetical concrete block contaminated with Cs-137.

1046. **Concrete demolition slabs** – Contaminated concrete slabs from a FED storage facility (Figure 21). The slabs are contaminated with H-3, Sr-90, Cs-137, Pu-239 and Am-241; which collectively account for 98% of all activity present. For simplicity, it is assumed that each named radionuclide accounts for 20% of the total measured activity. An average total activity concentration for the waste is 19 Bq g^{-1}. The concrete blocks are assumed to be 0.4 m deep, with all contamination on one surface only, to a depth of 1 cm. All radionuclides are assumed to have penetrated the concrete block equally to the same depth. The blocks are nominally assumed to measure 1.25 x 1.25 m surface area, but this assumption is relevant only insofar as the surface area is sufficient that a 10 cm diameter core may be drilled wholly through the block. The concrete is assumed to have a density of 1600 kg m^{-3}, the default density for which the external dose coefficients are derived.

Figure 21 Contaminated concrete block



1047. **Activated concrete shielding blocks** - Activated concrete shielding slabs from a research reactor. The slabs contain H-3 (as a contaminant) and the activation products Fe-55, Co-60, Ni-63 and Eu-152. These are present in equal proportions (i.e. they each account for 20% of the total activity) and are uniformly distributed to the same depth in the surface layer of the block. An average total activity concentration for the waste is 7 Bq g^{-1}. As before, the concrete blocks are 1.25 x 1.25 x 0.4 m, with all activity present in the surface 1 cm layer, and the concrete is also assumed to have a density of 1600 kg m^{-3}. All radionuclides are assumed to have penetrated the concrete block equally to the same depth.

1048. **Building rubble 1** - Concrete and rubble contaminated with tritium and C-14. The activity is present in the surface layer of the rubble, but the rubble is received as a mixed consignment. The average activity concentration is 136 Bq g^{-1} of which 99% is H-3. The rubble is assumed to have a density of 1600 kg m^{-3}.

1049. **Building rubble 2** - Concrete, soil and rubble from the demolition of a post-irradiation examination facility. The waste contains Co-60, Ni-63, Sr-90, Cs-137, Pu-241 and Am-241 in equal proportions. The activity is present in the surface layer of the rubble, but the rubble is received as a mixed consignment. The average activity concentration is 8 Bq g^{-1}. The rubble is assumed to have a density of 1600 kg m^{-3}.

1050. **Reinforced concrete** - Reinforced concrete blocks from dismantling a research facility. The blocks contain H-3 (11% of all activity), C-14 (1% of all activity) and Cs-

137 (88% of all activity). The activity is present in the surface 1 cm layer of the block. An average total activity concentration for the waste is 153 Bq g^{-1}. As before, the concrete blocks are 1.25 x 1.25 x 0.4 m, and the concrete is assumed to have a density of 1600 kg m^{-3}. All radionuclides are assumed to have penetrated the concrete block equally to the same depth.

1051. **Hypothetical concrete block** - A large concrete block 0.4 m deep, contaminated with Cs-137 and with all contamination on one surface only. The blocks are nominally assumed to measure 1.25 x 1.25 m surface area, and to have a density of 1600 kg m^{-3}, the default density for which the external dose coefficients are derived. The average activity concentration is 200 Bq g^{-1} and all of the activity is present in the surface layer.

1052. The primary parameters that may be subject to uncertainty are the exposure time (hr y^{-1}), the time at which exposure occurs (following emplacement of the waste), distance from the waste, breathing and ingestion rates, depth of contamination, incident angle of the exposed waste and density of the waste. These aspects are considered in presenting the results of the dose calculations for the hypothetical concrete block. Sensitivity to assumed depth profiles for distribution of activity is explored in Section E.7.3.

E.6.1.2. Assessment calculation for large contaminated items

Site occupant

1053. It is assumed that the surface layer of the disposal site is removed and the waste exposed. It is further assumed that a sufficient area is exposed such that the external dose rate can be approximated as a semi-infinite slab. It is also assumed that the site occupant breathes in, and inadvertently ingests, contaminated dust arising from drilling through the item.

1054. The dose to a site occupant can then be calculated as:

$$Dose_{occupier} = (D_{irr}^{Rn} \cdot \mathrm{T} \cdot A_{Rn}(t) + (D_{inh}^{Rn} \cdot \mathrm{T} \cdot \mathrm{B} \cdot \mathrm{M}_{inh} \cdot C_w(t)) + (D_{ing}^{Rn} \cdot \mathrm{T} \cdot \mathrm{B} \cdot \mathrm{M}_{ing} \cdot C_w(t))$$

where:

- D_{irr} — is the external semi-infinite slab irradiation dose coefficient for radionuclide Rn (mSv y^{-1} per Bq kg^{-1}), see Table 200;
- A_{Rn} — is the activity of the contamination (MBq)
- d — is the distance of the person from the source (m);
- M_{inh} — is the dust load of contaminated waste inhaled by the site occupant (kg m^{-3});
- M_{ing} — is the rate of ingestion of dust from the material (kg h^{-1});
- T — is the time the person is exposed to the material (h);
- B — is the breathing rate (m^3 h^{-1});

- D_{inh} and D_{ing} are the dose coefficients for radionuclide *Rn* (Sv Bq^{-1} and Sv Bq^{-1} respectively); and,
- $C_w(t)$ is the activity concentration of radionuclide *Rn* (Bq kg^{-1}) in the material at the time of intrusion, *t.*

1055. It is assumed that a person occupies the site for 1 hour per week (i.e. 52 hours per year). All activity in the contaminated waste is assumed to be in the surface 1 cm.

1056. The whole-body effective dose is determined assuming that the person is, on average, 2 m from the item of waste. Dust in air from the core is assumed to be present at 1 10^{-7} kg m^{-3}. These and other assumptions are tabulated below (Table 153).

Table 153 Parameters for site occupant

Parameter	Units	Value	Description
d	m	2	Distance of the site occupant from the point source
M_{inh}	kg m^{-3}	5.80 10^{-8}	dust load of contaminated waste
M_{ing}	kg hr^{-1}	3.42 10^{-6}	rate of ingestion of dust
T	hr y^{-1}	52	exposure time
B	m^3 hr^{-1}	1	breathing rate

1057. It is assumed that the site occupant is exposed 60 years after emplacement (the end of the period of authorization) as a result of an intrusion event. Calculations are also presented for exposure after erosion has uncovered the waste. Erosion is not expected to happen at the site. Scenarios determining the exposure of a coastal walker following erosion of the site assumed that erosion occurs 2540 years after closure. A nominal value of 2000 years post emplacement is used for the assessment of large contaminated items.

1058. Note, in this case the inadvertent rate of ingestion is an order of magnitude lower than assumed for the site driller. The adopted dust load is higher than in previous scenarios presenting the maximum for a single event rather than an average. The breathing rate is also somewhat lower, consistent with a more sedentary aspect. These parameter values were also used in ENRMF ESC (Eden NE, 2015a).

Site investigator (driller)

1059. The dose to a site investigator assumes that a drill core has a diameter of 10 cm. The depth of the core is assumed to be sufficient to penetrate through the waste and the incident angle of penetration is such that the surface contaminated layer is removed. The core is then sectioned so as to expose the contaminated surface area.

1060. The dose to a driller can then be calculated as:

$$Dose_{excavator} = \frac{(G_{irr}^{Rn} \cdot \mathrm{T} \cdot A_{Rn}(t)}{d^2} + (D_{inh}^{Rn} \cdot \mathrm{T} \cdot \mathrm{B} \cdot \mathrm{M}_{inh} \cdot C_w(t)) + (D_{ing}^{Rn} \cdot \mathrm{T} \cdot \mathrm{B} \cdot \mathrm{M}_{ing} \cdot C_w(t)$$

where parameters are as described in the previous equation except:

- G_{irr} is the point-source dose rate for radionuclide *Rn* at 1 m from a 1 MBq source (mSv hour^{-1} MBq^{-1}).

1061. It is assumed that a driller spends 2 hours examining a core and that all activity in the contaminated item is in the surface 1 cm. The whole-body effective dose is determined assuming that the worker is, on average, 1 m from the core. Dust in air from the core during the examination is assumed to be present at 6 10^{-7} kg m^{-3}. These and other assumptions are tabulated below (Table 154).

Table 154 Parameters for a site investigator (driller)

Parameter	Units	Value	Description
d	m	1	Distance of the driller from the point source
M_{inh}	kg m^{-3}	6.00E-07	dust load of contaminated waste
M_{ing}	kg hr^{-1}	3.42E-05	rate of ingestion of dust
T	hr y^{-1}	2	exposure time
B	m^3 hr^{-1}	1.2	breathing rate

1062. It is assumed conservatively that intrusion occurs 60 years from emplacement of the waste.

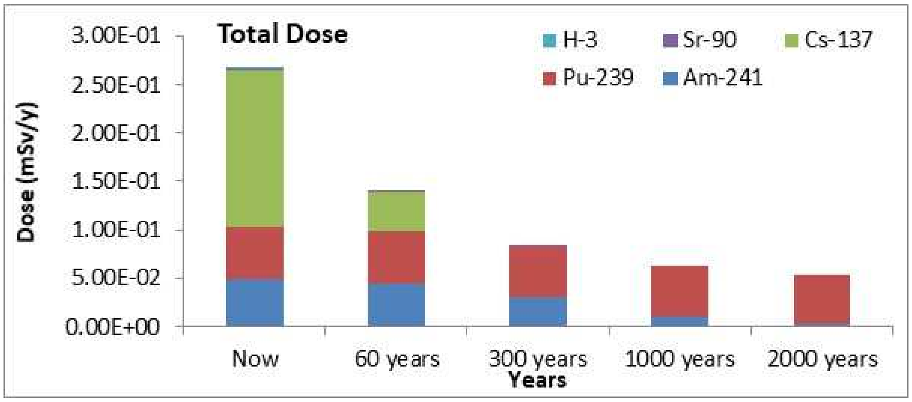
E.6.1.3. Dose from large contaminated waste items

1063. The doses to the site occupant and to the site investigator from each of the five characteristic waste types are presented in this section.

Concrete demolition slabs - dose to site occupant

1064. It is assumed that concrete demolition slabs (from a FED storage facility) are disposed of with an average activity of 19 Bq g^{-1} comprising H-3, Sr-90, Cs-137, Pu-239 and Am-241. If all of the activity is assumed to be in the surface 1 cm layer the total activity concentration in that layer is 760 Bq g^{-1}. The time dependence of the dose to a site occupant is shown in Figure 22. Although the reference time for a site occupant is 150 years or 200 years after site closure, results are also given for 60 years after emplacement, the end of the period of authorisation. The dose to a site occupant, at 60 years after emplacement of the waste, is 0.14 mSv y^{-1} (140 µSv y^{-1}), assuming 52 hours per year exposure and the dose at later times is lower. The dose is initially dominated by Cs-137 but at 60 years or later, the dose is dominated by the long-lived actinides, Pu-239 and Am-241 (see Figure 22). Exposure before 60 years is not credible as it is within the period of authorisation.

Figure 22 Time-dependant dose to site occupant from contaminated concrete demolition slabs



1065. The dose at 60 years is roughly evenly split between the external, inhalation and ingestion contributions. The external dose component is dominated by Cs-137. The ingestion and inhalation dose components are dominated by Pu-239 and Am-241 (see Table 155). The dose at 150 years would be dominated by ingestion and inhalation of Pu-239 and Am-241.

Table 155 Pathway-dependant dose to site occupant from contaminated concrete demolition slabs at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Am-241	4.76 10^{-3}	3.45 10^{-2}	4.91 10^{-3}	4.41 10^{-2}
Pu-239	2.55 10^{-5}	4.73 10^{-2}	6.75 10^{-3}	5.41 10^{-2}
Cs-137	4.07 10^{-2}	3.88 10^{-6}	8.86 10^{-5}	4.08 10^{-2}
Sr-90	3.47 10^{-4}	1.51 10^{-5}	1.96 10^{-4}	5.58 10^{-4}
H-3	0	3.52 10^{-9}	1.66 10^{-8}	2.02 10^{-8}
Total	4.59 10^{-2}	8.18 10^{-2}	1.19 10^{-2}	1.40 10^{-1}

1066. For human intrusion situations, the dose at 60 years or later should be compared to the human intrusion dose guidance values of 3-20 mSv (with the lower value being applicable for doses that may occur over extended periods). The doses are all below the lower guidance level. Considering exposure of the waste through natural processes, the risk guidance level is relevant. Extrapolating the dose out to 2000 years (a hypothetical earliest date at which 'natural' erosion could expose the waste) gives a dose estimate of 0.05 mSv y^{-1}, dominated by the ingestion and inhalation of dust containing Pu-239. This dose is equivalent to an annual risk of around 3.0 10^{-6}. Given the grossly conservative nature of the assumptions, it is considered that this risk is broadly consistent with the risk guidance criterion of 10^{-6} for the post-closure period.

Concrete demolition slabs - dose to Site Investigator

1067. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is 0.0159 mSv (15 µSv) per core handled (Figure 23). The dose is dominated by inhalation and unintentional ingestion of Pu-239 and Am-241 (see Table 156).

Figure 23 Time-dependant dose to site investigator from contaminated concrete demolition slabs

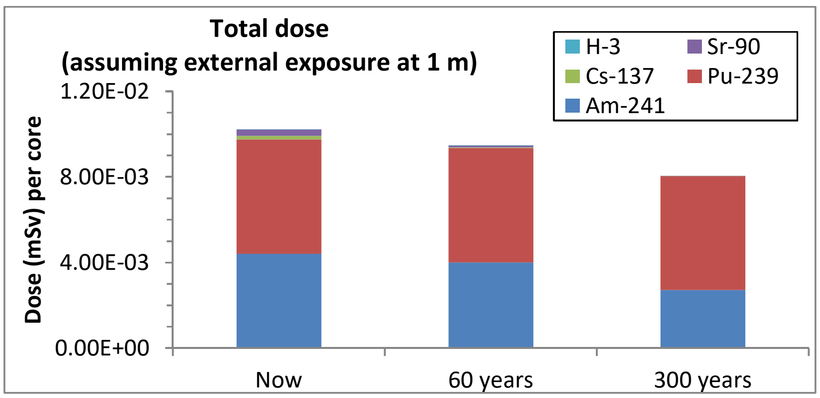


Table 156 Pathway-dependant dose to site investigator from contaminated concrete demolition slabs at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Am-241	$7.42\ 10^{-8}$	$4.77\ 10^{-3}$	$1.89\ 10^{-3}$	$6.66\ 10^{-3}$
Pu-239	$6.21\ 10^{-10}$	$6.56\ 10^{-3}$	$2.59\ 10^{-3}$	$9.15\ 10^{-3}$
Cs-137	$6.39\ 10^{-7}$	$5.38\ 10^{-7}$	$3.41\ 10^{-5}$	$3.52\ 10^{-5}$
Sr-90	$3.44\ 10^{-7}$	$2.08\ 10^{-6}$	$7.53\ 10^{-5}$	$7.77\ 10^{-5}$
H-3	0	$4.87\ 10^{-10}$	$6.40\ 10^{-9}$	$6.89\ 10^{-9}$
Total	$1.06\ 10^{-6}$	$1.13\ 10^{-2}$	$4.59\ 10^{-3}$	$1.59\ 10^{-2}$

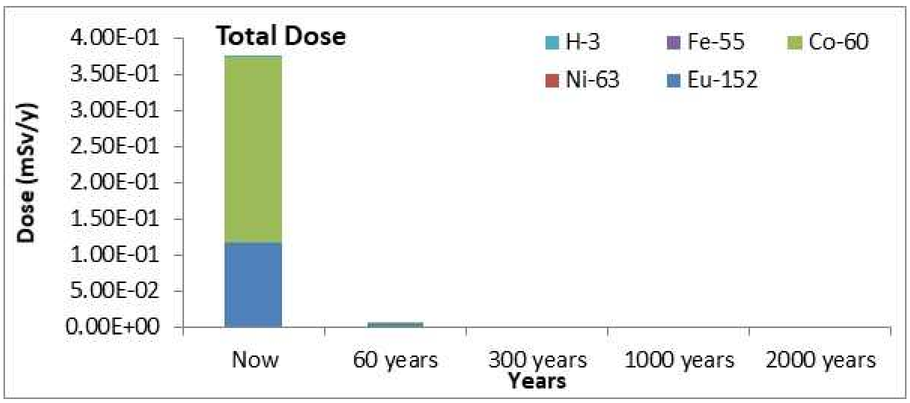
1068. The dose incurred at 60 years from emplacement of waste is low. Even if 10 cores were handled, all with similar characteristics, the dose would remain more than a factor of 18 below the lower dose guidance level of 3 mSv for human intrusion scenarios. The potential dose to skin from close handling of a core, assuming an average distance of 0.05 m (5 cm), is around 0.0004 mSv (0.4 µSv) per core at 60 years. This is well below skin organ dose limit for members of the public of 50 mSv y^{-1}.

Activated concrete shielding blocks - dose to Site Occupant

1069. It is assumed that activated concrete shielding blocks are disposed of with an average activity of 7 Bq g^{-1} comprising H-3, Fe-55, Co-60, Ni-63 and Eu-152. If all of the activity is assumed to be in the surface 1 cm layer the total activity concentration in that layer is 280 Bq g^{-1}.

1070. Although the reference time for a site occupant is 150 years or 200 years after site closure, results are also given for 60 years after emplacement, the end of the period of authorisation. The dose to a site occupant, at 60 years after emplacement of the waste, assuming 52 hours per year exposure, is 0.0056 mSv y^{-1}. The dose is initially dominated by Co-60 and Eu-152 (Figure 24) but the very short half-life of all of the radionuclides within the shielding blocks (Ni-63 has the longest half-life at ca. 100 years, Fe-55 and Co-60 have half-lives of ca. 2.7 and 5.3 years respectively) is such that by 60 years the dose is very low.

Figure 24 Time-dependant dose to site occupant from activated concrete shielding blocks



1071. The dose at 60 years is dominated by the external component from Eu-152 (Table 1570). The dose at later times is negligible.

Table 157 Pathway-dependant dose to site occupant from activated concrete shielding blocks

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Eu-152	$5.47\ 10^{-3}$	$2.84\ 10^{-7}$	$6.47\ 10^{-7}$	$5.47\ 10^{-3}$
Ni-63	0	$1.25\ 10^{-7}$	$9.86\ 10^{-7}$	$1.11\ 10^{-6}$
Co-60	$9.56\ 10^{-5}$	$1.69\ 10^{-9}$	$1.27\ 10^{-8}$	$9.56\ 10^{-5}$
Fe-55	0	$2.83\ 10^{-14}$	$8.30\ 10^{-13}$	$8.58\ 10^{-13}$
H-3	0	$1.30\ 10^{-9}$	$6.13\ 10^{-9}$	$7.43\ 10^{-9}$
Total	$5.56\ 10^{-3}$	$4.12\ 10^{-7}$	$1.65\ 10^{-6}$	$5.57\ 10^{-3}$

Activated concrete shielding blocks - dose to Site Investigator

1072. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is 0.0004 mSv (0.4 µSv) per core handled (Figure 25). This very low dose is attributable largely to the short half-lives of all of the radionuclides present. The dose incurred is dominated by external exposure from the small inventory of Co-60 remaining at 60 years (Table 158).

Figure 25 Time-dependant dose to site investigator from activated shielding blocks

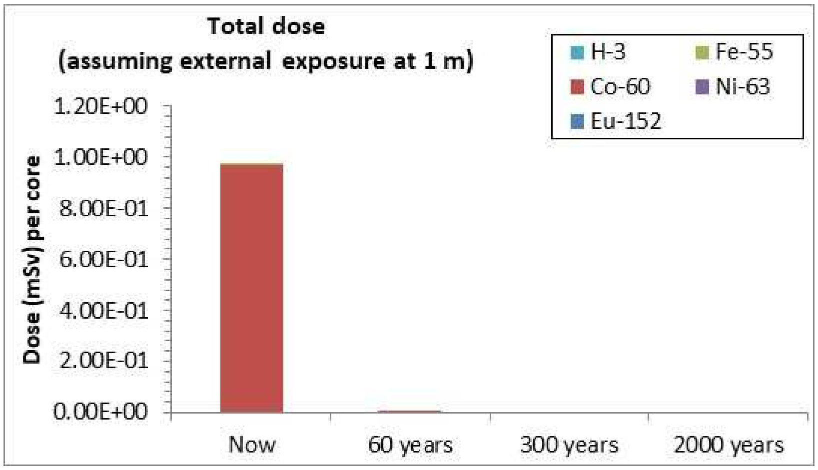


Table 158 Pathway-dependant dose to site investigator from contaminated concrete demolition slabs at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Eu-152	7.50 10^{-8}	3.93 10^{-8}	2.49 10^{-7}	3.63 10^{-7}
Ni-63	0	1.73 10^{-8}	3.79 10^{-7}	3.97 10^{-7}
Co-60	3.64 10^{-4}	2.34 10^{-10}	4.88 10^{-9}	3.64 10^{-4}
Fe-55	5.36 10^{-16}	3.92 10^{-15}	3.19 10^{-13}	3.24 10^{-13}
H-3	0	1.79 10^{-10}	2.36 10^{-9}	2.54 10^{-9}
Total	3.64 10^{-4}	5.70 10^{-8}	6.35 10^{-7}	3.65 10^{-4}

1073. The dose incurred at 60 years from emplacement of waste is very low and many orders of magnitude below the lower dose guidance level of 3 mSv for human intrusion scenarios.

1074. The potential dose to skin from close handling of a core, assuming an average distance of 0.05 m (5 cm), is around 0.15 mSv (150 µSv) per core at 60 years. Comparison may be made with a skin organ dose limit for members of the public of 50 mSv y^{-1}.

Building rubble 1 - dose to Site Occupant

1075. The dose to a site occupant from building rubble with an average activity of 136 Bq g^{-1}, comprising H-3 (99%) and C-14 (1%), at 60 years after emplacement of the waste, is 0.0002 µSv y^{-1}, assuming 52 hours per year exposure. In this case, the rubble is assumed to be well mixed as there is no credible mechanism for a contaminated surface layer to be exposed uniformly following disposal.

1076. Although the reference time for a site occupant is 150 years or 200 years after site closure, results were produced 60 years after emplacement, the end of the period of authorisation. The dose is dominated by the small C-14 inventory (ca. 1.4 Bq g^{-1}) as the very short half-life of H-3 (12.3 years) means that it has decayed to very low levels by 60 years. More than 75% of the dose is contributed from unintentional ingestion of contaminated dust.

Building rubble 1 - dose to Site Investigator

1077. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is about 0.003 µSv per core handled and arises exclusively from the ingestion / inhalation pathways.

Building rubble 2 - dose to Site Occupant

1078. Although the reference time for a site occupant is 150 years or 200 years after site closure, results are also given for 60 years after emplacement, the end of the period of authorisation. The dose to a site occupant from crushed concrete with an average activity of 8 Bq g^{-1}, comprising Co-60, Ni-63, Sr-90, Cs-137, Pu-241 and Am-241, at 60 years after emplacement of the waste, is 2.1 µSv y^{-1}, assuming 52 hours per year exposure. In this case, the concrete is assumed to be well mixed. As before, even if the concrete initially had all contamination present in the surface layer, once it is crushed there is no credible mechanism to expose only the contaminated material.

1079. The dose is initially dominated by Co-60 and Cs-137 (both through the external exposure) but by 60 years the dose contribution from Co-60 is negligible and the dose is dominated by Am-241 (inhalation) and Cs-137 (external), see Figure 26 and Table 159.

Figure 26 Time-dependant dose to site occupant from crushed concrete

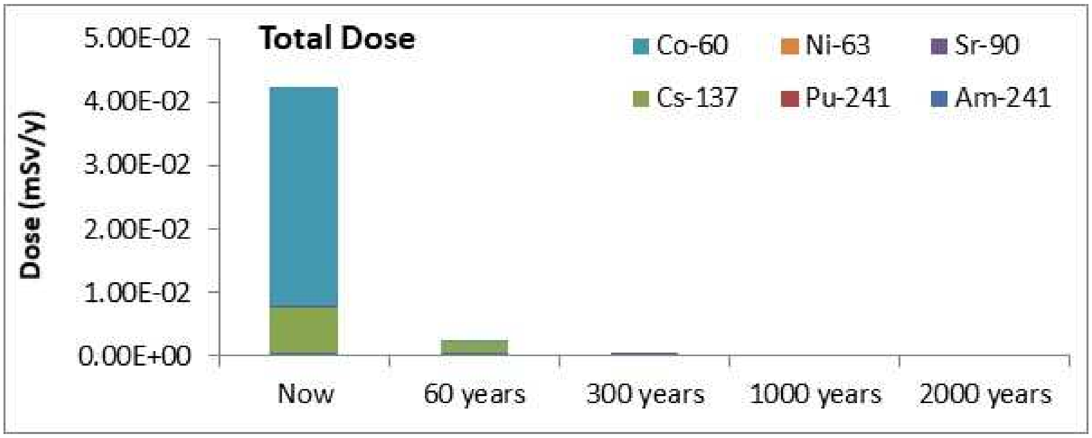


Table 159 Pathway-dependant dose to site occupant from crushed concrete

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Am-241	8.49 10^{-5}	3.02 10^{-4}	4.31 10^{-5}	4.30 10^{-4}
Pu-241	6.96 10^{-10}	4.40 10^{-7}	6.28 10^{-8}	5.03 10^{-7}
Cs-137	1.83 10^{-3}	3.41 10^{-8}	7.77 10^{-7}	1.84 10^{-3}
Sr-90	1.24 10^{-5}	1.32 10^{-7}	1.72 10^{-6}	1.43 10^{-5}
Ni-63	0	2.97 10^{-9}	2.35 10^{-8}	2.65 10^{-8}
Co-60	1.30 10^{-5}	4.03 10^{-11}	3.02 10^{-10}	1.30 10^{-5}
Total	1.94 10^{-3}	3.03 10^{-4}	4.57 10^{-5}	2.29 10^{-3}

Building rubble 2 - dose to Site Investigator

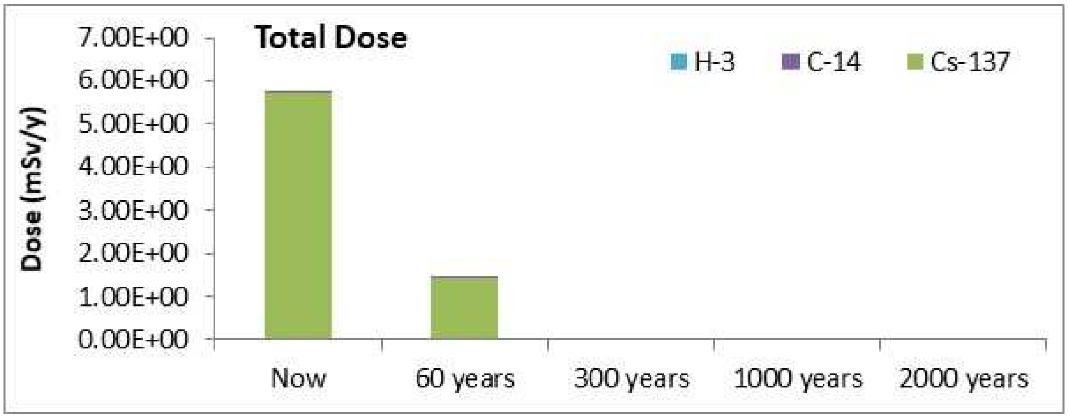
1080. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is ca. 2.5 µSv per core handled and is dominated by the presence of Am-241 in the ingestion / inhalation pathways.

Reinforced concrete - dose to Site Occupant

1081. It is assumed that contaminated concrete slabs are received with an average activity of 153 Bq g^{-1} comprising H-3 (11%), C-14 (1%) and Cs-137 (88%). If all of the activity is assumed to be in the surface 1 cm layer the total activity concentration in that layer is 6120 Bq g^{-1}.

1082. Although the reference time for a site occupant is 150 years or 200 years after site closure, results are also given for 60 years after emplacement, the end of the period of authorisation. The dose to a site occupant, at 60 years after emplacement of the waste, assuming 52 hours per year exposure, is 1.45 mSv y^{-1} (Figure 27).

Figure 27 Time-dependant dose to site occupant from contaminated reinforced concrete



1083. The dose is dominated by external exposure from Cs-137 (Table 160).

Table 160 Pathway-dependant dose to site occupant from contaminated reinforced concrete

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Cs-137	1.44 10^{0}	1.38 10^{-4}	3.14 10^{-3}	1.45 10^{0}
C-14	7.82 10^{-7}	9.16 10^{-7}	6.27 10^{-6}	7.97 10^{-6}
H-3	0	1.56 10^{-8}	7.37 10^{-8}	8.93 10^{-8}
Total	1.44 10^{0}	1.39 10^{-4}	3.14 10^{-3}	1.45 10^{0}

Reinforced concrete - dose to Site Investigator

1084. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is 0.0013 mSv (1.3 µSv) per core handled (0). The dose is dominated by unintentional ingestion of Cs-137 (Table 161).

Figure 28 Time-dependant dose to site investigator from contaminated reinforced concrete

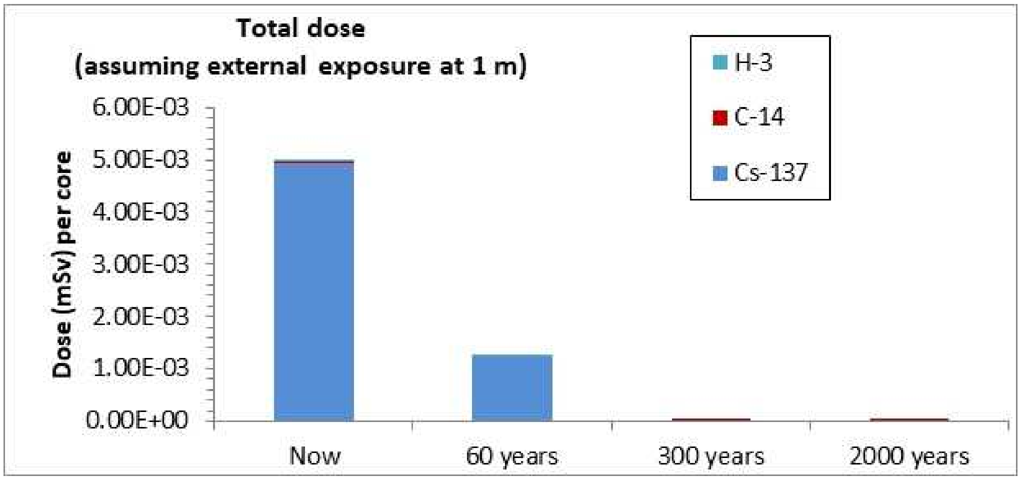


Table 161 Pathway-dependant dose to site investigator from contaminated reinforced concrete at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Cs-137	$2.27\ 10^{-5}$	$1.91\ 10^{-5}$	$1.21\ 10^{-3}$	$1.25\ 10^{-3}$
C-14	0	$1.27\ 10^{-7}$	$2.41\ 10^{-6}$	$2.54\ 10^{-6}$
H-3	0	$2.16\ 10^{-9}$	$2.84\ 10^{-8}$	$3.05\ 10^{-8}$
Total	$2.27\ 10^{-5}$	$1.92\ 10^{-5}$	$1.21\ 10^{-3}$	$1.25\ 10^{-3}$

1085. The dose incurred at 60 years from emplacement of waste is very low and many orders of magnitude below the lower dose guidance level of 3 mSv for human intrusion scenarios.

1086. The potential dose to skin from close handling of a core, assuming an average distance of 0.05 m (5 cm), is around 0.009 mSv (9 µSv) per core at 60 years. Comparison may be made with a skin organ dose limit for members of the public of 50 mSv y^{-1}.

Hypothetical concrete block - dose to Site Occupant

1087. In this hypothetical case, contamination is present at an average activity of 200 Bq g^{-1} but is in the surface 1 cm, where the activity concentration rises to 8000 Bq g^{-1}. The activity is assumed to be present as Cs-137.

1088. Although the reference time for a site occupant is 150 years or 200 years after site closure, results are also given for 60 years after emplacement, the end of the period of authorisation. The dose to a site occupant, 60 years after emplacement of the waste, is 2.25 mSv y^{-1}, assuming 52 hours per year exposure. The dose arises mainly from external exposure, accounting for more than 99% of the total dose.

Hypothetical concrete block - dose to Site Investigator

1089. The equivalent dose to a site investigator (driller), assuming an average distance from a point source of 1 m, is 1.9 µSv per core handled.

1090. A skin dose, assuming handling of the core at an average distance of 0.05 m, is 0.015 mSv (15 µSv) per core.

E.6.1.4. Discussion

1091. The doses to the site occupant at 60 years are compared to the human intrusion dose guidance values of 3 to 20 mSv (with the lower value being applicable for doses that may occur over extended periods). The doses from all the items considered here are all well below this.

1092. Extrapolating the dose out to 2,000 years (a hypothetical date for 'natural' erosion exposing the waste that is used to illustrate impact) gives a dose estimate to the site occupant 0.05 mSv y^{-1} (dominated by the ingestion and inhalation of dust containing Pu-239 for the particular waste item). This dose is equivalent to an annual risk of around 3 10^{-6}. Given the grossly conservative nature of the assumptions, it is considered that this risk is broadly consistent with the risk guidance criterion of 10^{-6} for the post-closure period.

1093. Hence, an assessment of the doses arising from exposure to typical decommissioning wastes that may contain large heterogeneously contaminated items has shown that they will comply with the dose and risk criteria in the NS-GRA (UK Environment Agencies, 2009).

E.6.2. Excavation of discrete items

1094. This scenario is included due to the possibility that the site will be eroded by the sea, and walkers along the bank of the estuary near the site may then come into contact with discrete items of waste that have become exposed. Erosion is not expected to happen before the year 4476 at the earliest.

1095. This scenario is not used to constrain landfill capacity. However, it places limits on the radioactivity of specific discrete items within consignments.

1096. LLW Repository Ltd (LLWR Ltd, 2013) define 'discrete items' as "a distinct item of waste that, by its characteristics, is recognisable as unusual or not of natural origin and could be a focus of interest, out of curiosity or potential for recovery and recycling/re-use of materials should the waste item be exposed after repository closure." This definition is adopted in this assessment.

1097. Examples of discrete items given by LLWR (LLWR Ltd, 2013) are hand tools, engineered items and equipment of durable materials (such as may be disposed with other wastes in drums for grouting or high-force compaction, or directly to a Disposal Container); grouted drums or pucks from high-force compaction; and large metal items, e.g. steel beams and plates, pipework, shielding, heavy equipment and flasks (but not general scrap metal) such as may be disposed directly to a Disposal Container.

1098. A discrete item has the potential to modify the behaviour of a person that encounters it, i.e. it is visible and therefore an individual may deliberately go towards and inspect, or (if small enough) pick up the item. This is different from the standard assessment calculations in which the estuary bank user carries out activities on the bank of the estuary without regard to the presence of the waste or the radioactive hazard it may pose. Thus, two situations can be envisaged: a casual encounter with a single item and a situation where a person deliberately seeks out, collects, takes away or disrupts discrete items. The future behaviours of people that might lead to them encountering radioactive discrete items uncovered by natural disruptive processes cannot be predicted, and so the probability of exposure cannot be quantified. In this respect, the exposure situation is similar to that of inadvertent human intrusion. Exposure to discrete items exposed by natural processes is specifically addressed in Requirement R12 of the Environment agencies GRR ((Environment Agencies, 2018)), which specifies that the results of illustrative calculations are compared with the dose guidance level for inadvertent human intrusion (3 mSv to 20 mSv); however this guidance relates to the clean-up of nuclear licensed sites, and does not apply to waste disposal sites.

1099. It is very difficult to estimate the exposure time of a person who deliberately seeks out, collects, takes away or disrupts discrete items. However, the lower dose guideline level for inadvertent intrusion is at least two orders of magnitude greater than the effective dose of 20 µSv y^{-1}, which corresponds to the risk guidance level specified in the NS-GRA, assuming a probability of unity. Thus, comparison of the dose from a casual encounter with a single item for 1 hour with a dose criterion of 20 µSv y^{-1} will ensure that the exposure to a person who deliberately seeks out, collects, takes away or disrupts discrete items will meet the lower dose guideline value of 3 mSv.

1100. As such, the case of a casual encounter for 1 hour assessed against a 20 µSv y^{-1} value is expected to be limiting. This conservative approach has been taken in this assessment.

E.6.2.1. Potentially Exposed Group

1101. The assessment of doses resulting from waste that becomes exposed on the estuary bank is based on work undertaken to assess discrete items at the LLWR (LLWR Ltd, 2013). Members of the exposed group are assumed to be adults and to be exposed as a result of external irradiation from contaminated objects and through the inadvertent ingestion of contaminated dust. Although different encounter characteristics could be postulated for children and infants, leading to different effective doses, the encounter by an adult is deemed the appropriate basis for deriving Discrete Item Limits, and is consistent with the approach taken by LLWR.

1102. The approach used by LLW Repository Ltd is followed. A Discrete Item Limit is suggested based on total activity for items with a mass of 1 kg or less and 100 kg or greater, with a transitional function between these total activity limits.

1103. Doses are calculated assuming that items can be represented as sphere of equivalent mass. For each size of sphere, doses from two cases are calculated:

1104. Activity is uniformly distributed over the surface of the sphere.

1105. Activity is distributed uniformly throughout the sphere volume.

1106. These two cases are expected to bound the actual distribution of activity on an item.

1107. All spheres are considered to have a reference density of 2000 kg m^{-3}.

1108. Similar to (LLWR Ltd, 2013), the inhalation pathway is not assessed on the basis that the ingestion pathway will be the dominant intake pathway for most radionuclides assuming the local dust in air carries the same average specific activity as the local substrates and that the discrete items will have a negligible contribution to the local respirable dust.

E.6.2.2. Estimating doses to a local estuary bank user

1109. It is assumed that adults may have an encounter with exposed wastes for a one hour duration, during which time they may closely inspect the item. As such, doses from external irradiation are evaluated at 0.3 m from the surface of the items. This distance is commensurate with being very close to a large item or handling a small item. The same approach was used by LLW Repository Ltd (LLWR Ltd, 2013).

1110. Effective doses from external irradiation are calculated by applying scaling factors to dose coefficients for a uniformly contaminated infinite plane, in the case of a surface contaminated sphere, and dose coefficients for a uniformly contaminated semi-infinite slab, in the case of a volume contaminated sphere.

1111. The dose rate at 1 m from a uniformly contaminated infinite plane is assumed to be equal to the dose rate at 0.3 m from a surface-contaminated sphere of radius 2 m (LLWR Ltd, 2013). The dose rate at 0.3 m from a smaller radius sphere is then obtained by appropriate scaling.

1112. The dose rate at 1 m from a uniformly contaminated semi-infinite slab is assumed to be equal to the dose rate at 0.3 m from a volume-contaminated sphere of radius 2 m (Thorne, 2010), (LLWR Ltd, 2013). The dose rate at 0.3 m from a smaller radius sphere is then obtained by appropriate scaling.

1113. For surface-contaminated items, a secondary ingestion coefficient of 10^{-6} m^2 h^{-1} is assumed. This secondary ingestion coefficient is based on a secondary ingestion coefficient of 10^{-4} m^2 h^{-1} for removable contamination, but assumes that only 1% of the surface activity present at the time of disposal is removable on contact following erosion.

1114. For volume-contaminated items, an inadvertent ingestion coefficient of 0.5 mg h^{-1} is assumed. The value of this coefficient is an order of magnitude less than the standard rate of inadvertent ingestion that would be applied on the basis that not all of the dust or dirt that will be ingested at the time of exposure will come from the discrete item. Such an approach is consistent with the methodology adopted by LLW Repository Ltd.

1115. The site is not expected to be eroded until the year 4476, approximately 2400 years after the expected site closure date. The effective doses from external radiation and committed effective doses from ingestion are calculated at 2400 years after consignment. Hence, the calculated Discrete Item Limits will be conservative because some consignments will be disposed of decades before site closure.

1116. An activity of 1 GBq is assumed to be present on the item when undertaking this assessment. Our assessment focuses on a 1 tonne sphere, but the activity required on an item to give rise to a 20 µSv dose for items of mass 10 g, 100 g, 1 kg, 10 kg, 100 kg and 10 tonnes are also assessed. In reality, discrete items with a mass greater than 10 tonnes would not be expected to be consigned to Port Clarence.

1117. An activity of 1 GBq on a one tonne sphere gives a specific activity concentration of 1 GBq te^{-1} (1000 Bq g^{-1}). The waste acceptance criteria will specify the specific activity concentration limits for a consignment and for a package. We are proposing nuclide dependent specific activity concentration limits. Our assessment is based on items that are at the upper end of acceptability at Port Clarence for radionuclides for which the waste acceptance criteria specify that an individual waste package containing that radionuclide should not exceed an activity concentration of 1000 Bq g^{-1}.

E.6.2.3. Assessment calculation for discrete items

1118. For a radionuclide with no decay chain, the dose arising from the inadvertent ingestion of contaminated material from a surface-contaminated item is given by:

$$Dose_{Surf,ing} = \frac{I \cdot e^{-\lambda t} \cdot SI_{ing} \cdot D_{ing}^{Rn} \cdot t_{ext}}{A}$$

where:

- I is the inventory of radionuclide *Rn* assumed on the object (Bq);
- λ is the decay constant ($year^{-1}$);
- t is the time at which the dose is assessed (years after consignment);
- t_{ext} is the exposure time (h);
- SI_{ing} is the secondary ingestion coefficient ($m^2\ h^{-1}$);
- D_{ing}^{Rn} is the ingestion dose coefficient for radionuclide *Rn* (Sv Bq^{-1}); and,

- A is the surface area of the sphere assumed to represent the surface-contaminated item (m^2).

1119. For a radionuclide with no decay chain, the dose arising from the inadvertent ingestion of contaminated material from a volume-contaminated item is given by:

$$Dose_{Vol,ing} = \frac{I \cdot e^{-\lambda t} \cdot I_{ing} \cdot D_{ing}^{Rn} \cdot t_{ext}}{M}$$

where:

- I is the inventory of radionuclide *Rn* assumed in the object (Bq);
- λ is the decay constant ($year^{-1}$);
- t is the time at which the dose is assessed (years after consignment);
- t_{ext} is the exposure time (h);
- I_{ing} is the inadvertent ingestion rate of contaminated material ($kg\ h^{-1}$);
- D_{ing}^{Rn} is the ingestion dose coefficient for radionuclide *Rn* ($Sv\ Bq^{-1}$); and,
- M is the mass of the volume-contaminated item (kg).

1120. For a radionuclide with no decay chain, the dose arising from external irradiation from a surface-contaminated item is given by:

$$Dose_{Surf,ext} = \frac{I \cdot e^{-\lambda t} \cdot t_{ext} \cdot SF_{Surf} \cdot D_{ext,Surf}^{Rn} \cdot t_{ext}}{A}$$

where:

- I is the inventory of radionuclide *Rn* assumed on the object (Bq);
- λ is the decay constant ($years^{-1}$);
- t is the time at which the dose is assessed (years etc);
- t_{ext} is the exposure time (h);
- SF_{Surf} is the scaling factor from an infinite plane source to a non-sorbing surface-contaminated spherical source at a distance of 0.3 m from the surface of the sphere (dimensionless, see Table 163);
- $D_{ext,Surf}^{Rn}$ is the dose coefficient from an infinite plane source for radionuclide *Rn* ($Sv\ y^{-1}$ per $Bq\ m^{-2}$; see Section E.9.3); and,
- A is the surface area of the sphere assumed to represent the surface-contaminated item (m^2).

1121. For a radionuclide with no decay chain, the dose arising from external irradiation to a volume-contaminated item is given by:

$$Dose_{Vol,ext} = \frac{I \cdot e^{-\lambda t} \cdot t_{ext} \cdot SF_{Vol} \cdot D_{ext,Vol}^{Rn} \cdot t_{ext}}{M}$$

where:

- I is the inventory of radionuclide *Rn* assumed in the object (Bq);
- λ is the decay constant (year^{-1});
- t is the time at which the dose is assessed (years after consignment);
- t_{ext} is the exposure time (h);
- SF_{Vol} is the scaling factor from a semi-infinite slab to a volume-contaminated spherical source at a distance of 0.3 m from the surface of the sphere (dimensionless, see Table 164);
- $D_{ext,Vol}^{Rn}$ is the dose coefficient from a semi-infinite slab for radionuclide *Rn* (Sv y^{-1} per Bq kg^{-1}; see Section E.9.3); and,
- M is the mass of the volume-contaminated item. units

1122. The doses from contaminated items disposed to Port Clarence with a given level of contamination will lie in between the doses calculated for a surface-contaminated sphere and a volume contaminated sphere. An indication of this dose can be given by the geometric mean of the doses from the surface-contaminated sphere and a volume-contaminated sphere:

$$Dose_{Geometric\ mean} = \left(Dose_{Vol} \cdot Dose_{Surf}\right)^{\frac{1}{2}}$$

1123. The parameters used in these calculations are given in Table 162. For radionuclides with decay chains, the approach set out in Section 3.2 is used to determine effective doses and thus Discrete Item Limits.

Table 162 Discrete Item Assessment parameters

Parameter	Units	Value	Description
I	Bq	$1\ 10^{9}$	Radionuclide inventory assumed on item
SI_{ing}	$m^2\ h^{-1}$	$1\ 10^{-6}$	Secondary ingestion coefficient for surface contamination
I_{ing}	$kg\ h^{-1}$	$1\ 10^{-7}$	Inadvertent ingestion rate for contaminated material
t_{ext}	$h\ y^{-1}$	1	Exposure time
ρ_{sphere}	$kg\ m^{-3}$	2000	Sphere density
t	y	2400	Time at which encounter occurs (years after disposal)

Table 163 Scaling factors from an infinite plane source to a non-sorbing surface-contaminated spherical source of given radius at a distance of 0.3 m from the surface of the sphere. Taken from Table A4 of (LLWR Ltd, 2013).

Sphere weight	10g	100g	1 kg	10 kg	100 kg	1 tonne	10 tonnes
Sphere weight (kg)	0.01	0.1	1	10	100	1000	10000
Sphere radius	0.011	0.023	0.049	0.106	0.229	0.492	1.061
Scaling factor	$1.09\ 10^{-3}$	$4.70\ 10^{-3}$	$1.86\ 10^{-2}$	$6.40\ 10^{-2}$	$1.84\ 10^{-1}$	$4.13\ 10^{-1}$	$7.49\ 10^{-1}$

Table 164 Scaling factors from a semi-infinite slab to a volume-contaminated spherical source of given radius at a distance of 0.3 m from the surface of the sphere. Taken from Table A4 of (LLWR Ltd, 2013).

Sphere weight	10g	100g	1 kg	10 kg	100 kg	1 tonne	10 tonnes
Sphere weight (kg)	0.01	0.1	1	10	100	1000	10000
Sphere radius	0.011	0.023	0.049	0.106	0.229	0.492	1.061
Scaling factor	$3.90\ 10^{-5}$	$3.61\ 10^{-4}$	$3.09\ 10^{-3}$	$2.28\ 10^{-2}$	$1.09\ 10^{-1}$	$3.50\ 10^{-1}$	$7.46\ 10^{-1}$

E.6.2.4. Doses from discrete items

1124. The doses arising from 1 hour encounter with a 1 tonne sphere contaminated with 1 GBq of each radionuclide are presented in Table 165. The doses are assessed at 2400 years after waste consignment. Doses are reported to a minimum value of $1\ 10^{-30}$ Sv.

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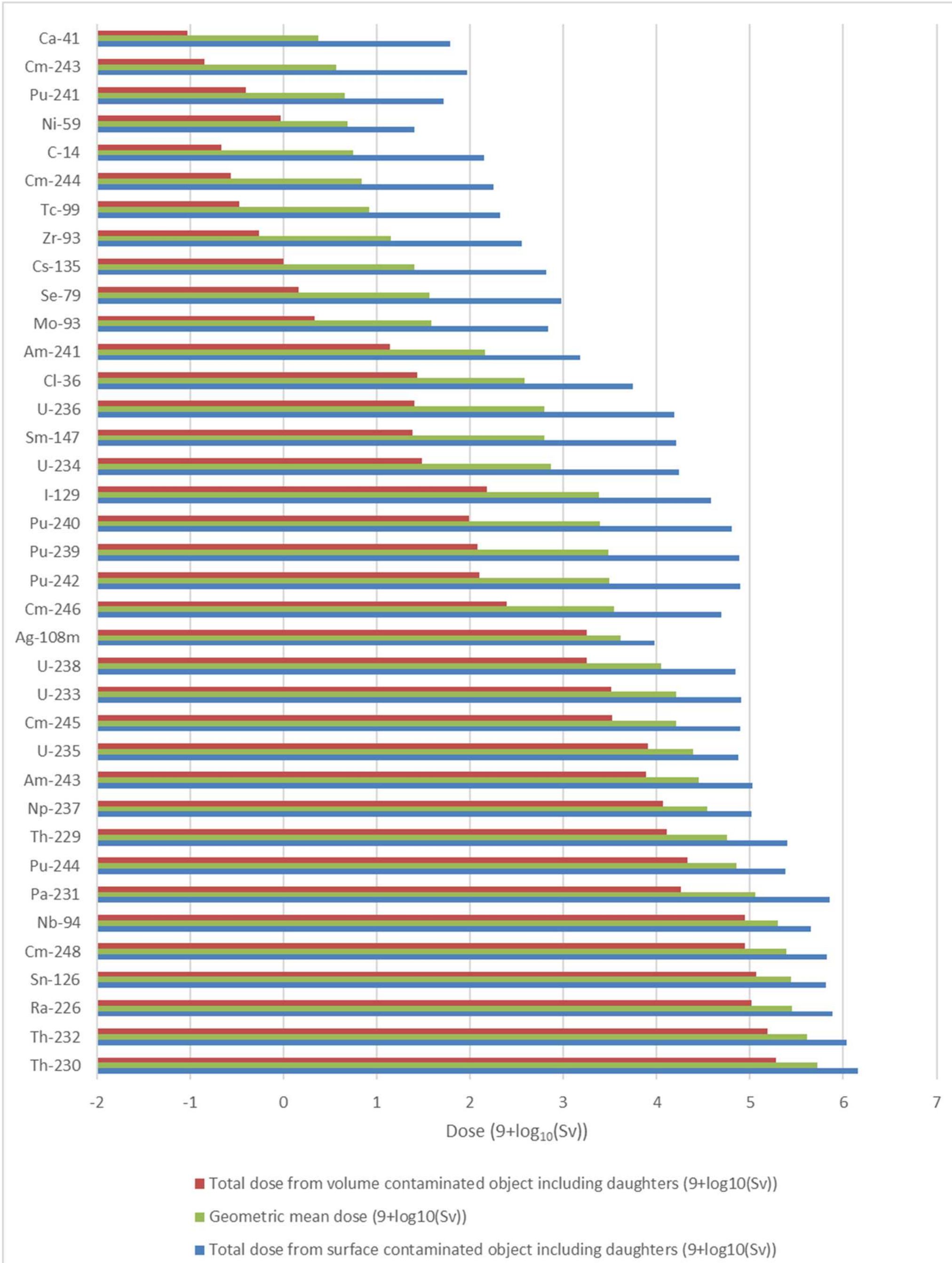
Table 165 Doses at 2400 years from encounter with a 1 tonne sphere contaminated with 1 GBq of radionuclide at time of consignment

Radionuclide	Total dose from surface contaminated object including daughters (Sv)	Total dose from volume contaminated object including daughters (Sv)	Geometric mean dose (Sv)
H-3	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
C-14	$1\ 10^{-7}$	$2\ 10^{-10}$	$6\ 10^{-9}$
Cl-36	$6\ 10^{-6}$	$3\ 10^{-8}$	$4\ 10^{-7}$
Ca-41	$6\ 10^{-8}$	$9\ 10^{-11}$	$2\ 10^{-9}$
Mn-54	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Fe-55	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Co-60	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Ni-59	$3\ 10^{-8}$	$9\ 10^{-10}$	$5\ 10^{-9}$
Ni-63	$3\ 10^{-15}$	$5\ 10^{-18}$	$1\ 10^{-16}$
Zn-65	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Se-79	$9\ 10^{-7}$	$1\ 10^{-9}$	$4\ 10^{-8}$
Sr-90	$5\ 10^{-30}$	$1\ 10^{-30}$	$2\ 10^{-30}$
Mo-93	$7\ 10^{-7}$	$2\ 10^{-9}$	$4\ 10^{-8}$
Zr-93	$4\ 10^{-7}$	$5\ 10^{-10}$	$1\ 10^{-8}$
Nb-93m	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Nb-94	$4\ 10^{-4}$	$9\ 10^{-5}$	$2\ 10^{-4}$
Tc-99	$2\ 10^{-7}$	$3\ 10^{-10}$	$8\ 10^{-9}$
Ru-106	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Ag-108m	$9\ 10^{-6}$	$2\ 10^{-6}$	$4\ 10^{-6}$
Ag-110m	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Cd-109	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Sb-125	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Sn-119m	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Sn-123	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Sn-126	$6\ 10^{-4}$	$1\ 10^{-4}$	$3\ 10^{-4}$
Te-127m	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
I-129	$4\ 10^{-5}$	$2\ 10^{-7}$	$2\ 10^{-6}$
Ba-133	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Cs-134	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Cs-135	$7\ 10^{-7}$	$1\ 10^{-9}$	$3\ 10^{-8}$
Cs-137	$2\ 10^{-28}$	$4\ 10^{-29}$	$9\ 10^{-29}$
Ce-144	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Pm-147	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Sm-147	$2\ 10^{-5}$	$2\ 10^{-8}$	$6\ 10^{-7}$
Sm-151	$3\ 10^{-16}$	$5\ 10^{-19}$	$1\ 10^{-17}$
Eu-152	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Eu-154	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Eu-155	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Gd-153	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Pb-210	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Po-210	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Ra-226	$8\ 10^{-4}$	$1\ 10^{-4}$	$3\ 10^{-4}$
Ra-228	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Ac-227	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$

Radionuclide	Total dose from surface contaminated object including daughters (Sv)	Total dose from volume contaminated object including daughters (Sv)	Geometric mean dose (Sv)
Th-228	$1\ 10^{-30}$	$1\ 10^{-30}$	$1\ 10^{-30}$
Th-229	$2\ 10^{-4}$	$1\ 10^{-5}$	$6\ 10^{-5}$
Th-230	$1\ 10^{-3}$	$2\ 10^{-4}$	$5\ 10^{-4}$
Th-232	$1\ 10^{-3}$	$2\ 10^{-4}$	$4\ 10^{-4}$
Pa-231	$7\ 10^{-4}$	$2\ 10^{-5}$	$1\ 10^{-4}$
U-232	$4\ 10^{-14}$	$8\ 10^{-15}$	$2\ 10^{-14}$
U-233	$8\ 10^{-5}$	$3\ 10^{-6}$	$2\ 10^{-5}$
U-234	$2\ 10^{-5}$	$3\ 10^{-8}$	$7\ 10^{-7}$
U-235	$8\ 10^{-5}$	$8\ 10^{-6}$	$2\ 10^{-5}$
U-236	$2\ 10^{-5}$	$3\ 10^{-8}$	$6\ 10^{-7}$
U-238	$7\ 10^{-5}$	$2\ 10^{-6}$	$1\ 10^{-5}$
Np-237	$1\ 10^{-4}$	$1\ 10^{-5}$	$4\ 10^{-5}$
Pu-238	$6\ 10^{-9}$	$1\ 10^{-11}$	$2\ 10^{-10}$
Pu-239	$8\ 10^{-5}$	$1\ 10^{-7}$	$3\ 10^{-6}$
Pu-240	$6\ 10^{-5}$	$1\ 10^{-7}$	$2\ 10^{-6}$
Pu-241	$5\ 10^{-8}$	$4\ 10^{-10}$	$5\ 10^{-9}$
Pu-242	$8\ 10^{-5}$	$1\ 10^{-7}$	$3\ 10^{-6}$
Pu-244	$2\ 10^{-4}$	$2\ 10^{-5}$	$7\ 10^{-5}$
Am-241	$2\ 10^{-6}$	$1\ 10^{-8}$	$1\ 10^{-7}$
Am-242m	$4\ 10^{-9}$	$9\ 10^{-12}$	$2\ 10^{-10}$
Am-243	$1\ 10^{-4}$	$8\ 10^{-6}$	$3\ 10^{-5}$
Cm-242	$2\ 10^{-15}$	$3\ 10^{-18}$	$9\ 10^{-17}$
Cm-243	$9\ 10^{-8}$	$1\ 10^{-10}$	$4\ 10^{-9}$
Cm-244	$2\ 10^{-7}$	$3\ 10^{-10}$	$7\ 10^{-9}$
Cm-245	$8\ 10^{-5}$	$3\ 10^{-6}$	$2\ 10^{-5}$
Cm-246	$5\ 10^{-5}$	$2\ 10^{-7}$	$4\ 10^{-6}$
Cm-248	$7\ 10^{-4}$	$9\ 10^{-5}$	$2\ 10^{-4}$

1125. These doses are also illustrated graphically in Figure 29, ordered according to geometric mean dose. A value of 3 on the scale corresponds to a dose of 1 microSievert, a value of 6 to 1 milliSievert, and a value of 0 to 1 nanoSievert. Radionuclides for which calculated geometric mean doses are less than 1 nanoSierverts are not shown.

Figure 29 Doses at 2400 years from encounter with a 1 tonne sphere contaminated with 1 GBq of radionuclide at consignment



1126. For most radionuclides, calculated geometric mean doses are less than 10 µSv. The highest calculated geometric mean dose is associated with disposal of Th-230, which gives rise to a geometric mean dose of 0.5 mSv, largely as a result of ingrowth of Ra-226. Other high impact radionuclides include Th-232, Ra-226, Sn-126, and Cm-248, which all give rise to geometric mean doses greater than 0.1 mSv. These radionuclides all have half-lives of thousands of years that emit, or decay to short-lived progeny that emit, significant photon emissions. All calculated doses are below 3 mSv.

1127. The activity that a discrete item must have to give rise to an effective dose of 20 µSv y^{-1} was also calculated for each radionuclide. These activities are presented in Table 166 for doses calculated at 2400 years assuming that the discrete item can be modelled as a 1 tonne sphere.

Table 166 Activity at consignment to give rise to an effective dose of 20 µSv y^{-1} based on doses calculated at 2400 years from a 1 hour encounter with a 1 tonne sphere.

Radionuclide	Activity to give effective dose limit from surface contaminated object including daughters (MBq)	Activity to give effective dose limit from volume contaminated object including daughters (MBq)	Activity to give effective dose limit using geometric mean dose (MBq)
H-3	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
C-14	$1\ 10^{5}$	$9\ 10^{7}$	$4\ 10^{6}$
Cl-36	$4\ 10^{3}$	$7\ 10^{5}$	$5\ 10^{4}$
Ca-41	$3\ 10^{5}$	$2\ 10^{8}$	$8\ 10^{6}$
Mn-54	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Fe-55	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Co-60	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Ni-59	$8\ 10^{5}$	$2\ 10^{7}$	$4\ 10^{6}$
Ni-63	$7\ 10^{12}$	$4\ 10^{15}$	$2\ 10^{14}$
Zn-65	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Se-79	$2\ 10^{4}$	$1\ 10^{7}$	$5\ 10^{5}$
Sr-90	$4\ 10^{27}$	$2\ 10^{28}$	$9\ 10^{27}$
Mo-93	$3\ 10^{4}$	$9\ 10^{6}$	$5\ 10^{5}$
Zr-93	$6\ 10^{4}$	$4\ 10^{7}$	$1\ 10^{6}$
Nb-93m	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Nb-94	$5\ 10^{1}$	$2\ 10^{2}$	$1\ 10^{2}$
Tc-99	$1\ 10^{5}$	$6\ 10^{7}$	$2\ 10^{6}$
Ru-106	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Ag-108m	$2\ 10^{3}$	$1\ 10^{4}$	$5\ 10^{3}$
Ag-110m	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Cd-109	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Sb-125	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Sn-119m	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Sn-123	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Sn-126	$3\ 10^{1}$	$2\ 10^{2}$	$7\ 10^{1}$
Te-127m	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
I-129	$5\ 10^{2}$	$1\ 10^{5}$	$8\ 10^{3}$
Ba-133	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Cs-134	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Cs-135	$3\ 10^{4}$	$2\ 10^{7}$	$8\ 10^{5}$

Radionuclide	Activity to give effective dose limit from surface contaminated object including daughters (MBq)	Activity to give effective dose limit from volume contaminated object including daughters (MBq)	Activity to give effective dose limit using geometric mean dose (MBq)
Cs-137	$1\ 10^{26}$	$5\ 10^{26}$	$2\ 10^{26}$
Ce-144	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Pm-147	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Sm-147	$1\ 10^{3}$	$8\ 10^{5}$	$3\ 10^{4}$
Sm-151	$7\ 10^{13}$	$4\ 10^{16}$	$2\ 10^{15}$
Eu-152	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Eu-154	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Eu-155	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Gd-153	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Pb-210	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Po-210	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Ra-226	$3\ 10^{1}$	$2\ 10^{2}$	$7\ 10^{1}$
Ra-228	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Ac-227	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Th-228	$2\ 10^{28}$	$2\ 10^{28}$	$2\ 10^{28}$
Th-229	$8\ 10^{1}$	$2\ 10^{3}$	$4\ 10^{2}$
Th-230	$1\ 10^{1}$	$1\ 10^{2}$	$4\ 10^{1}$
Th-232	$2\ 10^{1}$	$1\ 10^{2}$	$5\ 10^{1}$
Pa-231	$3\ 10^{1}$	$1\ 10^{3}$	$2\ 10^{2}$
U-232	$5\ 10^{11}$	$2\ 10^{12}$	$1\ 10^{12}$
U-233	$3\ 10^{2}$	$6\ 10^{3}$	$1\ 10^{3}$
U-234	$1\ 10^{3}$	$7\ 10^{5}$	$3\ 10^{4}$
U-235	$3\ 10^{2}$	$2\ 10^{3}$	$8\ 10^{2}$
U-236	$1\ 10^{3}$	$8\ 10^{5}$	$3\ 10^{4}$
U-238	$3\ 10^{2}$	$1\ 10^{4}$	$2\ 10^{3}$
Np-237	$2\ 10^{2}$	$2\ 10^{3}$	$6\ 10^{2}$
Pu-238	$3\ 10^{6}$	$2\ 10^{9}$	$8\ 10^{7}$
Pu-239	$3\ 10^{2}$	$2\ 10^{5}$	$7\ 10^{3}$
Pu-240	$3\ 10^{2}$	$2\ 10^{5}$	$8\ 10^{3}$
Pu-241	$4\ 10^{5}$	$5\ 10^{7}$	$4\ 10^{6}$
Pu-242	$3\ 10^{2}$	$2\ 10^{5}$	$6\ 10^{3}$
Pu-244	$8\ 10^{1}$	$9\ 10^{2}$	$3\ 10^{2}$
Am-241	$1\ 10^{4}$	$1\ 10^{6}$	$1\ 10^{5}$
Am-242m	$5\ 10^{6}$	$2\ 10^{9}$	$1\ 10^{8}$
Am-243	$2\ 10^{2}$	$3\ 10^{3}$	$7\ 10^{2}$
Cm-242	$9\ 10^{12}$	$6\ 10^{15}$	$2\ 10^{14}$
Cm-243	$2\ 10^{5}$	$1\ 10^{8}$	$5\ 10^{6}$
Cm-244	$1\ 10^{5}$	$7\ 10^{7}$	$3\ 10^{6}$
Cm-245	$3\ 10^{2}$	$6\ 10^{3}$	$1\ 10^{3}$
Cm-246	$4\ 10^{2}$	$8\ 10^{4}$	$6\ 10^{3}$
Cm-248	$3\ 10^{1}$	$2\ 10^{2}$	$8\ 10^{1}$

1128. These limits give the maximum level of activity that could be present on an item at consignment in order to give an assessed effective dose of less than 20 µSv y^{-1}, assuming the activity distribution specified.

E.6.2.5. Radionuclide groups and Port Clarence Discrete Item Limits

1129. Following the LLWR approach, rather than set limits on discrete items for every radionuclide, the radionuclides have been placed into groups based on the calculated effective dose from discrete items. Such an approach enables acceptability against a limit to be determined based on the radionuclides with the highest contribution to the total activity within each group, rather than having to make an explicit assessment for each radionuclide.

1130. Initial radionuclide groupings were taken from the assessment undertaken by LLW Repository Ltd. These groupings were subsequently refined based on a comparison of the LLWR Discrete Item Limits and the calculated activities to give an effective dose of 20 µSv y^{-1} for a surface-contaminated 1 tonne sphere at Port Clarence. Consequently, five radionuclide groups are proposed to give Discrete Item limits at Port Clarence. The details and justification of this method are described below.

1131. Where radionuclides were not explicitly assigned groups within LLW repository Ltd's assessment, the guidance provided in (LLWR Ltd, 2013) was used to allocate a group to that radionuclide. The initial radionuclide groupings used in this assessment are shown in Table 167.

1132. These initial radionuclide groupings enabled the calculated activities to give an effective dose of 20 µSv y^{-1} to be compared to the Discrete Item Limits at the LLWR (see Table 168) for each radionuclide. The calculated activity to give an effective dose of 20 µSv y^{-1} from a 1 hour exposure to a surface-contaminated 1 tonne sphere was compared to the LLWR Discrete Item Limit for a 1 tonne item. If this calculated activity was less than the LLWR Discrete Item Limit, then the Port Clarence Discrete Item Limit was set to a factor of 10 lower than the LLWR Discrete Item Limit. If this calculated activity was greater than the LLWR Discrete Item Limit then the LLWR Discrete Item Limit was applied. Hence, the Port Clarence Discrete item Limits are less than or equal to the LLWR Discrete Item Limits.

1133. This approach sets all of the Port Clarence Discrete Item Limits below the calculated activity to give an effective dose of 20 µSv y^{-1} from a 1 hour exposure to a surface-contaminated 1 tonne sphere for all radionuclides except U-233. For U-233, the Port Clarence Discrete Item Limit was decreased by a further factor of ten.

1134. As a result of this method, five radionuclide groups are proposed to give Discrete Item Limits at Port Clarence. These groups and the radionuclides in these groups are given in Table 169.[1] The corresponding Discrete Item Limits proposed for Port Clarence are given in Table 170. The relationship between the Discrete item Limit for different

[1] We denote the five radionuclide groups for Port Clarence using lower-case letters a, b, c, d and e to prevent confusion with the LLWR Discrete Item Limits.

masses is the same as that used by LLWR, namely for items less than 1 kg the Discrete Item Limit is 0.01 of the Discrete Item Limit for 100 kg or over, with a linear relationship for in masses in between.

Table 167 Radionuclide groups for Discrete Item Limits as used at LLWR (Table 6-2 of (LLW Repository Ltd, August 2013)). Radionuclides not explicitly grouped at the LLWR are highlighted in red.

Parameter	Radionuclides
Group A	Nb-94 Ag-108m Sn-126 Ra-226 Th-229 Th-230 Th-232 Pa-231 U-232 Np-237 Pu-244 Am-243 Cm-248
Group B1	Se-79 I-129 Sm-147 U-235 U-238 Pu-238 Pu-239 Pu-240 Pu-242 Am-241 Am-242m Cm-245 Cm-246
Group B2	C-14 Cl-36 Ca-41 Sr-90 Mo-93 Zr-93 Tc-99 Cs-135 Cs-137 Ac-227 Pb-210 U-233 U-234 U-236 Pu-241 Cm 243 Cm-244
Group C	H-3 Mn-54 Fe-55 Co-60 Ni-59 Ni-63 Zn-65 Nb-93m Ru-106 Ag-110m Cd-109 Sb-125 Sn-119m Sn-123 Te-127m Ba-133 Cs-134 Ce-144 Pm-147 Sm-151 Eu-152 Eu-154 Eu-155 Gd-153 Po-210 Ra-228 Th-228 Cm-242

Table 168 Discrete Item Limits for LLWR

	Weight 1 kg or less	Weight between 1 and 100 kg	Weight 100 kg or greater
Group A	0.001 GBq	1 GBq te^{-1}	0.1 GBq
Group B1	0.01 GBq	10 GBq te^{-1}	1 GBq
Group B2	0.1 GBq	100 GBq te^{-1}	10 GBq
Group C	1 GBq	1000 GBq te^{-1}	100 GBq

Table 169 Radionuclide groups for Discrete Item Limits at Port Clarence. Radionuclides with stricter Discrete Item Limits than those explicitly applied at the LLWR are in red.[2]

Parameter	Radionuclides
Group a	Nb-94 Sn-126 Ra-226 Th-229 Th-230 Th-232 Pa-231 Pu-244 Cm-248
Group b	Ag-108m I-129 U-232 U-233 U-235 U-238 Np-237 Pu-239 Pu-240 Pu-242 Am-243 Cm-245 Cm-246
Group c	Cl-36 Se-79 Sm-147 U-234 U-236 Pu-238 Am-241 Am-242m
Group d	C-14 Ca-41 Sr-90 Mo-93 Zr-93 Tc-99 Cs-135 Cs-137 Pb-210 Ac-227 Pu-241 Cm-243 Cm-244
Group e	H-3 Mn-54 Fe-55 Co-60 Ni-59 Ni-63 Zn-65 Nb-93m Ru-106 Ag-110m Cd-109 Sb-125 Sn-119m Sn-123 Te-127m Ba-133 Cs-134 Ce-144 Pm-147 Sm-151 Eu-152 Eu-154 Eu-155 Gd-153 Po-210 Ra-228 Th-228 Cm-242

Table 170 Discrete Item Limits for Port Clarence

	Weight 1 kg or less	Weight between 1 and 100 kg	Weight 100 kg or greater	LLWR group with the same limit
Group a	0.0001 GBq	0.1 GBq te^{-1}	0.01 GBq	Not used
Group b	0.001 GBq	1 GBq te^{-1}	0.1 GBq	A
Group c	0.01 GBq	10 GBq te^{-1}	1 GBq	B1
Group d	0.1 GBq	100 GBq te^{-1}	10 GBq	B2
Group e	1 GBq	1000 GBq te^{-1}	100 GBq	C

1135. The Discrete Item Limits at Port Clarence are lower (i.e. they are more restrictive) than the LLWR Discrete Item Limits for the following radionuclides: Nb-94, Sn-126, Ra-226, Th-229, Th-230, Th-232, Pa-231, Cm-248, I-129, U-233, U-235, U-238, Pu-239, Pu-240, Pu-242, Cm-245, Cm-246, Cl-36, U-234 and U-236.

1136. The application of equal or more restrictive limits for discrete items at Port Clarence compared to LLWR is conservative since the calculations indicate that higher Discrete Item Limits could be used for many radionuclides. The higher limits reflect the fact that the LLWR Discrete Item Limits were based on an assessment of doses some 300

[2] Only radionuclides explicitly grouped by the LLWR (those not highlighted in red in Table 169) are considered when indicating if there is a stricter limit.

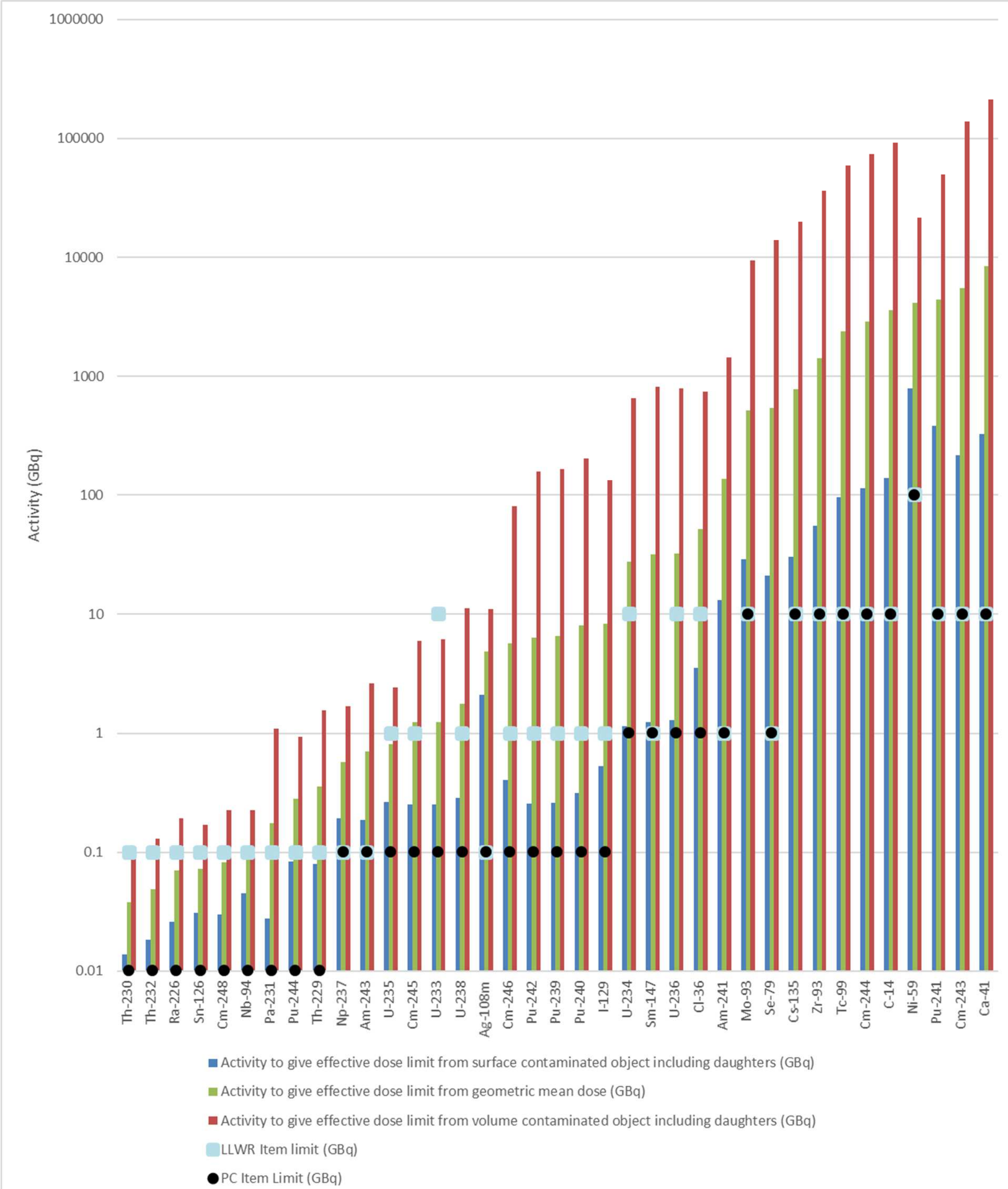
years after closure of that site, whereas at Port Clarence erosion, and hence exposure to items, is not expected to occur for 2400 years after closure.

1137. Another conservatism arises from the fact that the Port Clarence Discrete Item Limits have been set on the basis of the dose from a surface-contaminated item of mass 1 tonne. In reality, the activity within wastes will not be solely distributed on the surface of the contaminated items, ut will penetrate into the volume of the items. Consequently, the dose calculated from a surface-contaminated item will be an over-estimate of the anticipated dose, and the activity to give an effective dose of 20 µSv y^{-1} from a surface-contaminated sphere will be less than the activity to give an effective dose of 20 µSv y^{-1} for an item with a component of activity distributed within the volume of the item.

1138. Figure 30 demonstrates the item activity that would be required to give an effective dose of 20 µSv y^{-1} for surface- and volume- contaminated spheres of mass 1 tonne. Radionuclides are ordered in terms of geometric dose, and radionuclides for which calculated geometric mean doses are less than 1 nanoSievert are not shown.

1139. The LLWR Discrete Item Limits and proposed Port Clarence Discrete Item Limits for each radionuclide are also plotted on Figure 30. Radionuclides for which calculated geometric mean doses are less than 1 nanoSiervert are not shown.

Figure 30 Activity required to give an effective dose of 20 µSv y^{-1} from a 1 hour exposure to a 1 tonne contaminated sphere at 2400 years after consignment



1140. Not all items to be consigned to Port Clarence will have a mass of around 1 tonne, therefore smaller masses were also considered to check that the proposed Discrete Item Limits were appropriate.

1141. The item activities to give rise to an effective dose of 20 µSv y^{-1} from items with mass 10 g, 100 g, 1 kg and 10 tonnes were calculated for each radionuclide. The radionuclides with the greatest impact from a 10 g sphere, 10 kg sphere and 10 tonne sphere were then identified for each radionuclide group given in Table 169. The results of this analysis are presented in Table 171.

Table 171 Highest impact radionuclides within each Port Clarence group.

Highest impact radionuclide within specified group for specified sphere mass. The bracketed term gives the activity required to give an effective dose of 20 µSv y^{-1} for that radionuclide for the specified sphere mass to one significant figure (Bq)			
	10 g	10 kg	10 tonne
Group a	Pa-231 (6 10^{4})	Th-230 (5 10^{6})	Th-230 (1 10^{8})
Group b	Pu-242 (4 10^{5})	Np-237 (7 10^{7})	Np-237 (2 10^{9})
Group c	U-234 (2 10^{6})	U-234 (6 10^{8})	U-234 (2 10^{11})
Group d	Mo-93 (5 10^{7})	Mo-93 (2 10^{10})	Mo-93 (3 10^{12})
Group e	Ni-59 (2 10^{9})	Ni-59 (3 10^{11})	Ni-59 (2 10^{13})

1142. The proposed Port Clarence Discrete Item Limits for different masses are illustrated in the graphs in Figure 31 to

1143. Figure 35 for a number of these high impact radionuclides, namely Th-230, Np-237, U-234, Mo-93 and Ni-59.

1144. It is reasonable for the Discrete Item Limit to lie between the activity leading to an effective dose of 20 µSv y^{-1} for the volume- and surface-contaminated items because not all activity will be uniformly distributed within the volume of the item or over its surface. It is also reasonable for the Discrete Item Limit to be less restrictive for small items because, generally speaking, larger items are likely to be of more interest to an estuary bank user and to stay on the riverbank for longer, whereas smaller items are less likely to be seen and more likely to be removed by natural river or tidal action. In addition, comparison with the activity leading to a dose of 20 µSv y^{-1} is a conservative approach to ensuring that the dose to a person deliberately encountering a discrete item will meet the lower dose guidance level of 3 mSv.

1145. The graph for Th-230 (Figure 31) illustrates that the Discrete Item Limits for Group a radionuclides are below the geometric mean of the activities leading to a dose of 20 µSv y^{-1} from the volume- and surface-contaminated items for all masses above about 30 g. The Discrete Item Limits for Group a radionuclides are also below the activity leading to an effective dose of 20 µSv y^{-1} for all weights if the item is assumed to be volume-contaminated. On this basis, and noting that Th-230 is the radionuclide within Group a that gives rise to the greatest impact, the proposed Discrete Item Limits for Group a are deemed appropriate.

1146. The graph for Np-237 (Figure 32) illustrates that the Discrete Item Limits for Group b radionuclides are below the geometric mean of the activities leading to a dose of 20 µSv y^{-1} from the volume- and surface-contaminated items for all assessed masses. On this basis, and noting that Np-237 is one of the radionuclides within Group b that gives rise to the greatest impact, the proposed Discrete Item Limits for Group b are deemed appropriate.

1147. The graph for U-234 (Figure 33) illustrates that the Discrete Item Limits for Group c radionuclides will ensure that the effective dose is below 20 µSv y^{-1} for all weights above about 30 g if the item is assumed to be volume-contaminated, and for all weights above 100 g based on the geometric mean. On this basis, and noting that U-234 is one of the radionuclides within Group c that gives rise to the greatest impact, particularly for smaller item masses, the proposed Discrete Item Limits for Group c are deemed appropriate.

1148. The Group d Discrete Item Limits are illustrated in the graph for Mo-93 (Figure 34). The Discrete Item Limits for Group d radionuclides are below the activity leading to an effective dose of 20 µSv y^{-1} for all weights if the item is assumed to be volume-contaminated, and for all weights above about 30 g based on the geometric mean. As such, the proposed Discrete Item Limits for Group d radionuclides are deemed to be appropriate.

1149. The Group e Discrete Item Limits are illustrated in the graph for Ni-59 (Figure 35). The Discrete Item Limits for Group e radionuclides will ensure that the effective dose is below 20 µSv y^{-1} for all weights based on the geometric mean dose, and if the item is assumed to be volume-contaminated. If the item is assumed to be surface-contaminated then implementation of these Discrete Item Limits would ensure doses below 20 µSv y^{-1} for items of mass greater than about 30 g. Consequently, the Discrete Item Limits for Group e radionuclides are deemed to be cautious because some component of activity is likely to be distributed within the volume of the object.

Figure 31 Item activity leading to a dose of 20 µSv y^{-1} for Th-230

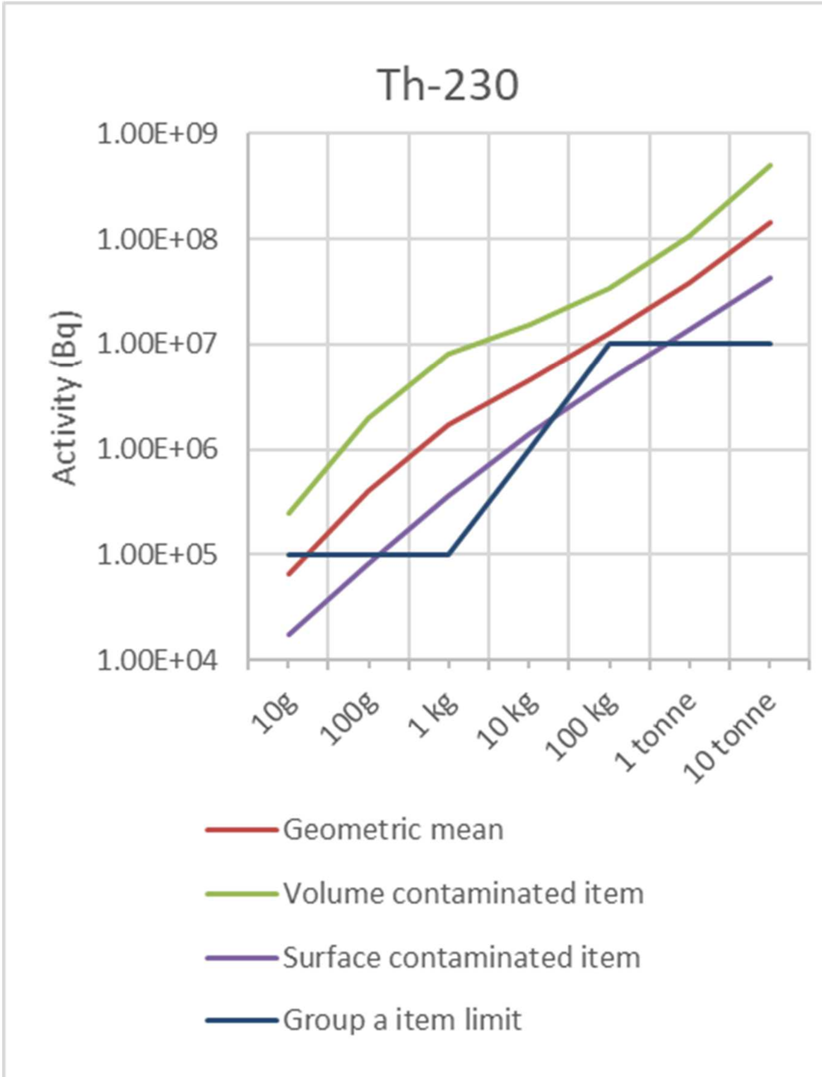


Figure 32 Item activity leading to a dose of 20 µSv y^{-1} for Np-237

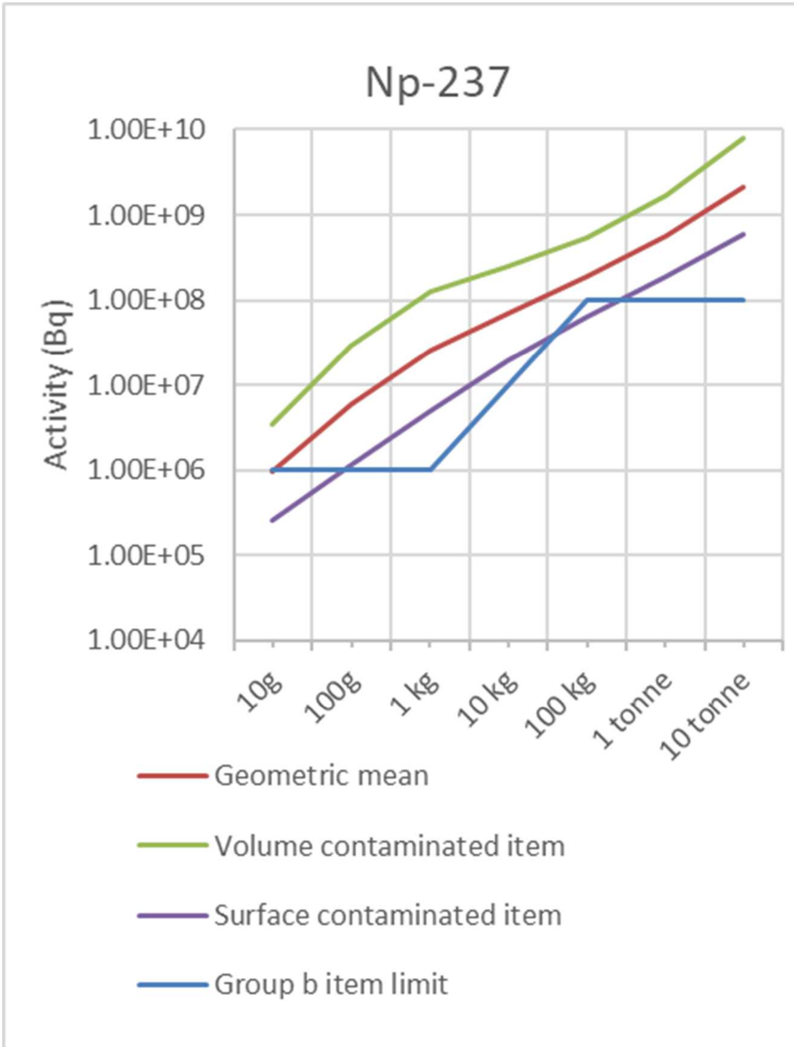


Figure 33 Item activity leading to a dose of 20 µSv y^{-1} for U-234

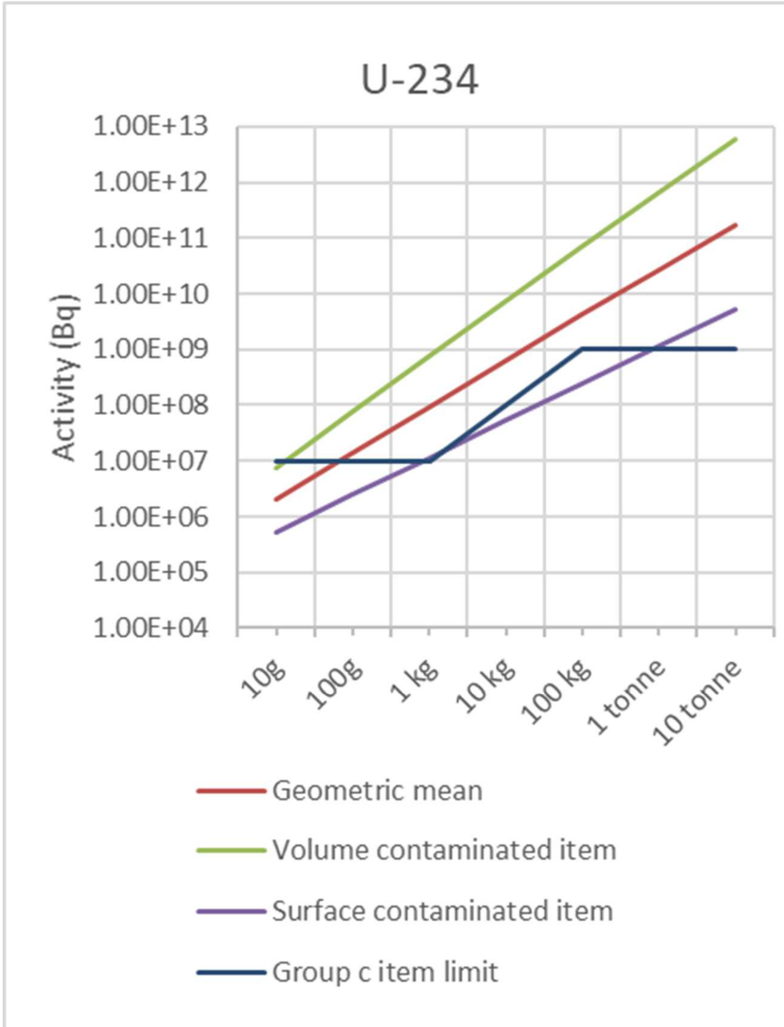


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Figure 34 Item activity leading to a dose of 20 µSv y^{-1} for Mo-93

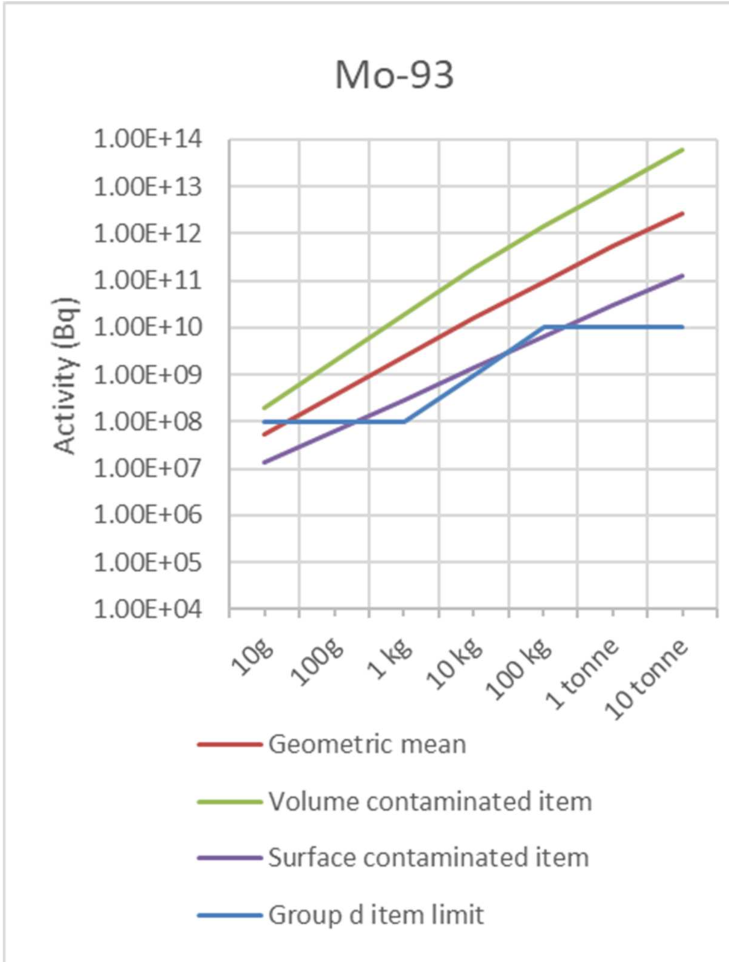
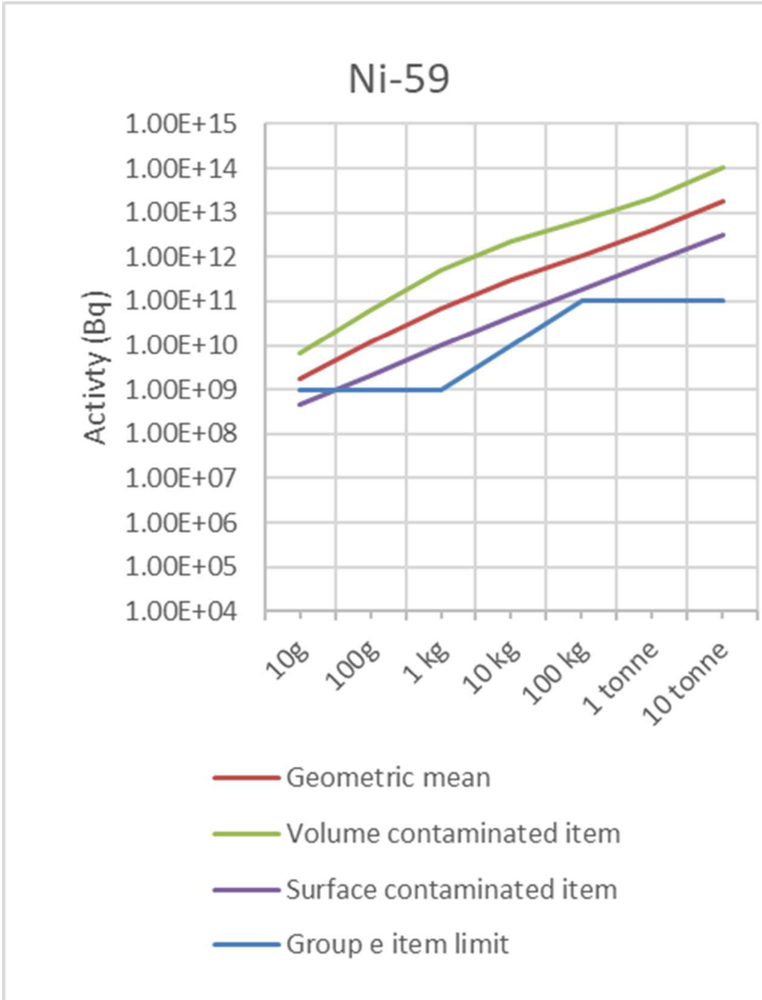


Figure 35 Item activity leading to a dose of 20 µSv y^{-1} for Ni-59



1150. Hence, the proposed Discrete Item Limits will limit the assessed effective doses to an estuary bank user, following erosion of the site, from the disposal of discrete items at Port Clarence to within acceptable levels.

1151. In addition, the specific activity concentration limits for each consignment and each package will tend to prevent items with activities near the Discrete Item Limits being disposed at Port Clarence. This is illustrated in Figure 36 for a consignment specific activity limit of 200 Bq g^{-1} and a specific activity concentration per package of 1000 Bq g^{-1}.

Figure 36 Comparison of Port Clarence Discrete Item Limits with those for individual consignments and waste packages

E.6.2.6. Waste Acceptance using the Port Clarence Discrete Item Limits

1152. It has been demonstrated that the proposed Discrete Item Limits will provide adequate protection to a potential future user of the bank of the estuary following future erosion of the landfill. Here, a process for applying these Discrete Item Limits to waste is identified.

1153. In the first instance, waste consignors should determine whether any items within a consignment should be classified as a discrete item. Guidance on what can be classified as a discrete item can be obtained by consulting LLW Repository Ltd's Discrete Item Library (LLWR Ltd, 2019). Waste consignors should also contact Augean Ltd for guidance.

1154. Based on the activity of each radionuclide on the item and the Discrete Item Limits for the item, a Sum of Fractions approach to determine acceptability of that item should then be used.

1155. The Sum of Fractions is given by:

$$SoF = \frac{Q_a}{L_a} + \frac{Q_b}{L_b} + \frac{Q_c}{L_c} + \frac{Q_d}{L_d} + \frac{Q_e}{L_e},$$

where Q_n is the total activity of group n radionuclides on the item and L_n is the Port Clarence Discrete Item Limit for that group (given in Table 170).

1156. If a radionuclide is known to be present on an item, is not listed in Table 169 and has a half-life greater than 200 years then the radionuclide should be cautiously assigned to Group a. Otherwise it should be assigned to Group e, unless it decays to an alpha-emitting daughter with a half-life a few tens to hundreds of time the parent half-life, in which case ingrown progeny are liable to determine the impact at 2400 years.

1157. If this Sum of Fraction is less than one, the item is acceptable for disposal within a consignment at Port Clarence, subject to meeting other Waste Acceptance Criteria including the activity concentration limits for a consignment and for a package.

E.6.3. Excavation of particles

1158. Radioactive particles are small items that could be as small as a grain of sand and could be incorporated in a radioactive waste stream or package. The possibility that future intrusion events could lead to unintentional recovery of, and exposure to, radioactive particles is considered. Migration of particles in groundwater or uptake from soil into the food chain is not considered credible.

1159. The methodology for assessing the dose implications of excavating waste materials that include particles is described here, together with the approach to waste acceptance criteria.

E.6.3.1. Assessment approach

1160. The assessment approach is based on that applied in the ENRMF ESC, see Appendix E, Section E5.10 (Eden NE, 2015a). It draws on the work (Sumerling, 2013) undertaken for the LLWR ESC. Whereas the ENRMF ESC considered a set of particle types (with different radionuclide characteristics) the methodology was subsequently developed so that it could assess the dose arising from any radionuclide associated with a particle. This methodology was implemented in an Excel workbook (Particle assessment tool v2.xlsx) for use by Augean on decisions regarding acceptability of waste at the ENRMF. This section describes the methodology implemented in the Excel workbook (referred to here as the spreadsheet) and lists the results of the dose calculations.

1161. The spreadsheet considers the radionuclides which are included in the current permit for ENRMF and allows the user to add "Other Radionuclides" with a half-life greater than 1 year or as specified in writing by the Environment Agency. All these radionuclides are included in the Port Clarence ESC. The dose calculations include the first daughter radionuclide of a decay chain not assumed within dose coefficients to be in secular equilibrium, i.e. those radionuclides listed as radioactive daughters that need to be considered explicitly. A worksheet is also included to allow calculation of doses assuming secular equilibrium.

1162. The pathways considered are as follows:

- Ingestion of 1 mm particle; and,
- External exposure to a 1 mm particle (whole body doses and skin doses).

1163. The doses from these pathways are not considered to be additive.

1164. Inhalation of particles is not considered as it is not relevant for particles of 1 mm in size and inhalation of particles up to 10 μm in size was found not to be an important pathway in other assessments of particles (Sumerling, 2013; HPA, 2005; HPA, 2011).

1165. Two different times of inadvertent intrusion are considered: 60 and 300 years after disposals end, respectively. The earliest time of inadvertent intrusion, 60 years, corresponds to intrusion occurring at the end of institutional oversight of the restored landfill. However, inadvertent intrusion after a longer period of time is considered more realistic and has been based on the maximum period of active institutional control considered by the Environment Agency (EA, 300 years) presented in the NS-GRA (UK Environment Agencies, 2009).

1166. The calculations take no account of the probability that the person who is intruding into the landfill site actually comes into contact with the particle being considered. Given the quantity of soil, waste and other material that would be excavated during an intrusion event, the probability of inadvertently ingesting the particle or of it becoming trapped against the skin or under a nail is extremely small.

1167. Measurements (HPA, 2005; Tyler, et al., 2013; HPA, 2011) have found that particles are not 100% soluble in the gastro-intestinal tract and therefore ingestion doses calculated using the standard ICRP gut uptake factors (ICRP, 2012) are unrealistically high. The spreadsheet allows the user to enter different uptake factors or to enter a particle solubility. The EA will require experimental evidence (for the particles being disposed) that the use of different uptake factors or reduced solubility is justified.

E.6.3.2. Methodology

1168. The methodology considers three exposure pathways: ingestion, external (whole body) and external (skin). The doses due to each of these pathways are not considered to be additive. It is assumed conservatively that exposure occurs 60 or 300 years after emplacement of the waste, as a result of deliberate excavation of the site.

Ingestion

1169. Inadvertent ingestion is typically size restricted and it is assumed here that particles for inadvertent ingestion are essentially spherical with a nominal diameter of 1 mm. The precise dimensions are not important providing the particle is sufficiently small that it remains inadvertently ingestible (e.g. anywhere in the range of 1 to a few mm diameter). The dose, if ingested, depends on the activity of the particle rather than the size. Dose is estimated on a per particle basis.

1170. Ingestion dose is calculated in one of two ways, depending on how the solubility of the particle is taken into account. The first method uses a specified particle solubility, *Sol,* to scale the dose:

$$Dose_{ing} = D_{ing}^{Rn}.Sol.A_{Rn}(t)$$

where:

- $Dose_{ing}$ is the dose from ingestion of the particle (Sv);
- D_{ing}^{Rn} is the ICRP dose coefficient for ingestion of radionuclide *Rn* (Sv Bq^{-1});
- $A_{Rn}(t)$ is the activity of the particle (Bq) at the time of exposure (t);
- *Sol* is the solubility of the particle in the gastro-intestinal tract.

1171. *Sol* applies to all radionuclides in the particle, including short lived daughter radionuclides in secular equilibrium.

1172. The second method uses specified particle solubility related gut uptake values for each radionuclide on the particle, $f1_{Rn}$, to scale the dose coefficient:

$$Dose_{ing} = D_{ing}^{Rn}.\frac{f1_{Rn}}{ICRP\ f1_{Rn}}.A_{Rn}(t)$$

Where:

- $f1_{Rn}$ is the realistic gut uptake factor for radionuclide *Rn* allowing for the solubility of the particle; and,
- $ICRP\ f1_{Rn}$ is the ICRP gut uptake factor corresponding to D_{ing}^{Rn}.

1173. The linear scaling of the ICRP dose coefficient shown above is applied in all cases except for Co, Sr, U and Pu isotopes where comparison of the ICRP dose coefficients for different f1 values for these radionuclides showed a non-linear response. The following approach is taken:

$$D_{ing}^{Rn,revised} = (a_{Rn} + b_{Rn}.f1_{Rn})$$

where:

a_{Rn} and b_{Rn} are empirically derived constants, representing the dose from the particle to the gut per unit ingestion and the dose from unit uptake in blood, respectively.

1174. Values for a_{Rn} and b_{Rn} are obtained by fitting ICRP dose coefficients for different f1 values for the same radionuclide. The values are given in Table 172.

Table 172 Constants for non-linear scaling of dose per unit ingestion calculations

Radionuclide	Dose to gut per unit ingestion, a_{Rn}	Dose per unit to blood, b_{Rn}
Co-60	1.60 10^{-9}	1.80 10^{-8}
Sr-90	1.83 10^{-9}	8.72 10^{-8}
U-232	4.50 10^{-9}	1.63 10^{-5}
U-233	3.90 10^{-9}	2.31 10^{-6}
U-234	3.80 10^{-9}	2.26 10^{-6}
U-235	4.15 10^{-9}	2.09 10^{-6}
U-236	3.70 10^{-9}	2.12 10^{-6}
U-238	3.60 10^{-9}	2.02 10^{-6}
Pu-238	4.30 10^{-9}	4.51 10^{-4}
Pu-239	4.10 10^{-9}	4.92 10^{-4}
Pu-240	4.10 10^{-9}	4.92 10^{-4}
Pu-241	2.50 10^{-11}	9.35 10^{-6}
Pu-242	3.90 10^{-9}	4.72 10^{-4}

1175. The dose coefficients for daughter radionuclides are scaled according to the ratio of the realistic f1 to the ICRP f1 for the parent:

$$D_{ing}^{Rn,revised} = D_{ing}^{Rn}.\frac{f1_{Rn,parent}}{ICRP\ f1_{Rn,parent}}$$

External exposure (whole body)

1176. External exposure is not limited by the size of the particle. A larger particle with the same activity will deliver the same external dose. An exposure time of 8 hours is assumed.

1177. External dose is thus calculated as:

$$Dose_{ext,wb} = G_{wb}^{Rn}.T_{ext}.A_{Rn}(t)$$

where:

- $Dose_{ext,wb}$ is the external effective (whole body) dose;
- G_{wb}^{Rn} is the whole body dose rate for radionuclide Rn (Sv hour^{-1} Bq^{-1}), from β and / or γ radiation as appropriate for that radionuclide;
- $A_{Rn}(t)$ is the activity of the particle (Bq) at the time of exposure (t); and,
- T_{ext} is the exposure time for external exposure (hours).

External exposure (skin)

1178. Skin dose will depend on the size of a particle. A larger particle will remain in contact with the skin for a shorter time and there will be self-absorption within the particle. It is assumed that the particle becomes lodged in direct contact with the skin (for example under a fingernail) and remains in situ for 1 hour. A nominal particle size of 1 mm is consistent with this assumption.

1179. ICRP (ICRP, 2007) recommends that for radiological protection purposes the skin dose should be evaluated to the cells of the basal layer. The depth of these cells is often referred to in radiation protection as the skin thickness and ICRP recommend a value of 70 μm is used for routine skin dose assessment. For non-uniform exposures the ICRP recommend that this dose should be averaged over the most highly exposed area of 1 cm^2.

1180. External dose (skin) is thus calculated as:

$$Dose_{ext,skin} = G_{skin}^{Rn}.T_{skin}.A_{Rn}(t)$$

where:

- $Dose_{ext,skin}$ is the skin (organ) dose;
- G_{skin}^{Rn} is the point-source effective dose rate for radionuclide *Rn* in contact with the skin (Sv $hour^{-1}$ Bq^{-1}) from β and / or γ radiation as appropriate for that radionuclide, assuming a skin thickness of 70 μm (note that β dose rate factors for 40 μm were used in the calculations as no other data were available);
- $A_{Rn}(t)$ is the activity of the particle (Bq) at the time of exposure (t); and,
- T_{skin} is the exposure time for skin exposure (hours).

E.6.3.3. Results for selected radionuclides

1181. Table 173 gives the results of the dose assessments using the particle assessment tool for ten of the radionuclides considered in the ESC. This gives the dose from ingestion (assuming ICRP f1 values), external exposure and skin exposure arising from 1 MBq of each radionuclide, for intrusion at 60 years, and for intrusion at 300 years.

Table 173 Dose (mSv) from 1MBq on a particle, for two intrusion times

Radionuclide	Dose from ingestion at 60 y	Whole body dose at 60 y	Skin Dose at 60 y	Dose from ingestion at 300 y	Whole body dose at 300 y	Skin Dose at 300 y
Pa-231	$1.74\ 10^{3}$	$3.81\ 10^{-5}$	$5.85\ 10^{3}$	$1.91\ 10^{3}$	$3.79\ 10^{-5}$	$6.80\ 10^{3}$
Ra-226	$1.17\ 10^{3}$	$1.77\ 10^{-3}$	$8.47\ 10^{3}$	$1.06\ 10^{3}$	$1.60\ 10^{-3}$	$7.63\ 10^{3}$
Th-232	$1.03\ 10^{3}$	$1.43\ 10^{-3}$	$2.60\ 10^{2}$	$1.03\ 10^{3}$	$1.43\ 10^{-3}$	$2.60\ 10^{2}$
Th-229	$5.96\ 10^{2}$	$3.81\ 10^{-4}$	$8.58\ 10^{3}$	$5.83\ 10^{2}$	$3.72\ 10^{-4}$	$8.39\ 10^{3}$
Sn-126	$7.13\ 10^{0}$	0	$4.67\ 10^{3}$	$7.13\ 10^{0}$	0	$4.67\ 10^{3}$
Pu-239	$2.50\ 10^{2}$	$1.30\ 10^{-7}$	$1.43\ 10^{0}$	$2.48\ 10^{2}$	$1.29\ 10^{-7}$	$1.42\ 10^{0}$
Pu-240	$2.48\ 10^{2}$	$1.38\ 10^{-6}$	$2.58\ 10^{0}$	$2.42\ 10^{2}$	$1.34\ 10^{-6}$	$2.52\ 10^{0}$
Th-230	$2.41\ 10^{2}$	$4.90\ 10^{-5}$	$3.31\ 10^{2}$	$3.56\ 10^{2}$	$2.24\ 10^{-4}$	$1.17\ 10^{3}$
Pu-242	$2.40\ 10^{2}$	$1.15\ 10^{-6}$	$3.07\ 10^{0}$	$2.40\ 10^{2}$	$1.15\ 10^{-6}$	$3.07\ 10^{0}$
U-238	$4.79\ 10^{1}$	$3.51\ 10^{-4}$	$3.83\ 10^{3}$	$4.80\ 10^{1}$	$3.51\ 10^{-4}$	$3.83\ 10^{3}$

E.6.3.4. Waste acceptance

1182. It is not possible to determine generic waste acceptance criteria for waste containing particles as the characteristics of the particle (e.g. nuclides, size, solubility) will be specific to the consignment. Therefore waste containing particles will be considered on a case by case basis.

1183. Decisions regarding acceptance for waste containing high activity particles can be made by comparison of the results of dose calculations for the activity on the particle with the NS-GRA intrusion dose guidance level. The ingestion dose and external (whole body) dose are therefore compared to the annual dose guidance level of 3 to 20 mSv. The exposure is regarded as a 'one-off' event and hence the appropriate dose guidance value would lie towards the upper end of the range cited.The dose from contact with the skin is compared with the 50 mSv annual dose limit for the equivalent dose to skin for members of the public, as specified in the NS-GRA. Wastes that do not meet these dose guidance levels are not accepted without specific approval from the Environment Agency. Demonstration that the disposal route adopted represents BAT would also be required. The activity limits are given in Table 174.

Table 174 Activity limit on a particle, for two intrusion times

Radionuclide	Particle activity limit (MBq) at 60 y	Limiting dose criterion$ at 60 y	Particle activity limit (MBq) at 300 y	Limiting dose criterion$ at 300 y
Pa-231	$1.72\ 10^{-3}$	Ingestion	$1.57\ 10^{-3}$	Ingestion
Ra-226	$2.56\ 10^{-3}$	Ingestion	$2.84\ 10^{-3}$	Ingestion
Th-232	$2.90\ 10^{-3}$	Ingestion	$2.90\ 10^{-3}$	Ingestion
Th-229	$5.03\ 10^{-3}$	Ingestion	$5.15\ 10^{-3}$	Ingestion
Sn-126	$1.07\ 10^{-2}$	Skin	$8.44\ 10^{-3}$	Ingestion
Pu-239	$1.20\ 10^{-2}$	Ingestion	$1.07\ 10^{-2}$	Skin
Pu-240	$1.21\ 10^{-2}$	Ingestion	$1.21\ 10^{-2}$	Ingestion
Th-230	$1.25\ 10^{-2}$	Ingestion	$1.24\ 10^{-2}$	Ingestion
Pu-242	$1.25\ 10^{-2}$	Ingestion	$1.25\ 10^{-2}$	Ingestion
U-238	$1.31\ 10^{-2}$	Skin	$1.31\ 10^{-2}$	Skin

$ Ingestion and whole body dose criterion of 3 mSv, for the skin it is 50 mSv

1184. The waste acceptance procedure is therefore described by the following steps:

- Use the particle assessment spreadsheet tool to assess the dose from the type of particle in the waste.
- Identify the package and consignment activity concentration limits relevant to the nuclides in the package.
- For ESC radionuclides where the ingestion dose is less than 3 mSv, the external dose to whole body is less than 3 mSv, the skin dose due to external exposure is less than 50 mSv, and the package and consignment meet their respective activity concentration limits, a consignment of particles may be disposed of without consulting the Environment Agency.
- Where the ingestion dose is between 3 mSv and 20 mSv or the external dose to whole body is between 3 mSv and 20 mSv, then the Environment Agency should be consulted.
- Where the ingestion dose is above 20 mSv or the external dose to whole body is above 20 mSv or the skin dose due to external exposure is above 50 mSv the consignment would not be acceptable for disposal.
- For radionuclides not considered in the ESC or where alternative f1 values or low solubility are proposed then the Environment Agency should be consulted.

E.6.4. Exposure to particles following site erosion

1185. Exposure of members of the public to particles that have become accessible on the bank of the estuary as a result of erosion is compared to the risk guidance level. The relevant pathways are external exposure through a particle becoming trapped in a fingernail or toenail and inadvertent ingestion. The dose if encountered is the same as that calculated in Section E.6.3. The probability of encounter for a walker on the estuary bank is obtained from the number of particles per gram of eroded material on the estuary bank and the time spent on the estuary bank.

1186. There are two important pathways ingestion and skin exposure. The results in Table 174 indicate that some radionuclides are limited by ingestion, and some by skin exposure. This enables simple estimates of the risk arising from particles eroded onto the estuary bank to be made.

1187. The risk from inadvertent ingestion of a particle that meets the acceptance criterion described in paragraph 1184 is given in Table 175. The total risk from particles on the bank is this risk multiplied by the probability of encounter.

Table 175 Risk calculations for inadvertent ingestion following site erosion

Parameter	Value
Dose criterion for ingestion used to set acceptance criterion (mSv)	3
Health risk from ingestion of particle meeting dose criterion (y^{-1})	$1.8\ 10^{-4}$
Probability of encounter that would meet 1E-6 y^{-1}	$5.6\ 10^{-3}$
Quantity of material inadvertently ingested by walker (g)	0.9
Number of particles per 1 kg of eroded material to meet risk guidance level	6
Number of particles per kg of disposed radioactive waste to meet risk guidance level (assuming landfill contains 20% radioactive waste)	30

1188. The risk guidance level of 10^{-6} y^{-1} is met if there are 6 or fewer particles that meet the acceptance criterion (i.e. give a dose of 3 mSv if ingested) per kg of eroded material. An expected average landfill content of 20% radioactive waste indicates that up to 30 particles per kg of radioactive waste will meet the risk criterion. Since particles are not expected to be in every consignment, this pathway is not likely to be limiting.

1189. The risk from skin exposure to a particle that meets the skin dose criterion is given in Table 176. The risk guidance level is met even if it is assumed that all the activity present on the estuary bank after erosion is radioactive waste that is in the form of particles that meet the skin dose criterion. Hence this pathway is not limiting.

Table 176 Risk calculations for skin exposure following site erosion

Parameter	Value
Dose criterion for skin (equivalent dose) used to set acceptance criterion (mSv)	50
Health risk from skin exposure to 1mm particle meeting dose criterion (y^{-1})	$3\ 10^{-8}$
Quantity of material under fingernail ($g\ h^{-1}$)	0.12
Quantity of material under fingernail for walker(g)	0.9
Number of particles in 1 kg waste assuming it is all in the form of particles	20
Probability of particle under fingernail for walker (y^{-1})	0.175
Overall risk to health assuming all eroded material is radioactive waste and in the form of particles (y^{-1})	$5\ 10^{-9}$

1190. Hence waste acceptance criteria that restrict the activity per particle on the basis of the dose if encountered will ensure that the public is protected if the site erodes and waste becomes exposed on the estuary bank.

E.6.5. Radionuclide activity concentrations

1191. The assessments undertaken to support the ESC for Port Clarence are used to calculate activity concentration limits for disposed waste. Our approach uses an intrusion scenario, the dropped bag scenario, and worker doses after emplacement and during handling[3] to derive an activity concentration per consignment for each radionuclide. To simplify operational control these calculated activities are then grouped and the values used are 100, 200, 500, 1000 and 4000 Bq g^{-1}. The lowest activity is 100 Bq g^{-1} and the only radionuclides assigned to this group are Pu-239 and Am-241, corresponding to the levels specified in the LLW exclusion NEA convention (NEA, 2017).

1192. The ESC for the ENRMF (Eden NE, 2015a) used an activity concentration averaged over a consignment of 200 Bq g^{-1} for all waste and used the Trial pit excavator scenario was used to show that averaging over 10 t of waste allowed an upper limit of 1000 Bq g^{-1} in 10% of the waste consignment. This intrusion scenario has been used to calculate the activity concentrations in waste (per consignment and per package) that could be disposed at Port Clarence and not exceed the dose criterion for intrusion.

1193. Radionuclides are assigned to activity concentration groupings as follows:

- Beta gamma emitting radionuclides assigned a maximum value per consignment of 10,000 Bq g^{-1} – calculated activity concentrations greater than or equal to 10,000 Bq g^{-1} are included in the group (27 radionuclides);
- Beta gamma emitting radionuclides assigned a maximum value per consignment of 5000 Bq g^{-1} – calculated activity concentrations greater than or equal to 5000 Bq g^{-1} are included in the group (0 radionuclides so group not used);
- radionuclides assigned a maximum value per consignment of 2000 Bq g^{-1} – calculated activity concentrations greater than or equal to 2000 Bq g^{-1} not otherwise allocated to a group are included in the group (18 radionuclides);
- radionuclides assigned a maximum value per consignment of 1000 Bq g^{-1} – calculated concentrations less than 2000 Bq g^{-1} and greater than or equal to 1000 Bq g^{-1} (8 radionuclides);
- radionuclides assigned a maximum value per consignment of 500 Bq g^{-1} – calculated concentrations less than 1000 Bq g^{-1} and greater than or equal to 500 Bq g^{-1} (6 radionuclides);
- radionuclides assigned a maximum value per consignment of 200 Bq g^{-1} – calculated concentrations less than 500 Bq g^{-1} and greater than or equal to 200 Bq g^{-1} (10 radionuclides); and,

[3] In the cases where the total worker dose from handling and after emplacement exceeded 2 mSv y^{-1} after the activity concentration limit based on the emplacement scenario was applied, a further activity concentration limit was applied using the activity concentration limit based on the handling scenario. 2 mSv y^{-1} was used as the approach to calculate doses from the package was cautious. In practice the 10 µSv h^{-1} dose rate limit on packages will prevent doses exceeding 1 mSv y^{-1}.

- radionuclides assigned a maximum value pera consignment of 100 Bq g^{-1} – calculated concentrations less than 200 Bq g^{-1} and greater than or equal to 100 Bq g^{-1} (2 radionuclides).

1194. The calculated concentrations are presented in Table 177 and the groupings are summarized in Table 178.

Table 177 Calculated concentrations for disposal of packaged waste (Bq g^{-1})

Radionuclide	Worker emplacement	Worker handling	Dropped load (Bag) public	Trial pit	Paris Convention	Limiting concentration
H-3			$6.35\ 10^{9}$	$1.06\ 10^{10}$	10000	10000
C-14			$3.74\ 10^{8}$	$1.07\ 10^{7}$	10000	10000
Cl-36			$2.44\ 10^{8}$	$1.46\ 10^{6}$		10000
Ca-41			$1.06\ 10^{10}$	$5.99\ 10^{7}$		10000
Mn-54	$1.32\ 10^{3}$	$5.57\ 10^{2}$	$1.02\ 10^{9}$	$1.28\ 10^{24}$		500
Fe-55			$1.99\ 10^{9}$	$1.22\ 10^{14}$		10000
Co-60	$2.64\ 10^{2}$		$7.39\ 10^{7}$	$8.01\ 10^{5}$	200	200
Ni-59			$4.24\ 10^{9}$	$7.82\ 10^{7}$		10000
Ni-63			$1.48\ 10^{9}$	$7.95\ 10^{7}$		10000
Zn-65	$1.37\ 10^{3}$	$7.72\ 10^{2}$	$6.35\ 10^{8}$	$1.46\ 10^{30}$		500
Se-79			$3.18\ 10^{8}$	$3.36\ 10^{6}$		10000
Sr-90			$1.55\ 10^{7}$	$4.93\ 10^{5}$	200	200
Mo-93	$8.07\ 10^{8}$		$1.10\ 10^{9}$	$2.57\ 10^{6}$		10000
Zr-93	$5.36\ 10^{9}$		$2.54\ 10^{8}$	$4.55\ 10^{6}$		10000
Nb-93m	$5.32\ 10^{9}$		$9.77\ 10^{8}$	$3.27\ 10^{8}$		10000
Nb-94	$7.64\ 10^{2}$	$2.98\ 10^{2}$	$5.29\ 10^{7}$	$5.04\ 10^{2}$		200
Tc-99	$8.04\ 10^{12}$		$1.72\ 10^{8}$	$5.89\ 10^{6}$	200	200
Ru-106	$1.23\ 10^{4}$	$3.78\ 10^{3}$	$2.76\ 10^{7}$	$1.70\ 10^{21}$		2000
Ag-108m	$1.08\ 10^{3}$	$3.11\ 10^{2}$	$7.30\ 10^{7}$	$5.57\ 10^{2}$		200
Ag-110m	$3.64\ 10^{2}$		$1.55\ 10^{8}$	$7.22\ 10^{28}$		200
Cd-109	$2.76\ 10^{9}$		$1.72\ 10^{8}$	$6.12\ 10^{19}$		10000
Sb-125	$4.93\ 10^{3}$		$1.55\ 10^{8}$	$6.98\ 10^{9}$		2000
Sn-119m	$6.18\ 10^{49}$		$8.04\ 10^{8}$	$4.79\ 10^{28}$		10000
Sn-123	$1.15\ 10^{5}$		$2.05\ 10^{8}$	$1.08\ 10^{56}$		10000
Sn-126	$7.87\ 10^{2}$		$6.21\ 10^{7}$	$4.11\ 10^{2}$		200
Te-127m	$4.10\ 10^{7}$		$1.93\ 10^{8}$	$1.08\ 10^{68}$		10000
I-129	$5.16\ 10^{21}$		$7.39\ 10^{7}$	$8.32\ 10^{4}$		10000
Ba-133	$1.40\ 10^{4}$	$1.92\ 10^{3}$	$2.19\ 10^{8}$	$1.28\ 10^{5}$		1000
Cs-134	$8.85\ 10^{2}$	$3.09\ 10^{2}$	$1.01\ 10^{8}$	$2.86\ 10^{11}$		200
Cs-135			$2.65\ 10^{8}$	$4.26\ 10^{6}$		10000
Cs-137	$2.52\ 10^{3}$		$6.35\ 10^{7}$	$5.66\ 10^{3}$	200	200
Ce-144	$1.86\ 10^{6}$		$2.35\ 10^{7}$	$9.15\ 10^{27}$		10000
Pm-147	$6.54\ 10^{11}$		$3.53\ 10^{8}$	$9.93\ 10^{13}$		10000
Sm-147			$2.76\ 10^{5}$	$1.99\ 10^{4}$		2000
Sm-151	$1.01\ 10^{70}$		$6.35\ 10^{8}$	$5.66\ 10^{7}$		10000
Eu-152	$8.43\ 10^{2}$		$6.35\ 10^{7}$	$1.50\ 10^{4}$		500
Eu-154	$7.32\ 10^{2}$		$4.24\ 10^{7}$	$8.01\ 10^{4}$		500
Eu-155	$1.88\ 10^{7}$		$2.76\ 10^{8}$	$1.66\ 10^{8}$		10000
Gd-153	$2.61\ 10^{7}$		$5.29\ 10^{8}$	$5.50\ 10^{31}$		10000

Radionuclide	Worker emplacement	Worker handling	Dropped load (Bag) public	Trial pit	Paris Convention	Limiting concentration
Pb-210	$1.13\ 10^{6}$		$1.97\ 10^{5}$	$3.15\ 10^{4}$		10000
Po-210	$1.32\ 10^{8}$		$4.54\ 10^{5}$	$3.87\ 10^{51}$		2000
Ra-226	$3.36\ 10^{2}$		$1.03\ 10^{5}$	$3.98\ 10^{2}$		200
Ra-228	$1.47\ 10^{2}$		$3.05\ 10^{4}$	$3.76\ 10^{5}$		100
Ac-227	$5.46\ 10^{3}$		$3.84\ 10^{3}$	$2.13\ 10^{3}$		2000
Th-228	$2.34\ 10^{2}$		$3.98\ 10^{4}$	$1.21\ 10^{12}$		200
Th-229	$2.98\ 10^{3}$		$1.14\ 10^{4}$	$6.26\ 10^{2}$		500
Th-230	$5.85\ 10^{8}$		$3.18\ 10^{4}$	$1.77\ 10^{3}$		1000
Th-232	$3.95\ 10^{2}$		$1.48\ 10^{4}$	$4.66\ 10^{2}$		200
Pa-231	$1.75\ 10^{5}$		$2.76\ 10^{4}$	$2.89\ 10^{2}$		200
U-232	$2.87\ 10^{2}$		$2.47\ 10^{4}$	$4.28\ 10^{3}$		200
U-233	$7.82\ 10^{6}$		$2.12\ 10^{5}$	$1.68\ 10^{4}$		2000
U-234	$1.24\ 10^{10}$		$2.19\ 10^{5}$	$2.02\ 10^{4}$		2000
U-235	$1.58\ 10^{5}$		$2.44\ 10^{5}$	$5.15\ 10^{3}$		2000
U-236	$1.05\ 10^{10}$		$2.35\ 10^{5}$	$2.19\ 10^{4}$		2000
U-238	$6.26\ 10^{7}$		$2.54\ 10^{5}$	$2.34\ 10^{4}$	200	200
Np-237	$3.56\ 10^{4}$		$6.83\ 10^{4}$	$3.77\ 10^{3}$		2000
Pu-238	$2.65\ 10^{11}$		$3.34\ 10^{4}$	$2.94\ 10^{3}$		2000
Pu-239	$4.34\ 10^{8}$		$3.18\ 10^{4}$	$1.68\ 10^{3}$	100	100
Pu-240	$2.16\ 10^{12}$		$3.18\ 10^{4}$	$1.69\ 10^{3}$		1000
Pu-241	$9.02\ 10^{8}$		$2.19\ 10^{6}$	$6.69\ 10^{4}$		10000
Pu-242	$6.78\ 10^{13}$		$3.34\ 10^{4}$	$1.82\ 10^{3}$		1000
Pu-244	$5.40\ 10^{3}$		$3.34\ 10^{4}$	$1.81\ 10^{3}$		1000
Am-241	$3.65\ 10^{10}$		$3.53\ 10^{4}$	$2.26\ 10^{3}$	100	100
Am-242m	$9.88\ 10^{4}$		$3.18\ 10^{4}$	$1.65\ 10^{3}$		1000
Am-243	$1.40\ 10^{5}$		$3.74\ 10^{4}$	$1.98\ 10^{3}$		1000
Cm-242	$4.63\ 10^{10}$		$3.03\ 10^{5}$	$5.75\ 10^{5}$		2000
Cm-243	$2.15\ 10^{5}$		$4.22\ 10^{4}$	$8.94\ 10^{3}$		2000
Cm-244	$3.69\ 10^{52}$		$4.89\ 10^{4}$	$3.34\ 10^{4}$		2000
Cm-245	$2.10\ 10^{6}$		$3.53\ 10^{4}$	$1.75\ 10^{3}$		1000
Cm-246	$8.77\ 10^{23}$		$3.53\ 10^{4}$	$2.07\ 10^{3}$		2000
Cm-248	$1.98\ 10^{49}$		$9.77\ 10^{3}$	$5.58\ 10^{2}$		500

Table 178 Activity concentrations used to limit disposal of LLW at Port Clarence

Assigned activity concentration (Bq g^{-1})	Radionuclides
100	Ra-228, Pu-239*, Am-241*
200	Co-60*, Sr-90*, Nb-94, Tc-99*, Ag-108m, Ag-110m, Sn-126, Cs-134, Cs-137*, Ra-226, Th-228, Th-232, Pa-231, U-232, U-238*
500	Mn-54, Zn-65, Eu-152, Eu-154, Th-229, Cm-248
1,000	Ba-133, Th-230, Pu-240, Pu-242, Pu-244, Am-242m, Am-243, Cm-245
2,000	Ru-106, Sb-125, Sm-147, Po-210, Ac-227, U-233, U-234, U-235, U-236, Np-237, Pu-238, Cm-242, Cm-243, Cm-244, Cm-246
10,000	H-3*, C-14*, Cl-36, Ca-41, Fe-55, Ni-59, Ni-63, Se-79, Mo-93, Zr-93, Nb-93m, Cd-109, Sn-119m, Sn-123, Te-127m, I-129, Cs-135, Ce-144, Pm-147, Sm-151, Eu-155, Gd-153, Pb-210, Pu-241

Note: * radionuclides listed in the Paris Convention LLW exclusion. The assigned values are equal to the values listed in the LLW exclusion.

1195. The heterogeneity of waste within a consignment is also considered. For each radionuclide the ratio of the calculated concentration to the assigned concentration provides an indication of the flexibility that could be applied and meet the dose criteria. The analysis performed for the ENRMF (Eden NE, 2015a) demonstrated that a ratio of 5 for the activity concentration in a package compared to the average in the consignment would also meet the dose criterion. Hence, ENRMF applied an average of 200 Bq g^{-1} in a consignment with a maximum of 1000 Bq g^{-1} in a package comprising part of a larger consignment, for all radionuclides.

1196. The upper limit for LLW is 4000 Bq/g for alpha emitting radionuclides and 12000 Bq/g for beta or gamma emitting radionuclides. Hence, these are the upper limits per package. However, the waste to be disposed of is not expected to be at the upper end of LLW (see Appendix D).

1197. A ratio of 5 for the activity in a package compared to the average in the consignment will be applied at Port Clarence for wastes with an assigned activity concentration of 100 Bq/g and 200 Bq/g. A ratio of 3 will be applied to wastes with an average concentration of 500 Bq/g and 1000Bq/g per consignment. A ratio of 2 will be applied to wastes with an average concentration of 2000 Bq/g per consignment and a ratio of 1.2 to wastes with an average concentration of 10,000 Bq/g per consignment. Hence, every package will comply with the LLW activity concentration limit. This gives the following activity concentration levels in Table 179.

Table 179 Maximum average activity in consignment and in a package

Group	Maximum activity concentration averaged over a consignment (Bq g^{-1})	Maximum activity concentration in a package that is part of a consignment (Bq g^{-1})
1	100	500
2	200	1000
3	500	1500
4	1000	3000
5	4000	10,000

1198. The activity limits have also been calculated for loose tipping of waste (see Table 68) and these are used to group radionuclides by assigning values of 5, 10, 50 and 100 Bq g^{-1} for loose tipped waste (Table 180).

Table 180 Activity concentrations used to limit tipping of loose LLW at Port Clarence

Assigned activity concentration (Bq g^{-1})	Radionuclides
5	Ac-227, Th-229, Th-232, Pa-231, Pu-239, Pu-240, Cm-248
10	Ra-228, Th-228, Th-230, U-232, Np-237, Pu-238, Pu-242, Pu-244, Am-241, Am-242m, Am-243, Cm-243, Cm-244, Cm-245, Cm-246
50	Ra-226
100	H-3, C-14, Cl-36, Ca-41, Mn-54, Fe-55, Co-60, Ni-59, Ni-63, Zn-65, Se-79, Sr-90, Mo-93, Zr-93, Nb-93m, Nb-94, Tc-99, Ru-106, Ag-108m, Ag-110m, Cd-109, Sb-125, Sn-119m, Sn-123, Sn-126, Te-127m, I-129, Ba-133, Cs-134, Cs-135, Cs-137, Ce-144, Pm-147, Sm-147, Sm-151, Eu-152, Eu-154, Eu-155, Gd-153, Pb-210, Po-210, U-233, U-234, U-235, U-236, U-238, Pu-241, Cm-242

E.7. Environmental radioactivity {R9}

E.7.1. Exposure to wildlife from all sources

1199. A radiological assessment of the potential effects on non-human biota (NHB) from the disposal of LLW at Port Clarence has been undertaken using the ERICA (Environmental Risk from Ionising Contaminants: Assessment and Management) Assessment Tool. The ERICA tool is a software system that has a structure based upon the tiered ERICA Integrated Approach to assessing the radiological risk to terrestrial, freshwater and marine biota.

1200. ERICA was developed under an EC funded international programme. Further details are available at: http://www.erica-tool.com/. The ERICA assessment tool is updated periodically. Version 1.3.1.33 was used in this assessment, which includes updated concentration rations and corresponding environmental media concentration limits (EMCLs).

1201. The ERICA assessment tool allows consideration of three ecosystems: terrestrial, freshwater and marine. All three ecosystems are applicable to the environment surrounding Port Clarence. It is assumed that the estuary close to the site can be treated as a marine environment. Within these ecosystems, the ERICA tool considers a range of organisms and wildlife groups as shown in Table 181.

Table 181 Wildlife groups considered in the ERICA tool

Terrestrial	Freshwater	Marine
Amphibian	Amphibian	Benthic Fish
Annelid	Benthic fish	Bird
Anthropod - detritivorous	Bird	Crustacean
Bird	Crustacean	Macroalgae
Flying insects	Insect larvae	Mammal
Grasses and herbs	Mammal	Mollusc – bivalve
Lichen and bryophytes	Mollusc – bivalve	Pelagic fish
Mammal large	Mollusc – gastropod	Phytoplankton
Mammal small – burrowing	Pelagic fish	Polychaete worm
Mollusc - gastropod	Phytoplankton	Reptile
Reptile	Reptile	Sea anemones and true coral
Shrub	Vascular plant	Vascular plant
Tree	Zooplankton	Zooplankton

1202. During the operational and active management phases, radioactivity could be released to the biosphere as gas (e.g. very low gas production rates may result in C-14 labelled carbon dioxide or tritiated hydrogen gas), or in discharges from leachate treatment facilities. After the period of authorisation, the majority of releases of radioactivity are likely to be associated with groundwater or as a result of intrusion into the waste.

1203. Input data for the non-human biota dose assessment are radioactivity concentrations in soil and air (terrestrial ecosystem assessment) and water or sediment (freshwater and marine ecosystem assessment). The activity concentrations of radionuclides in soil and water are calculated using the same approaches underlying the dose calculations to the public.

1204. The impact on terrestrial animals that dig into the waste is also considered, based on activity concentrations in the waste.

E.7.2. The ERICA assessment tool

1205. The Tool guides the user through the assessment process, recording information and decisions and allowing the necessary calculations to be performed to estimate risks to selected animals and plants. The tiered approach offers increasing opportunities to introduce site specific factors. For the NHB assessments we have used ERICA Version 1.3.1.33, updated 28.05.2019.

1206. Tier 1 assessments are based on media concentration and use pre-calculated environmental media concentration limits (EMCLs) to estimate risk quotients. Tier 2 calculates dose rates but allows the user to examine and edit most of the parameters used in the calculation including concentration ratios, distribution coefficients, percentage dry weight soil or sediment, dose conversion coefficients, radiation weighting factors and occupancy factors. The user can also input biota whole-body activity concentrations in Tier 2 if available rather than rely upon concentration ratios. Tier 3 offers the same flexibility as Tier 2 but allows the option to run the assessment probabilistically if the underling parameter probability distribution functions are defined.

1207. This assessment has been undertaken using a Tier 1 approach for a marine ecosystem (representing the estuary) and Tier 2 approach for the ERICA terrestrial and freshwater ecosystems. Dose to burrowing animals in the waste cells after the Period of Authorisation was also considered. It should be noted that the philosophy behind a landfill site is to concentrate and contain the waste to protect the environment. The environment inside the landfill is not part of the environment that is to be protected.

1208. Within the terrestrial, marine and freshwater ecosystems, the ERICA Tool considers a range of wildlife groups considered to be representative (see Table 181). The organisms are intended to be interpreted in a generic fashion rather than as individual species, although the categorisation strays across several taxonomic levels or groupings. Apart from bird eggs, life cycle stages are not addressed specifically and the nomenclature adopted indicates that organism types have been identified based on a number of considerations such as food source (detritivorous invertebrates), habitat (flying insects), size (rat and deer, both representing mammals) etc. Specifically, the organism types do not represent individual species. Thus the ‘rat’ and ‘deer’ represent small and large mammals respectively and should not be identified as Roe deer or Red deer (Britain’s two native deer species) or Brown rat (the most common, if not strictly native, rat in the UK).

1209. Similar, but not identical, ICRP Reference Animals and Plants (RAPs) have been adopted (ICRP, 2008). A RAP is defined by ICRP as:

> ‘a hypothetical entity, with the assumed basic biological characteristics of a particular type of animal or plant, as described to the generality of the taxonomic level of family, with defined anatomical, physiological, and life-history properties,

that can be used for the purposes of relating exposure to dose, and dose to effects, for that type of living organism'.

1210. It is considered that the range of organism types represented within the ERICA assessment tool is sufficiently broad to characterise the reference ecosystems.

E.7.3. Screening levels

1211. Different approaches are available to derive numerical benchmark values for protection of NHB. A detailed explanation and proposed framework has been proposed by Jackson and co-workers [(Jackson, et al., 2014), (Smith, et al., 2010) and (Robinson, et al., 2010)]. A number of national and international studies have identified screening criteria, although consistency between countries has not been achieved (Copplestone, et al., 2010). For the present purposes, two screening values for protection of NHB have gained relatively widespread application.

1212. The EC ERICA and PROTECT derived screening value of 10 µGy h^{-1} above background is generally recognized in the UK and Europe. Although concerns may be raised as to whether this value is below natural variability in background exposures, for example (Brown, et al., 2004) indicated that wildlife might receive up to 60 µGy h^{-1} from natural sources in European ecosystems, it does have a demonstrable provenance, being based on the effects database (FREDERICA) developed within the EC FASSET and ERICA programmes (Copplestone, et al., 2008). It also has a clear definition, representing the dose rate at which 95% of species will not experience more than a 10% change in the observed effect, relative to a control group (this is termed HDR_5).

1213. The FREDERICA database is available online from www.frederica-online.org. The FREDERICA database contains over 1500 references and contains 29,400 data entries. Summary information is available on the effects of ionising radiation on different wildlife groups under seven umbrella endpoints: mutation, morbidity, reproductive capacity, mortality, stimulation, adaptation and ecological fitness. The database can be updated online.

1214. Some organisms, e.g. mammals, are more radiosensitive than others. The EU PROTECT (European Commission, 2008) project, which compares different available screening values, proposes first screening values of 2 µGy h^{-1} for vertebrates, 200 µGy h^{-1} for invertebrates and 70 µGy h^{-1} for plants. In America and Canada, an alternative approach is typically adopted, following the review of available effects data by (UNSCEAR, 2011) [including reference to previous studies by both (IAEA, 1992) and (UNSCEAR, 1996)], who concluded that, "*chronic dose rates of less than 0.1 mGy/h to the most exposed individuals would be unlikely to have significant effects on most terrestrial communities and chronic dose rates of less than 0.4 mGy/h to any individual in aquatic populations would be unlikely to have any detrimental effect at the population level*". This is also consistent with an evaluation of the FREDERICA database for plants, fish and mammals by (Real, et al., 2004), who noted that: "*the reviewed effects data give few indications for readily observable effects at chronic dose rates below 100 µGy/h*". Indeed below 1000 µGy h^{-1} there appears to be little evidence for irreversible impairment, although the general paucity of the database led (Real, et al., 2004) to give a cautionary note when seeking to establish environmentally ‘safe levels’ of radiation exposure.

1215. The Environment Agency for England and Wales currently recognise a 40 µGy h^{-1} “regulatory action level” such that, if the dose rates predicted to wildlife inhabiting a particular conservation site exceed 40 µGy h^{-1} then the regulators need to consider possible action, although again, this is not a ‘limit’ and following consideration no action may be required (Environment Agency, 2009). This action level considers all permitted discharges that might affect the conservation sites. It is unlikely that other sites have permitted radioactive discharges that could affect the environment local to Port Clarence.

1216. The current EA ‘regulatory action level’ was defined on the basis of the FASSET biological effects work that concluded that no adverse effects would be expected on populations at dose rates below 100 µGy h^{-1}, as noted in the preceding text. This was used in combination with a generic background dose rate for European ecosystems of 50 µGy h^{-1} and a safety margin of 10 µGy h^{-1} to account for the background dose rate not being specific to the UK. Below this dose rate, the Environment Agency currently considers that adverse impact is unlikely.

E.7.4. ERICA assessment for a Marine Ecosystem

1217. An assessment of a marine ecosystem was considered to be representative of the estuary close to the Port Clarence site.

1218. The limiting environmental concentrations determined from the ERICA assessment tool are determined for each radionuclide based on a screening dose rate of 10 µGy h^{-1}. This is considered to represent a conservative approach.

1219. It should be noted that the calculated dose rate for the same environmental concentration differs between organisms (e.g. as a function of the concentration factor applied). Therefore, the limiting concentration does not necessarily apply to all organism types. Rather, within the Tier 1 assessment, the limiting concentration is based on the single organism type that has the highest dose rate per unit radioactivity in the relevant environmental medium.

1220. Not all radionuclides considered in the Port Clarence assessment are available in Tier 1 of the ERICA assessment tool. There are 28 radionuclides not included in the Tier 1 list of radionuclides within ERICA. Of these, Ac-227, Ag-108m, Am-242m, Am-243, Ba-133, Cm-245, Cm-246, Cm-248, Eu-155, Gd-153, Mo-93, Nb-93m, Pm-147, Pu-244, Sm-147, Sm-151, Sn-199m, Sn-123, Sn-126, Te-127m, U-236 and Zr-93 represent a small fraction in the UK Radioactive Waste Inventory (see Table 45 for the LLW composition); U-232 and U-233 are minor contributors to the U-vector; Th-229 is a minor contributor to the Th-vector; and, Pu-242 is a minor contributor to the Pu-alpha-vector. The remaining radionuclide, Ca-41, was therefore the only one considered further, in a Tier 2 assessment.

1221. Table 182 lists the radionuclide specific limiting environmental activity concentrations in the marine ecosystem, corresponding to 10 µGy h^{-1}. A Tier 2 assessment was undertaken to determine a limiting concentration in water and sediment for Ca-41. The limiting concentration in a marine ecosystem (corresponding to 10 µGy h^{-1}) was found to be 1.26 10^4 Bq l^{-1} for water and 1.68 10^7 Bq kg^{-1} (dry weight) for sediment.

Table 182 Radionuclide specific limiting environmental activity concentrations in the marine ecosystem, corresponding to 10 µGy h^{-1}

Radionuclide	Limiting concentrations in marine ecosystem			
	Water (Bq l^{-1})	limiting organism	Sediment (Bq kg^{-1} dry weight)	limiting organism
H-3	3.94 10^5	Mollusc - bivalve	1.10 10^5	Phytoplankton
C-14	1.16 10^1	Zooplankton	3.27 10^3	Zooplankton
Cl-36	3.76 10^4	Vascular plant	1.96 10^2	Phytoplankton
Ca-41	1.26 10^4	Phytoplankton	1.68 10^7	Phytoplankton
Mn-54	1.05 10^{-3}	Polychaete worm	2.13 10^4	Polychaete worm
Co-60	2.30 10^{-3}	Polychaete worm	6.99 10^3	Polychaete worm
Ni-59	4.37 10^1	Sea anemones & True coral	3.15 10^5	Sea anemones & True coral
Ni-63	3.32 10^1	Sea anemones & True coral	2.33 10^5	Sea anemones & True coral
Zn-65	4.27 10^{-2}	Polychaete worm	4.15 10^3	Crustacean
Se-79	1.02 10^1	Crustacean	8.47 10^3	Crustacean
Sr-90	2.53 10^1	Reptile	7.46 10^1	Mammal
Nb-94	1.41 10^{-3}	Polychaete worm	1.12 10^4	Polychaete worm
Tc-99	1.07 10^0	Vascular plant	3.25 10^1	Vascular plant
Ru-106	7.35 10^{-2}	Polychaete worm	6.54 10^3	Zooplankton
Ag-110m	6.37 10^{-2}	Polychaete worm	4.85 10^2	Reptile
Cd-109	4.03 10^{-1}	Mollusc - bivalve	5.08 10^3	Mollusc - bivalve
Sb-125	1.94 10^0	Polychaete worm	1.18 10^3	Reptile
I-129	6.17 10^0	Polychaete worm	1.67 10^2	Polychaete worm
Cs-134	2.87 10^{-1}	Polychaete worm	1.07 10^4	Polychaete worm
Cs-135	1.59 10^2	Reptile	1.99 10^5	Bird
Cs-137	7.63 10^{-1}	Polychaete worm	2.48 10^4	Reptile
Ce-144	1.66 10^{-3}	Polychaete worm	4.90 10^4	Polychaete worm
Eu-152	8.33 10^{-4}	Polychaete worm	1.52 10^4	Polychaete worm
Eu-154	7.63 10^{-4}	Polychaete worm	1.37 10^4	Polychaete worm
Pb-210	6.41 10^{-2}	Phytoplankton	2.33 10^3	Phytoplankton
Po-210	3.79 10^{-4}	Polychaete worm	1.39 10^3	Polychaete worm
Ra-226	1.90 10^{-2}	Phytoplankton	1.81 10^1	Phytoplankton
Ra-228	8.93 10^{-1}	Polychaete worm	1.31 10^4	Polychaete worm
Th-228	2.59 10^{-5}	Phytoplankton	2.09 10^1	Phytoplankton
Th-230	1.81 10^{-4}	Phytoplankton	1.44 10^2	Phytoplankton
Th-232	2.12 10^{-4}	Phytoplankton	1.68 10^2	Phytoplankton
Pa-231	8.06 10^{-3}	Polychaete worm	1.54 10^5	Phytoplankton
U-234	1.96 10^{-1}	Polychaete worm	3.34 10^1	Polychaete worm
U-235	2.12 10^{-1}	Polychaete worm	3.62 10^1	Polychaete worm
U-238	2.29 10^{-1}	Polychaete worm	3.89 10^1	Polychaete worm

Radionuclide	Limiting concentrations in marine ecosystem			
	Water ($Bq\ l^{-1}$)	limiting organism	Sediment ($Bq\ kg^{-1}$ dry weight)	limiting organism
Np-237	$1.98\ 10^{-1}$	Sea anemones & True coral	$2.74\ 10^{1}$	Reptile
Pu-238	$8.20\ 10^{-4}$	Phytoplankton	$2.45\ 10^{1}$	Phytoplankton
Pu-239	$8.70\ 10^{-4}$	Phytoplankton	$2.61\ 10^{1}$	Phytoplankton
Pu-240	$8.70\ 10^{-4}$	Phytoplankton	$2.61\ 10^{1}$	Phytoplankton
Pu-241	$3.19\ 10^{0}$	Phytoplankton	$9.62\ 10^{4}$	Phytoplankton
Am-241	$5.32\ 10^{-4}$	Phytoplankton	$2.79\ 10^{2}$	Phytoplankton
Cm-242	$4.22\ 10^{-4}$	Phytoplankton	$1.96\ 10^{2}$	Phytoplankton
Cm-243	$4.42\ 10^{-4}$	Phytoplankton	$2.06\ 10^{2}$	Phytoplankton
Cm-244	$4.44\ 10^{-4}$	Phytoplankton	$2.06\ 10^{2}$	Phytoplankton

1222. Peak activity concentrations of Tier 1 radionuclides and their Tier 1 daughters in seawater and sediment were taken from the output of the DORIS assessment for the groundwater release to estuary scenario, as discussed in paragraph 900, subsection E.4.4. The sea water and seabed sediment concentrations for each radionuclide were then scaled to account for the maximum inventory of each radionuclide. Activity concentrations for radionuclides that were ingrown through radioactive decay were calculated separately. The ERICA assessment tool was then used to calculate a risk quotient for each radionuclide, which is defined as the radionuclide specific activity concentration in a medium divided by the limiting activity concentration for that radionuclide and medium. If the risk quotient is higher than one, the dose rate to the most limiting organism exceeds the ERICA screening dose rate of 10 µGy h^{-1}. If the risk quotient is higher than 4, the dose rate to the most limiting organism exceeds the EA guidance dose rate of 40 µGy h^{-1}.

1223. Table 183 below summarises the results of the wildlife assessment for the estuary (marine) ecosystem.

Table 183 Radionuclide specific risk quotients for the estuarine ecosystem, based on a generic screening level of 10 µGy h^{-1}.

Radionuclide	Daughter	Marine ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
H-3		$3.90\ 10^{-20}$		Phytoplankton
C-14		$2.24\ 10^{-14}$		Zooplankton
Cl-36		$4.31\ 10^{-13}$		Phytoplankton
Ca-41		$6.78\ 10^{-16}$		Phytoplankton

Radionuclide	Daughter	Marine ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
Mn-54		0		
Co-60		0		
Ni-59		$5.65\ 10^{-15}$		Sea anemones & True coral
Ni-63		$4.53\ 10^{-29}$		Sea anemones & True coral
Zn-65		0		
Se-79		$3.93\ 10^{-14}$		Crustacean
Sr-90		$1.53\ 10^{-27}$		Mammal
Nb-94		$8.59\ 10^{-13}$		Polychaete worm
Tc-99		$1.19\ 10^{-10}$		Vascular plant
Ru-106		0		
Ag-110m		0		
Cd-109		0		
Sb-125		0		
I-129		$1.57\ 10^{-12}$		Polychaete worm
Cs-134		0		
Cs-135		$3.86\ 10^{-16}$		Reptile
Cs-137		0		
Ce-144		0		
Eu-152		0		
Eu-154		0		
Pb-210		0		
Pb-210	Po-210	0		
Po-210		0		
Ra-226		$1.03\ 10^{-23}$	$6.23\ 10^{-21}$	Phytoplankton
Ra-226	Pb-210	$3.89\ 10^{-24}$		Phytoplankton
Ra-226	Po-210	$6.21\ 10^{-21}$		Phytoplankton
Ra-228		0		
Ra-228	Th-228	0		
Th-228		0		
Th-230		$2.94\ 10^{-11}$	$1.55\ 10^{-10}$	Phytoplankton
Th-230	Ra-226	$2.12\ 10^{-13}$		Phytoplankton
Th-230	Pb-210	$7.86\ 10^{-14}$		Phytoplankton
Th-230	Po-210	$1.25\ 10^{-10}$		Phytoplankton
Th-232		$6.28\ 10^{-11}$	$5.74\ 10^{-10}$	Phytoplankton
Th-232	Ra-228	$1.13\ 10^{-14}$		Phytoplankton
Th-232	Th-228	$5.11\ 10^{-10}$		Phytoplankton
Pa-231		$1.54\ 10^{-13}$		Polychaete worm
U-234		$2.02\ 10^{-12}$	$2.44\ 10^{-10}$	Polychaete worm
U-234	Th-230	$4.59\ 10^{-11}$		Polychaete worm
U-234	Ra-226	$3.31\ 10^{-13}$		Polychaete worm
U-234	Pb-210	$1.23\ 10^{-13}$		Polychaete worm
U-234	Po-210	$1.95\ 10^{-10}$		Polychaete worm

Radionuclide	Daughter	Marine ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
U-235		$1.99\ 10^{-12}$	$3.86\ 10^{-12}$	Polychaete worm
U-235	Pa-231	$1.87\ 10^{-12}$		Polychaete worm
U-238		$1.84\ 10^{-12}$	$2.00\ 10^{-11}$	Polychaete worm
U-238	U-234	$1.34\ 10^{-13}$		Polychaete worm
U-238	Th-230	$3.44\ 10^{-12}$		Polychaete worm
U-238	Ra-226	$2.46\ 10^{-14}$		Polychaete worm
U-238	Pb-210	$9.11\ 10^{-15}$		Polychaete worm
U-238	Po-210	$1.45\ 10^{-11}$		Polychaete worm
Np-237		$1.19\ 10^{-11}$		Sea anemones & True coral
Pu-238		$1.58\ 10^{-32}$	$8.71\ 10^{-14}$	Phytoplankton
Pu-238	U-234	$7.24\ 10^{-16}$		Phytoplankton
Pu-238	Th-230	$1.64\ 10^{-14}$		Phytoplankton
Pu-238	Ra-226	$1.18\ 10^{-16}$		Phytoplankton
Pu-238	Pb-210	$4.38\ 10^{-17}$		Phytoplankton
Pu-238	Po-210	$6.98\ 10^{-14}$		Phytoplankton
Pu-239		$2.07\ 10^{-11}$	$2.07\ 10^{-11}$	Phytoplankton
Pu-239	U-235	$3.28\ 10^{-17}$		Phytoplankton
Pu-239	Pa-231	$4.56\ 10^{-17}$		Phytoplankton
Pu-240		$6.22\ 10^{-13}$	$6.22\ 10^{-13}$	Phytoplankton
Pu-240	Th-232	$8.69\ 10^{-20}$		Phytoplankton
Pu-240	Ra-228	$1.57\ 10^{-23}$		Phytoplankton
Pu-240	Th-228	$7.07\ 10^{-19}$		Phytoplankton
Pu-241		0	$7.65\ 10^{-17}$	
Pu-241	Am-241	$2.14\ 10^{-31}$		
Pu-241	Np-237	$7.65\ 10^{-17}$		
Am-241		$6.22\ 10^{-30}$	$2.31\ 10^{-15}$	Phytoplankton
Am-241	Np-237	$2.31\ 10^{-15}$		Phytoplankton
Cm-242		0	$4.42\ 10^{-16}$	
Cm-242	Pu-238	$1.15\ 10^{-34}$		
Cm-242	U-234	$3.68\ 10^{-18}$		
Cm-242	Th-230	$8.33\ 10^{-17}$		
Cm-242	Ra-226	$6.02\ 10^{-19}$		
Cm-242	Pb-210	$2.23\ 10^{-19}$		
Cm-242	Po-210	$3.55\ 10^{-16}$		
Cm-243		0	$2.50\ 10^{-14}$	
Cm-243	Pu-239	$2.50\ 10^{-14}$		
Cm-243	U-235	$3.96\ 10^{-20}$		
Cm-243	Pa-231	$5.51\ 10^{-20}$		
Cm-244		0	$1.72\ 10^{-15}$	
Cm-244	Pu-240	$1.72\ 10^{-15}$		

Radionuclide	Daughter	Marine ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
Cm-244	Th-232	$2.39\ 10^{-22}$		
Cm-244	Ra-228	$4.32\ 10^{-26}$		
Cm-244	Th-228	$1.95\ 10^{-21}$		

1224. All risk quotients are well below 1, therefore all non-human biota in the estuary are considered to be sufficiently protected.

E.7.5. ERICA assessment for a Freshwater Ecosystem

1225. There is an existing freshwater pond at the north-west corner of the site and further ponds are planned in the restoration plan, as shown in Figure 7. Radionuclides may be transferred to these bodies from the landfill by water that has become contaminated assuming that leachate from the landfill overtops the liner (the bathtubbing scenario). The distance of the pond from the landfill liner is 80 m.

1226. The activity concentration in the pond was cautiously assumed to be equal to the peak activity in water in topsoil for each radionuclide, calculated using the GoldSim model (see Subsection E.4.3.5), reduced by the following factors:

- A factor of 100 to account for transport to the upper soil layers, as discussed in paragraph 888.
- A factor of 10 to account for losses between the overflow point and the pond, and dilution in pond water. This factor was determined by modelling transport from the overflow point to the existing pond. There was a reduction in activity concentration in the pond compared to the topsoil compartment at the bathtubbing point that varied from a factor of 10 to a factor of $1\ 10^{13}$. and much more than 10 for some radionuclides, depending on their mobility. A factor of 10 was conservatively used in the assessment.

1227. The activity concentration in pond sediment was calculated using kd values from ERICA.

1228. The activity concentration in the pond slowly increases if bathtubbing is assumed to continue. The maximum activity concentration was used for the assessment. The time of the maximum activity concentration varies depending on the radionuclide, and the earliest time of maximum was 2000 years after closure.

1229. The ERICA assessment tool was then used to calculate a risk quotient for each radionuclide, initially using a Tier 1 assessment, and then using a Tier 2 assessment.

1230. The Tier 2 assessment determined the risk quotients for each radionuclide for all organisms based on a screening level of 10 µGy h^{-1}. The most restrictive organism was selected. Table 185 shows the calculated risk quotients.

Table 184 Radionuclide specific risk quotients for freshwater ecosystems, based on a generic screening level of 10 µGy h^{-1} and the reduction required to reduce the dose rate to 40 µGy h^{-1}

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted Radiological capacity (MBq)
H-3	$6.43\ 10^{9}$	$1.73\ 10^{-6}$		
C-14	$1.87\ 10^{8}$	$1.15\ 10^{-1}$		
Cl-36	$1.56\ 10^{8}$	$9.04\ 10^{-1}$		
Ca-41	$5.77\ 10^{9}$	$6.44\ 10^{-2}$		
Mn-54	$1.12\ 10^{13}$	0		
Fe-55	$1.86\ 10^{13}$	0		
Co-60	$3.58\ 10^{11}$	$3.20\ 10^{-7}$		
Ni-59	$1.95\ 10^{11}$	$4.98\ 10^{-1}$		
Ni-63	$2.42\ 10^{11}$	$1.15\ 10^{-2}$		
Zn-65	$8.95\ 10^{11}$	0		
Se-79	$8.98\ 10^{8}$	$1.03\ 10^{-2}$		
Sr-90	$3.83\ 10^{8}$	$4.77\ 10^{-3}$		
Mo-93	$1.44\ 10^{9}$	$2.03\ 10^{-2}$		
Zr-93	$3.12\ 10^{11}$	$2.38\ 10^{-1}$		
Nb-93m	$5.06\ 10^{10}$	$5.13\ 10^{-7}$		
Nb-94	$6.09\ 10^{6}$	$5.17\ 10^{-5}$		
Tc-99	$6.12\ 10^{8}$	$1.19\ 10^{-1}$		
Ru-106	$9.14\ 10^{11}$	0		
Ag-108m	$2.65\ 10^{8}$	$1.37\ 10^{-1}$		
Ag-110m	$6.41\ 10^{12}$	0		
Cd-109	$1.04\ 10^{12}$	0		
Sb-125	$4.17\ 10^{11}$	0		
Sn-119m	$8.43\ 10^{12}$	0		
Sn-123	$2.97\ 10^{12}$	0		
Sn-126	$4.60\ 10^{6}$	$8.64\ 10^{-4}$		
Te-127m	$4.07\ 10^{12}$	0		
I-129	$3.01\ 10^{8}$	$1.36\ 10^{0}$		
Ba-133	$7.18\ 10^{9}$	$3.44\ 10^{-2}$		
Cs-134	$1.01\ 10^{11}$	0		
Cs-135	$1.55\ 10^{9}$	$6.36\ 10^{-3}$		
Cs-137	$9.69\ 10^{8}$	$2.23\ 10^{-4}$		
Ce-144	$4.81\ 10^{12}$	0		
Pm-147	$2.14\ 10^{13}$	0		
Sm-147	$4.81\ 10^{8}$	$5.35\ 10^{-1}$		
Sm-151	$7.23\ 10^{11}$	$8.56\ 10^{-3}$		
Eu-152	$8.05\ 10^{9}$	$1.31\ 10^{-5}$		
Eu-154	$4.18\ 10^{10}$	$1.61\ 10^{-6}$		
Eu-155	$8.81\ 10^{12}$	$9.61\ 10^{-9}$		

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted Radiological capacity (MBq)
Gd-153	4.83 10^{13}	0		
Pb-210	4.85 10^{8}	8.59 10^{-3}		
Po-210	6.17 10^{9}	0		
Ra-226	3.89 10^{6}	1.27 10^{-1}		
Ra-228	2.25 10^{10}	1.37 10^{-6}		
Ac-227	3.04 10^{9}	6.54 10^{-5}		
Th-228	1.72 10^{11}	0		
Th-229	2.88 10^{7}	3.23 10^{-1}		
Th-230	1.98 10^{6}	4.84 10^{-1}		
Th-232	7.95 10^{6}	1.87 10^{0}		
Pa-231	1.36 10^{7}	4.48 10^{-2}		
U-232	4.04 10^{8}	2.19 10^{0}		
U-233	1.02 10^{8}	1.28 10^{0}		
U-234	1.45 10^{8}	4.29 10^{0}	1.07	1.35 10^{8}
U-235	6.93 10^{7}	1.23 10^{-1}		
U-236	1.48 10^{9}	1.48 10^{0}		
U-238	1.60 10^{9}	3.34 10^{0}		
Np-237	1.42 10^{7}	7.86 10^{-2}		
Pu-238	7.56 10^{8}	3.77 10^{-2}		
Pu-239	1.55 10^{8}	4.38 10^{-1}		
Pu-240	1.89 10^{8}	3.93 10^{-1}		
Pu-241	9.39 10^{9}	1.17 10^{-2}		
Pu-242	1.58 10^{8}	4.89 10^{-1}		
Pu-244	1.26 10^{8}	1.80 10^{0}		
Am-241	3.03 10^{8}	1.10 10^{-2}		
Am-242m	1.75 10^{7}	1.63 10^{-3}		
Am-243	1.46 10^{8}	1.85 10^{-1}		
Cm-242	1.48 10^{11}	3.77 10^{-2}		
Cm-243	4.89 10^{7}	1.67 10^{-4}		
Cm-244	1.16 10^{8}	6.68 10^{-4}		
Cm-245	1.26 10^{7}	5.30 10^{-3}		
Cm-246	1.27 10^{7}	8.04 10^{-4}		
Cm-248	1.45 10^{7}	3.06 10^{-3}		

1231. For U-234, the dose rate is slightly larger than 40 µGy h^{-1}. Since the factor required to reduce the dose rate to 40 µGy h^{-1} is only 1.07 and the dose arises from the bathtubbing scenario, the radiological capacity for U-234 was not adjusted.

E.7.6. ERICA assessment for a Terrestrial Ecosystem

1232. Peak radionuclide concentrations in soil for the terrestrial ecosystem assessment were taken from the GoldSim results for the bathtubbing scenario for the GoldSim ‘topsoil’ compartment (see section E.4.3.5), reduced by a factor of 100 to account for transport to the upper soil layers, as discussed in paragraph 888. For C-14 and H-3, activity concentrations in air (Bq/m3) are also required to calculate doses to non-human biota. Activity concentrations were taken from the gas operations scenario assessment (see subsection E.3.5.3). The soil and air concentrations for each radionuclide were then scaled to account for the radiological capacity of each radionuclide. Activity concentrations for radionuclides that were ingrown through radioactive decay were calculated separately.

1233. The ERICA assessment tool was then used to calculate a risk quotient for each radionuclide, initially using the Tier 1 assessment, and then using a Tier 2 assessment.

1234. The Tier 2 assessment determined the risk quotients for each radionuclide for all organisms based on a screening level of 10 µGy h^{-1}. The most restrictive organism was selected. Table 185 shows the calculated risk quotients.

Table 185 Radionuclide specific risk quotients for terrestrial ecosystems, based on a generic screening level of 10 µGy h^{-1} and the reduction required to reduce the dose rate to 40 µGy h^{-1}

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted Radiological capacity (MBq)
H-3	$6.43\ 10^{9}$	$7.28\ 10^{-4}$		
C-14	$1.87\ 10^{8}$	$1.05\ 10^{-3}$		
Cl-36	$1.56\ 10^{8}$	$4.25\ 10^{-2}$		
Ca-41	$5.77\ 10^{9}$	$3.07\ 10^{-2}$		
Mn-54	$1.12\ 10^{13}$	0		
Fe-55	$1.86\ 10^{13}$	0		
Co-60	$3.58\ 10^{11}$	$1.30\ 10^{-8}$		
Ni-59	$1.95\ 10^{11}$	$7.62\ 10^{-2}$		
Ni-63	$2.42\ 10^{11}$	$4.71\ 10^{-3}$		
Zn-65	$8.95\ 10^{11}$	0		
Se-79	$8.98\ 10^{8}$	$1.26\ 10^{-2}$		
Sr-90	$3.83\ 10^{8}$	$4.82\ 10^{-4}$		
Mo-93	$1.44\ 10^{9}$	$3.66\ 10^{-4}$		
Zr-93	$3.12\ 10^{11}$	$1.03\ 10^{-1}$		
Nb-93m	$5.06\ 10^{10}$	$3.10\ 10^{-7}$		
Nb-94	$6.09\ 10^{6}$	$5.33\ 10^{-4}$		
Tc-99	$6.12\ 10^{8}$	$3.79\ 10^{-2}$		
Ru-106	$9.14\ 10^{11}$	0		
Ag-108m	$2.65\ 10^{8}$	$3.60\ 10^{-3}$		
Ag-110m	$6.41\ 10^{12}$	0		
Cd-109	$1.04\ 10^{12}$	0		
Sb-125	$4.17\ 10^{11}$	0		
Sn-119m	$8.43\ 10^{12}$	0		
Sn-123	$2.97\ 10^{12}$	0		
Sn-126	$4.60\ 10^{6}$	$6.63\ 10^{-4}$		
Te-127m	$4.07\ 10^{12}$	0		
I-129	$3.01\ 10^{8}$	$1.07\ 10^{-3}$		
Ba-133	$7.18\ 10^{9}$	$1.35\ 10^{-4}$		
Cs-134	$1.01\ 10^{11}$	0		
Cs-135	$1.55\ 10^{9}$	$3.13\ 10^{-2}$		

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted Radiological capacity (MBq)
Cs-137	$9.69\ 10^{8}$	$6.27\ 10^{-5}$		
Ce-144	$4.81\ 10^{12}$	0		
Pm-147	$2.14\ 10^{13}$	0		
Sm-147	$4.81\ 10^{8}$	$4.93\ 10^{-1}$		
Sm-151	$7.23\ 10^{11}$	$7.89\ 10^{-3}$		
Eu-152	$8.05\ 10^{9}$	$1.53\ 10^{-5}$		
Eu-154	$4.18\ 10^{10}$	$1.36\ 10^{-6}$		
Eu-155	$8.81\ 10^{12}$	$2.67\ 10^{-9}$		
Gd-153	$4.83\ 10^{13}$	0		
Pb-210	$4.85\ 10^{8}$	$3.76\ 10^{-4}$		
Po-210	$6.17\ 10^{9}$	0		
Ra-226	$3.89\ 10^{6}$	$1.15\ 10^{-2}$		
Ra-228	$2.25\ 10^{10}$	$6.17\ 10^{-8}$		
Ac-227	$3.04\ 10^{9}$	$4.17\ 10^{-6}$		
Th-228	$1.72\ 10^{11}$	0		
Th-229	$2.88\ 10^{7}$	$1.99\ 10^{-2}$		
Th-230	$1.98\ 10^{6}$	$4.64\ 10^{-2}$		
Th-232	$7.95\ 10^{6}$	$8.82\ 10^{-2}$		
Pa-231	$1.36\ 10^{7}$	$4.15\ 10^{-2}$		
U-232	$4.04\ 10^{8}$	$1.36\ 10^{-1}$		
U-233	$1.02\ 10^{8}$	$4.28\ 10^{-1}$		
U-234	$1.45\ 10^{8}$	$8.94\ 10^{-1}$		
U-235	$6.93\ 10^{7}$	$2.80\ 10^{-1}$		
U-236	$1.48\ 10^{9}$	$4.84\ 10^{0}$	1.21	$1.23\ 10^{9}$
U-238	$1.60\ 10^{9}$	$5.15\ 10^{0}$	1.29	$1.25\ 10^{9}$
Np-237	$1.42\ 10^{7}$	$4.92\ 10^{-2}$		
Pu-238	$7.56\ 10^{8}$	$6.82\ 10^{-3}$		
Pu-239	$1.55\ 10^{8}$	$7.13\ 10^{-2}$		
Pu-240	$1.89\ 10^{8}$	$6.41\ 10^{-2}$		
Pu-241	$9.39\ 10^{9}$	$2.79\ 10^{-2}$		
Pu-242	$1.58\ 10^{8}$	$7.97\ 10^{-2}$		
Pu-244	$1.26\ 10^{8}$	$1.10\ 10^{-1}$		
Am-241	$3.03\ 10^{8}$	$2.62\ 10^{-2}$		
Am-242m	$1.75\ 10^{7}$	$8.55\ 10^{-4}$		
Am-243	$1.46\ 10^{8}$	$2.56\ 10^{-1}$		
Cm-242	$1.48\ 10^{11}$	$6.82\ 10^{-3}$		
Cm-243	$4.89\ 10^{7}$	$3.77\ 10^{-5}$		
Cm-244	$1.16\ 10^{8}$	$1.11\ 10^{-4}$		
Cm-245	$1.26\ 10^{7}$	$2.30\ 10^{-2}$		
Cm-246	$1.27\ 10^{7}$	$6.22\ 10^{-3}$		
Cm-248	$1.45\ 10^{7}$	$4.71\ 10^{-2}$		

1235. The dose rates from U-236 and U-238 are slightly larger than 40 µGy h^{-1}, with dose rates of 48 µGy h^{-1} and 52 µGy h^{-1} respectively. The factor required to reduce the dose rate from these radionuclides was calculated, along with the resulting radiological capacity if the reduction is applied, which are also given in Table 185.

1236. However, the activity concentration limit will also limit the total amount of a radionuclide that can be disposed of at Port Clarence. U-238 is limited to an activity concentration of 200 Bq g^{-1}, as discussed in Subsection 7.4.2.2 of the main report. Using this activity concentration, and assuming 20% of the site void contains LLW at this activity concentration, the total amount of U-238 activity that could be placed within Port Clarence is about a factor of 6 less than the radiological capacity that has been calculated based on exposure scenarios. When this lower total activity is used to calculate the dose rate, the dose rate from U-238 reduces to 8.2 µGy h^{-1}. Hence, terrestrial ecosystems will be protected.

1237. U-236 is a minor constituent of LLW and the typical content is a factor of 10 less than that of U-238 (see Table 45). Although the limiting activity concentration for U-236 is higher than that of U-238, it will always be present at lower activity concentrations than U-238 in typical LLW. Hence, based on a realistic waste composition, terrestrial non-human biota are sufficiently protected.

E.7.7. Tier 2 assessment for Burrowing Animals

1238. A minimum of 1 m soil will be put on top of the facility as a restoration layer. Below this soil, there are membranes and 30 cm of granular material. In addition, the top 1 m of waste is not radioactive, which means that the depth at which the first radioactive material is found is at least 2.3 m. The granular layer deters burrowing animals at least for a few hundred years, until it is naturally broken down and mixed with soil.

1239. A review of soil movement by burrowing animals was published as part of the Nirex Safety Studies (Bishop, 1989) and reported a maximum depth of 2.7 m for a warren, with a typical depth of 1.8 m. More recent information suggests a maximum depth of a rabbit burrow or warren is 3 m (Rabbitmatters, n.d.). Hence, although a typical warren will not intercept waste, one at the maximum depth could. Rabbits are not a protected species.

1240. Badgers are a common, protected species. They like to dig their setts where the ground is easy to dig, e.g. in sandy soil, and in sites where the sett stays dry. Badgers do not like digging into clay, as this is wet and sticky. Badger tunnels can be four metres deep, though most are less than one metre underground and follow surface contours (http://www.badgerland.co.uk/animals/sett.html). Other burrowing animals (mice, voles, moles) have a maximum burrow depth that is less than 1 m and therefore will not burrow into the waste.

1241. A Tier 2 ERICA assessment was therefore undertaken to estimate the potential dose to animals burrowing into the waste cells after closure. While this assessment was focused on burrowing animals (rabbits), all the parameters (concentration factors, dose factors, etc.) were set to default ERICA values.

1242. Radionuclide concentrations in the waste cells 60 years after closure were calculated in the GoldSim groundwater model. The concentrations for each radionuclide were then scaled to account for the radiological capacity, see Section 7.4.2.2. Activity concentrations of radionuclides that were ingrown through radioactive decay were calculated separately.

1243. A Tier 2 assessment was carried out within ERICA to calculate a risk quotient for each radionuclide for burrowing mammals. If the risk quotient is higher than one, the dose rate to the most limiting organism exceeds the screening dose rate of 10 µGy h^{-1}. Table 186 shows the calculated risk quotients.

Table 186 Radionuclide specific risk quotients for terrestrial ecosystems for burrowing animals in the waste cells, based on a generic screening level of 10 µGy h^{-1}

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted radiological capacity (MBq)
H-3	$6.43\ 10^{9}$	$7.28\ 10^{-2}$		
C-14	$1.87\ 10^{8}$	$1.05\ 10^{-1}$		
Cl-36	$1.56\ 10^{8}$	$3.31\ 10^{0}$		
Ca-41	$4.83\ 10^{10}$	$3.35\ 10^{1}$	8.37	$5.77\ 10^{9}$
Mn-54	$1.12\ 10^{13}$	$2.39\ 10^{-6}$		
Fe-55	$1.86\ 10^{13}$	$2.28\ 10^{-4}$		
Co-60	$3.58\ 10^{11}$	$3.56\ 10^{0}$		
Ni-59	$3.20\ 10^{11}$	$6.56\ 10^{0}$	1.64	$1.95\ 10^{11}$
Ni-63	$3.10\ 10^{11}$	$5.13\ 10^{0}$	1.28	$2.42\ 10^{11}$
Zn-65	$8.95\ 10^{11}$	$1.02\ 10^{-7}$		
Se-79	$8.98\ 10^{8}$	$1.04\ 10^{-1}$		
Sr-90	$3.83\ 10^{8}$	$1.78\ 10^{0}$		
Mo-93	$1.44\ 10^{9}$	$3.44\ 10^{-2}$		
Zr-93	$3.12\ 10^{11}$	$1.99\ 10^{-2}$		
Nb-93m	$5.06\ 10^{10}$	$7.14\ 10^{-2}$		
Nb-94	$6.09\ 10^{6}$	$9.20\ 10^{-2}$		
Tc-99	$6.12\ 10^{8}$	$2.64\ 10^{-1}$		
Ru-106	$9.14\ 10^{11}$	$2.37\ 10^{-8}$		
Ag-108m	$3.78\ 10^{8}$	$5.70\ 10^{0}$	1.43	$2.65\ 10^{8}$
Ag-110m	$6.41\ 10^{12}$	$1.90\ 10^{-6}$		
Cd-109	$1.04\ 10^{12}$	$1.53\ 10^{-8}$		
Sb-125	$4.17\ 10^{11}$	$7.45\ 10^{-4}$		
Sn-119m	$8.43\ 10^{12}$	$6.85\ 10^{-9}$		
Sn-123	$2.97\ 10^{12}$	$6.66\ 10^{-9}$		
Sn-126	$4.60\ 10^{6}$	$7.02\ 10^{-2}$		
Te-127m	$4.07\ 10^{12}$	$2.09\ 10^{-8}$		
I-129	$3.01\ 10^{8}$	$1.36\ 10^{-1}$		
Ba-133	$7.18\ 10^{9}$	$4.53\ 10^{-1}$		
Cs-134	$1.01\ 10^{11}$	$1.04\ 10^{-5}$		
Cs-135	$5.88\ 10^{10}$	$1.52\ 10^{2}$	37.91	$1.55\ 10^{9}$
Cs-137	$8.61\ 10^{9}$	$3.56\ 10^{1}$	8.89	$9.69\ 10^{8}$
Ce-144	$4.81\ 10^{12}$	$1.56\ 10^{-8}$		
Pm-147	$2.14\ 10^{13}$	$1.08\ 10^{-4}$		
Sm-147	$1.80\ 10^{9}$	$1.50\ 10^{1}$	3.74	$4.81\ 10^{8}$
Sm-151	$2.61\ 10^{13}$	$1.45\ 10^{2}$	36.13	$7.23\ 10^{11}$
Eu-152	$2.28\ 10^{10}$	$1.13\ 10^{1}$	2.83	$8.05\ 10^{9}$
Eu-154	$1.22\ 10^{11}$	$1.17\ 10^{1}$	2.92	$4.18\ 10^{10}$
Eu-155	$8.81\ 10^{12}$	$5.17\ 10^{-1}$		

Radionuclide	Radiological capacity (MBq)	Risk quotient (10 µGy h^{-1} screening)	Reduction factor to reduce dose rate to below 40 µGy h^{-1}	Adjusted radiological capacity (MBq)
Gd-153	$4.83\ 10^{13}$	$2.84\ 10^{-7}$		
Pb-210	$1.33\ 10^{9}$	$1.10\ 10^{1}$	2.74	$4.85\ 10^{8}$
Po-210	$6.17\ 10^{9}$	$2.43\ 10^{-9}$		
Ra-226	$3.89\ 10^{6}$	$6.60\ 10^{-1}$		
Ra-228	$2.25\ 10^{10}$	$5.60\ 10^{-1}$		
Ac-227	$3.04\ 10^{9}$	$8.79\ 10^{-4}$		
Th-228	$1.72\ 10^{11}$	$2.28\ 10^{-6}$		
Th-229	$2.88\ 10^{7}$	$1.62\ 10^{-2}$		
Th-230	$1.98\ 10^{6}$	$8.16\ 10^{-3}$		
Th-232	$7.95\ 10^{6}$	$1.89\ 10^{-1}$		
Pa-231	$3.84\ 10^{7}$	$1.13\ 10^{1}$	2.83	$1.36\ 10^{7}$
U-232	$6.82\ 10^{9}$	$6.75\ 10^{1}$	16.89	$4.04\ 10^{8}$
U-233	$1.02\ 10^{8}$	$2.96\ 10^{-1}$		
U-234	$1.45\ 10^{8}$	$4.23\ 10^{-1}$		
U-235	$6.93\ 10^{7}$	$2.94\ 10^{-1}$		
U-236	$1.97\ 10^{9}$	$5.31\ 10^{0}$	1.33	$1.48\ 10^{9}$
U-238	$2.02\ 10^{9}$	$5.04\ 10^{0}$	1.26	$1.60\ 10^{9}$
Np-237	$2.31\ 10^{8}$	$6.52\ 10^{1}$	16.29	$1.42\ 10^{7}$
Pu-238	$4.67\ 10^{9}$	$2.47\ 10^{1}$	6.18	$7.56\ 10^{8}$
Pu-239	$1.55\ 10^{8}$	$1.24\ 10^{0}$		
Pu-240	$1.89\ 10^{8}$	$1.50\ 10^{0}$		
Pu-241	$1.06\ 10^{11}$	$4.53\ 10^{1}$	11.32	$9.39\ 10^{9}$
Pu-242	$1.58\ 10^{8}$	$1.19\ 10^{0}$		
Pu-244	$1.26\ 10^{8}$	$1.30\ 10^{0}$		
Am-241	$3.60\ 10^{9}$	$4.75\ 10^{1}$	11.89	$3.03\ 10^{8}$
Am-242m	$2.62\ 10^{9}$	$6.00\ 10^{2}$	150	$1.75\ 10^{7}$
Am-243	$1.46\ 10^{8}$	$2.22\ 10^{0}$		
Cm-242	$9.14\ 10^{11}$	$2.47\ 10^{1}$	6.18	$1.48\ 10^{11}$
Cm-243	$1.41\ 10^{10}$	$1.15\ 10^{3}$	287	$4.89\ 10^{7}$
Cm-244	$5.31\ 10^{10}$	$1.83\ 10^{3}$	457	$1.16\ 10^{8}$
Cm-245	$1.02\ 10^{8}$	$3.26\ 10^{1}$	8.14	$1.26\ 10^{7}$
Cm-246	$2.55\ 10^{8}$	$8.03\ 10^{1}$	20.08	$1.27\ 10^{7}$
Cm-248	$4.84\ 10^{7}$	$1.33\ 10^{1}$	3.33	$1.45\ 10^{7}$

1244. There are 26 radionuclides for which the dose rate to the burrowing mammal is greater than 40 µGy h^{-1}, by up to two orders of magnitude. Given the design of the landfill facility and the design of the cap, it seems very unlikely that burrowing animals will build their nesting chambers in the disposed waste. In addition, the purpose of the landfill site is to concentrate and contain the waste to protect the environment, so the environment in the actual landfill (the waste cell) is not the part of the environment that is being protected (it is not a conservation area).

1245. Rabbits are not a protected species. Their high fecundity also means that the population will recover quickly if 10% are affected and a more reasonable value to use for protecting the population may be the HDR50 (or even the EDR50). However, badgers are a protected species.

1246. We note that in their review of the ENRMF ESC, the EA commented that it would be precautionary to apply radiological capacity reduction factors based on the ERICA Tier 2 assessment for burrowing animals.

1247. As such, the radiological capacities were reduced to ensure that dose rates would be below 40 µGy h^{-1}, to the values shown in the right-hand column of Table 186. This ensures that burrowing mammals will be sufficiently protected. The radiological capacity without this reduction is shown in the uncertainty section (see Section E.8).

1248. The alternative is to ensure that wastes containing LLW is buried at a depth below the surface that is greater than 4 m. This will ensure that rabbit warrens or badger tunnels will not enter the waste.

E.8. Management of uncertainty

1249. Uncertainties in dose assessments arise from natural variability, limitations in the knowledge of processes or data, alternative interpretations, and the potential for change in the future, and are generally assigned to one of three categories:

- conceptual model uncertainty - uncertainty in the appropriateness of models used to represent the system;
- scenario uncertainty - uncertainty in the completeness of the set of exposure scenarios; and,
- parameter uncertainty - uncertainty in the parameter values selected for use in the assessment.

1250. Conceptual model uncertainties are not examined in detail within this ESC and are addressed by adopting a generally conservative approach to defining pathways and uptake routes.

1251. Scenario uncertainties relate to the choice of scenarios. A wide range of scenarios has been considered in this ESC based on an analysis of FEPs and other ESCs. Hence it is considered that the scenarios encompass the range of future exposure scenarios.

1252. Parameter value uncertainties have been considered in terms of the sensitivity of the limiting dose assessments to parameter selection. The analysis is described in the next section.

E.8.1. Parameter sensitivity

1253. The equations used in the assessment models are, with the exception of radon migration through a cap and the groundwater pathway, linear. For the linear cases the effect of parameter changes is simply multiplicative.

1254. The radon calculations include an exponential decay of radon through the cap, controlled by the ratio of the cap thickness to the radon relaxation length in the cap.

Changing either of these two parameters will result in large changes in radon flux (note, though, that the dose is linear with respect to the flux, so once the change in flux has been determined, the change in dose is again linear).

1255. Non-linearities also arise in the determination of the source term where there is an exponential term to model radioactive decay or leaching of radionuclides out of the waste cells. The key uncertainty here is likely to be the hydraulic conductivity of the barrier, as this will dictate the long-term radionuclide concentration in the waste cells.

E.8.1.1. Intrusion: Smallholder

1256. The Smallholder scenario (see Section E.5.8) limits the radiological capacity for 6 radionuclides. The impact of changes to parameter values has considered the dilution factor and the timing of the event. The impact is to produce a linear change in the radiological capacity.

1257. The reference dilution factor used in the assessment is 0.0108, based on a dilution factor of waste with cap material during excavation of 0.54, a dilution of 0.2 to account for the average LLW content of the landfill, and a dilution of spoil with clean soil of 0.1. The sensitivity of dose to dilution factors of 0.001, 0.0108, and 0.108 is shown in Table 187. The radionuclides listed are those where the Smallholder scenario constrains the radiological capacity. The impact of a numerically smaller dilution factor is to reduce the dose since it is applied as a multiplying factor.

Table 187 Sensitivity of dose to dilution factor: Smallholder

Radionuclide	Dose per unit disposal; *DIL*=0.001 (µSv MBq^{-1})	Dose per unit disposal; *DIL*=0.0108 (µSv MBq^{-1})	Dose per unit disposal; *DIL*=0.108 (µSv MBq^{-1})
Cl-36	$1.78\ 10^{-6}$	$1.92\ 10^{-5}$	$1.92\ 10^{-4}$
Se-79	$3.09\ 10^{-7}$	$3.34\ 10^{-6}$	$3.34\ 10^{-5}$
Sr-90	$7.25\ 10^{-7}$	$7.83\ 10^{-6}$	$7.83\ 10^{-5}$
Mo-93	$1.93\ 10^{-7}$	$2.09\ 10^{-6}$	$2.09\ 10^{-5}$
Tc-99	$4.54\ 10^{-7}$	$4.90\ 10^{-6}$	$4.90\ 10^{-5}$
I-129	$9.23\ 10^{-7}$	$9.97\ 10^{-6}$	$9.97\ 10^{-5}$

1258. The change in dilution factor results in a linear change in dose, equivalent to the factors applied. The impact on radiological capacity is illustrated in Table 188. The radionuclides listed in Table 188 would become constrained by the Smallholder scenario if a higher value of the dilution factor is applied.

Table 188 Sensitivity of radiological capacity to dilution factor: Smallholder at 60 y

Radionuclide	Scenario radiological capacity;	Scenario radiological capacity;	Scenario radiological capacity;

	DIL=0.001 (MBq)	*DIL*=0.0108 (MBq)	*DIL*=0.108 (MBq)
Ca-41	5.21 10^{11}	4.83 10^{10}	4.83 10^{9}
Co-60	3.37 10^{13}	3.12 10^{12}	3.12 10^{11}
Ni-59	5.47 10^{12}	5.06 10^{11}	5.06 10^{10}
Ni-63	3.35 10^{12}	3.10 10^{11}	3.10 10^{10}
Zr-93	1.60 10^{13}	1.48 10^{12}	1.48 10^{11}
Ag-108m	2.38 10^{10}	2.20 10^{9}	2.20 10^{8}
Eu-152	6.39 10^{11}	5.92 10^{10}	5.92 10^{9}
Eu-154	3.42 10^{12}	3.16 10^{11}	3.16 10^{10}
Pb-210	1.44 10^{10}	1.33 10^{9}	1.33 10^{8}
Ra-228	1.02 10^{12}	9.46 10^{10}	9.46 10^{9}
Ac-227	2.27 10^{11}	2.10 10^{10}	2.10 10^{9}

1259. The impact of changing the timing of the intrusion event is shown for two times of intrusion: 60 y after closure and 200 years after closure. The results are shown in Table 189 for the radionuclides where the Smallholder scenario constrains the radiological capacity. It can be seen that a later intrusion time results in a lower dose for some radionuclides, since the radiological capacity increases, and for other radionuclides the dose remains the same. Using an intrusion time of 200 y, the smallholder scenario remains limiting for all six radionuclides.

Table 189 Sensitivity of projected dose to timing of intrusion: Smallholder

Radionuclide	Radiological capacity when intrusion occurs at 60y	Radiological capacity when intrusion occurs at 200y
Cl-36	1.56 10^{8}	1.56 10^{8}
Se-79	8.98 10^{8}	8.99 10^{8}
Sr-90	3.83 10^{8}	1.11 10^{10}
Mo-93	1.44 10^{9}	1.47 10^{9}
Tc-99	6.12 10^{8}	6.12 10^{8}
I-129	3.01 10^{8}	3.01 10^{8}

E.8.1.2. Groundwater

1260. Several conservative assumptions underlie the Goldsim groundwater model. It is assumed that there is no sorption of radionuclides to waste materials, whereas in reality the waste received at Port Clarence is likely to provide sorption sites within waste cells. Radionuclides are assumed to interact with other soil like materials and with the clay barrier but not with the alluvium within the aquifer. The rate of infiltration to the landfill through the cap is also conservative (see paragraph 781). The depth of clay beneath the landfill used in the assessment (1 m) is less than that beneath the hazardous landfill site (1.5 m).

1261. The application of peak dose output from the model to calculate radiological capacity for use in the sum of fractions is also conservative since the time to peak dose varies from radionuclide to nuclide and the sum of fractions approach assumes that each radionuclide affects the same individual at the same time.

E.8.1.3. Erosion: Dog walker

1262. The erosion: dog walker scenario limits the radiological capacity for 11 radionuclides (see Section E.4.5).

1263. The external dose to the dog walker following coastal erosion uses the external dose coefficient for a slab, which is conservative.

1264. The sensitivity to different exposure times to eroded material was investigated for the walker. If the walker were to spend 1.5 times more time close to the waste, there are three radionuclides for which the erosion: dog walker scenario becomes the most limiting: Ag-108m, U-236 and U-238. The impact on the radiological capacity is summarised in Table 190. Although the radiological capacity would be reduced for these radionuclides, it is not much smaller than the current radiological capacity, limited by a different scenario.

Table 190 Impact on capacity of increased exposure time for the erosion: dog walker scenario

Radionuclide	Maximum capacity using original exposure time (MBq)	Maximum capacity using increased exposure time (MBq)	Current radiological capacity (MBq)
Ag-108m	$3.78\ 10^{8}$	$2.52\ 10^{8}$	$2.65\ 10^{8}$
U-236	$1.97\ 10^{9}$	$1.31\ 10^{9}$	$1.48\ 10^{9}$
U-238	$2.02\ 10^{9}$	$1.35\ 10^{9}$	$1.60\ 10^{9}$

1265. Dose per MBq was calculated for the cases where the walker spends 1.5 times more time by the eroded material and 1.5 less time by the material. The doses for the original 11 limiting radionuclides and the additional three that become limiting when more time is spent close to the waste are shown in Table 191.

Table 191 Sensitivity of projected dose to exposure time: Erosion

Radionuclide	Dose per MBq (default) µSv MBq^{-1}	Dose per MBq (longer exposure time) µSv MBq^{-1}	Dose per MBq (shorter time) µSv MBq^{-1}
Nb-94	$3.28\ 10^{-6}$	$4.93\ 10^{-6}$	$2.19\ 10^{-6}$
Ag-108m	$5.29\ 10^{-8}$	$7.93\ 10^{-8}$	$3.52\ 10^{-8}$
Sn-126	$4.35\ 10^{-6}$	$6.53\ 10^{-6}$	$2.90\ 10^{-6}$
Th-229	$6.95\ 10^{-7}$	$1.04\ 10^{-6}$	$4.63\ 10^{-7}$
Th-232	$2.52\ 10^{-6}$	$3.77\ 10^{-6}$	$1.68\ 10^{-6}$
U-233	$1.96\ 10^{-7}$	$2.93\ 10^{-7}$	$1.30\ 10^{-7}$

U-235	$2.89\ 10^{-7}$	$4.33\ 10^{-7}$	$1.92\ 10^{-7}$
U-236	$1.02\ 10^{-8}$	$1.53\ 10^{-8}$	$6.78\ 10^{-9}$
U-238	$9.90\ 10^{-9}$	$1.49\ 10^{-8}$	$6.60\ 10^{-9}$
Pu-239	$1.29\ 10^{-7}$	$1.93\ 10^{-7}$	$8.58\ 10^{-8}$
Pu-240	$1.06\ 10^{-7}$	$1.59\ 10^{-7}$	$7.05\ 10^{-8}$
Pu-242	$1.26\ 10^{-7}$	$1.89\ 10^{-7}$	$8.42\ 10^{-8}$
Pu-244	$1.59\ 10^{-7}$	$2.38\ 10^{-7}$	$1.06\ 10^{-7}$
Am-243	$1.37\ 10^{-7}$	$2.06\ 10^{-7}$	$9.14\ 10^{-8}$

E.8.1.4. Erosion coast

1266. Erosion is not certain to occur. The sensitivity to including or not including erosion was investigated for the erosion to coast (Section E.4.6) and erosion: dog walker scenarios (Section E.4.5), which limit capacity for 15 radionuclides. The radiological capacity as a result of the coastal erosion scenario, as the capacity if they are excluded is presented in Table 192.

Table 192 Sensitivity of projected dose to erosion rate: Erosion

Radionuclide	Radiological capacity from coastal erosion (MBq)	Radiological capacity if coastal erosion is excluded (MBq)	New limiting scenario
Zr-93	$3.12\ 10^{11}$	$1.48\ 10^{12}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Nb-94	$6.09\ 10^{6}$	$7.66\ 10^{8}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Sn-126	$4.60\ 10^{6}$	$6.25\ 10^{8}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Ra-226	$3.89\ 10^{6}$	$5.01\ 10^{8}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Th-229	$2.88\ 10^{7}$	$9.86\ 10^{8}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Th-230	$1.98\ 10^{6}$	$1.53\ 10^{9}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Th-232	$7.95\ 10^{6}$	$7.23\ 10^{8}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
U-233	$1.02\ 10^{8}$	$2.66\ 10^{10}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
U-234	$1.45\ 10^{8}$	$3.21\ 10^{10}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
U-235	$6.93\ 10^{7}$	$7.93\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pu-239	$1.55\ 10^{8}$	$2.67\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pu-240	$1.89\ 10^{8}$	$2.68\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pu-242	$1.58\ 10^{8}$	$2.90\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth

Radionuclide	Radiological capacity from coastal erosion (MBq)	Radiological capacity if coastal erosion is excluded (MBq)	New limiting scenario
Pu-244	$1.26\ 10^{8}$	$2.88\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Am-243	$1.46\ 10^{8}$	$3.14\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth

1267. For all 15 radionuclides, the radiological capacity increases by at least one order of magnitude if coastal erosion does not occur.

E.8.1.5. Leachate spillage

1268. The leachate spillage scenario limits the capacity for 21 radionuclides (see Section E.3.10).

1269. If irrigation using water containing spilled leachate did not occur, the radiological capacities would change as shown in Table 193.

Table 193 Radiological capacity limits if irrigation using water incorporating spilled leachate does not occur

Radionuclide	Radiological capacity from leachate spillage (MBq)	Radiological capacity if no leachate spillage (MBq)	New limiting scenario
Mn-54	$1.12\ 10^{13}$	$3.74\ 10^{19}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Fe-55	$1.86\ 10^{13}$	$1.16\ 10^{19}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Co-60	$3.58\ 10^{11}$	$1.22\ 10^{12}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Zn-65	$8.95\ 10^{11}$	$5.24\ 10^{18}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Ru-106	$9.14\ 10^{11}$	$2.04\ 10^{21}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Ag-110m	$6.41\ 10^{12}$	$5.23\ 10^{18}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Cd-109	$1.04\ 10^{12}$	$4.51\ 10^{23}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Sb-125	$4.17\ 10^{11}$	$1.06\ 10^{16}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Sn-119m	$8.43\ 10^{12}$	$2.56\ 10^{33}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Sn-123	$2.97\ 10^{12}$	$1.89\ 10^{21}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Te-127m	$4.07\ 10^{12}$	$2.96\ 10^{39}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Ba-133	$7.18\ 10^{9}$	$1.95\ 10^{11}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth

Radionuclide	Radiological capacity from leachate spillage (MBq)	Radiological capacity if no leachate spillage (MBq)	New limiting scenario
Cs-134	$1.01\ 10^{11}$	$4.36\ 10^{17}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Ce-144	$4.81\ 10^{12}$	$1.40\ 10^{34}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pm-147	$2.14\ 10^{13}$	$2.89\ 10^{19}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Eu-155	$8.81\ 10^{12}$	$2.52\ 10^{14}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Gd-153	$4.83\ 10^{13}$	$8.36\ 10^{37}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Po-210	$6.17\ 10^{9}$	$7.08\ 10^{24}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth
Ra-228	$2.25\ 10^{10}$	$9.46\ 10^{10}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Ac-227	$3.04\ 10^{9}$	$3.38\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Th-228	$1.72\ 10^{11}$	$1.03\ 10^{16}$	Gas + Ext. (Recreational 0y) All ages Ra-226 at 5m depth

1270. For most radionuclides, the radiological capacity would increase by at least five orders of magnitude if exposure as a result of irrigation did not occur. The exceptions are Ba-133, Eu-155, Ac-227, for which the increase is smaller.

1271. It was assumed that the volume of the water body into which leachate is spilled was 2,000,000 m^3. The sensitivity of dose to this volume was investigated by varying it by a factor of 1.5, and the results are shown in Table 194 for the 21 limiting radionuclides. It was assumed that dose is linear with respect to leachate dilution. No other radionuclides become limiting if the volume of the diluting water is decreased.

Table 194 Sensitivity of projected dose to dilution: Leachate Spillage

Radionuclide	Dose per MBq (default dilution) µSv MBq^{-1}	Dose per MBq (increased dilution) µSv MBq^{-1}	Dose per MBq (decreased dilution) µSv MBq^{-1}
Mn-54	$2.68\ 10^{-11}$	$1.78\ 10^{-11}$	$4.02\ 10^{-11}$
Fe-55	$1.62\ 10^{-11}$	$1.08\ 10^{-11}$	$2.43\ 10^{-11}$
Co-60	$8.38\ 10^{-10}$	$5.59\ 10^{-10}$	$1.26\ 10^{-9}$
Zn-65	$3.35\ 10^{-10}$	$2.24\ 10^{-10}$	$5.03\ 10^{-10}$
Ru-106	$3.28\ 10^{-10}$	$2.19\ 10^{-10}$	$4.92\ 10^{-10}$
Ag-110m	$4.68\ 10^{-11}$	$3.12\ 10^{-11}$	$7.02\ 10^{-11}$
Cd-109	$2.88\ 10^{-10}$	$1.92\ 10^{-10}$	$4.33\ 10^{-10}$
Sb-125	$7.19\ 10^{-10}$	$4.79\ 10^{-10}$	$1.08\ 10^{-9}$
Sn-119m	$3.56\ 10^{-11}$	$2.37\ 10^{-11}$	$5.34\ 10^{-11}$
Sn-123	$1.01\ 10^{-10}$	$6.74\ 10^{-11}$	$1.52\ 10^{-10}$

Te-127m	$7.38\ 10^{-11}$	$4.92\ 10^{-11}$	$1.11\ 10^{-10}$
Ba-133	$4.18\ 10^{-8}$	$2.79\ 10^{-8}$	$6.27\ 10^{-8}$
Cs-134	$2.97\ 10^{-9}$	$1.98\ 10^{-9}$	$4.45\ 10^{-9}$
Ce-144	$6.23\ 10^{-11}$	$4.16\ 10^{-11}$	$9.35\ 10^{-11}$
Pm-147	$1.40\ 10^{-11}$	$9.34\ 10^{-12}$	$2.10\ 10^{-11}$
Eu-155	$3.40\ 10^{-11}$	$2.27\ 10^{-11}$	$5.11\ 10^{-11}$
Gd-153	$6.22\ 10^{-12}$	$4.15\ 10^{-12}$	$9.33\ 10^{-12}$
Po-210	$4.86\ 10^{-8}$	$3.24\ 10^{-8}$	$7.29\ 10^{-8}$
Ra-228	$1.33\ 10^{-8}$	$8.89\ 10^{-9}$	$2.00\ 10^{-8}$
Ac-227	$9.87\ 10^{-8}$	$6.58\ 10^{-8}$	$1.48\ 10^{-7}$
Th-228	$1.75\ 10^{-9}$	$1.16\ 10^{-9}$	$2.62\ 10^{-9}$

E.8.1.6. Recreational use (Gas and external)

1272. The gas and external doses to a recreational user at 0 y (see Section E.4.2) limits the capacity for three radionuclides: H-3, C-14 and Nb-93m. The sensitivity of the projected dose as a result of gas generation rate (for H-3 and C-14) and for the exposure time have been examined. The impacts on the projected dose per unit inventory are given in Table 195.

Table 195 Sensitivity of projected dose: Recreational use

Radionuclide	Dose per MBq (slow gas)	Dose per MBq (fast gas)	Dose per MBq (default)	Dose per MBq (longer time)	Dose per MBq (shorter time)
H-3	$2.07\ 10^{-15}$	$4.66\ 10^{-15}$	$3.11\ 10^{-15}$	$4.66\ 10^{-15}$	$2.07\ 10^{-15}$
C-14	$7.11\ 10^{-14}$	$1.60\ 10^{-13}$	$1.07\ 10^{-13}$	$1.60\ 10^{-13}$	$7.11\ 10^{-14}$
Nb-93m	$3.95\ 10^{-16}$	$3.95\ 10^{-16}$	$3.95\ 10^{-16}$	$5.93\ 10^{-16}$	$2.64\ 10^{-16}$

E.8.1.7. Small burrowing mammals

1273. A Tier 2 assessment was performed for burrowing animals that enter the waste. Given the design of the landfill facility and the design of the cap, it seems very unlikely that burrowing animals will build their nesting chambers in the disposed waste.

1274. The radiological capacity for 26 radionuclides was reduced to ensure that the dose rates to burrowing mammals from these radionuclides do not exceed 40 μGy h^{-1}.

1275. An alternative approach is to ensure that LLW is buried sufficiently deep that burrowing mammals would not enter the waste. If this is the case, the radiological capacity for these 26 radionuclides will increase to the values presented in Table 196.

Table 196 Radiological capacity for radionuclides if burrowing mammals are not impacted.

Radionuclide	rad capacity from burrowing mammals (MBq)	rad capacity if burrowing mammals is excluded (MBq)	new limiting scenario
Ca-41	$5.77\ 10^{9}$	$4.83\ 10^{10}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Ni-59	$1.95\ 10^{11}$	$3.20\ 10^{11}$	Erosion to coast (2540y) All ages (PC-Cream)
Ni-63	$2.42\ 10^{11}$	$3.10\ 10^{11}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Ag-108m	$2.65\ 10^{8}$	$3.78\ 10^{8}$	Erosion - Dog walker (2540y) All ages
Cs-135	$1.55\ 10^{9}$	$5.88\ 10^{10}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Cs-137	$9.69\ 10^{8}$	$8.61\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Sm-147	$4.81\ 10^{8}$	$1.80\ 10^{9}$	Erosion - Dog walker (2540y) All ages
Sm-151	$7.23\ 10^{11}$	$2.61\ 10^{13}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Eu-152	$8.05\ 10^{9}$	$2.28\ 10^{10}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Eu-154	$4.18\ 10^{10}$	$1.22\ 10^{11}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pb-210	$4.85\ 10^{8}$	$1.33\ 10^{9}$	Intrusion - Smallholder (60y) All ages, Ra-226 at 5m depth
Pa-231	$1.36\ 10^{7}$	$3.84\ 10^{7}$	Erosion - Dog walker (2540y) All ages
U-232	$4.04\ 10^{8}$	$6.82\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
U-236	$1.48\ 10^{9}$	$1.97\ 10^{9}$	Erosion - Dog walker (2540y) All ages
U-238	$1.60\ 10^{9}$	$2.02\ 10^{9}$	Erosion - Dog walker (2540y) All ages
Np-237	$1.42\ 10^{7}$	$2.31\ 10^{8}$	Erosion - Dog walker (2540y) All ages
Pu-238	$7.56\ 10^{8}$	$4.67\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Pu-241	$9.39\ 10^{9}$	$1.06\ 10^{11}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Am-241	$3.03\ 10^{8}$	$3.60\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Am-242m	$1.75\ 10^{7}$	$2.62\ 10^{9}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Cm-242	$1.48\ 10^{11}$	$9.14\ 10^{11}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Cm-243	$4.89\ 10^{7}$	$1.41\ 10^{10}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Cm-244	$1.16\ 10^{8}$	$5.31\ 10^{10}$	Intrusion - Borehole excavator (60y) - worker Ra-226 at 5m depth
Cm-245	$1.26\ 10^{7}$	$1.02\ 10^{8}$	Erosion - Dog walker (2540y) All ages
Cm-246	$1.27\ 10^{7}$	$2.55\ 10^{8}$	Erosion - Dog walker (2540y) All ages
Cm-248	$1.45\ 10^{7}$	$4.84\ 10^{7}$	Erosion - Dog walker (2540y) All ages

Trial pit excavator

1276. The dose to a trial pit excavator who uncovers only LLW i.e. a single consignment of 10 t, corresponding to 10 packages, each with a specific activity of 200 Bq g^{-1} was investigated in the ESC for ENRMF. The highest doses are for Th-232, Sn-126 and Pa-231 which were between 2 and 2.5 mSv y^{-1} (see Section E.5.3.2). Further analysis was undertaken to consider the dose that could occur if a disproportionate amount of activity in a 10 t consignment was in a single package and this was examined for disproportionately longer by the excavator. It is cautiously assumed that there are 10 packages of 1 t each and that 1 package contains 50% of the consignment activity (giving a maximum activity concentration of 1000 Bq g^{-1}) with an exposure to this package lasting 4 hours (the remaining exposure time, 16 hours, and activity is split between the other 9 packages). In these circumstances, the dose to the trial pit excavator increases and the highest doses are between 3 and 4 mSv y^{-1} (see Table 197).

Table 197 Sensitivity to package content in 10 t consignment

Radionuclide	Dose with all packages at 200 Bq g^{-1} (mSv)	Dose with 1 package at 1000 Bq g^{-1} (mSv)
Th-232	2.53	3.66
Pa-231	2.07	2.99
Ra-226	1.50	2.17
Nb-94	1.19	1.72
Ag-108m	1.08	1.56
Th-229	$9.59\ 10^{-1}$	1.39
Pu-239	$3.57\ 10^{-1}$	$5.16\ 10^{-1}$
Pu-240	$3.55\ 10^{-1}$	$5.13\ 10^{-1}$
Th-230	$3.38\ 10^{-1}$	$4.88\ 10^{-1}$
Pu-242	$3.28\ 10^{-1}$	$4.74\ 10^{-1}$

E.9. Tables of universal model parameters

1277. The following Tables list model parameters that are common to all models used in the assessments.

E.9.1. Radionuclide half-lives and decay constants

Table 198 Radionuclide half-lives and decay constants

Radionuclide	Half-life (y)	Decay constant (y^{-1})
H-3	12.3	$5.63\ 10^{-2}$
C-14	$5.70\ 10^{3}$	$1.22\ 10^{-4}$
Cl-36	$3.01\ 10^{5}$	$2.30\ 10^{-6}$
Ca-41	$1.02\ 10^{5}$	$6.80\ 10^{-6}$
Mn-54	0.855	$8.11\ 10^{-1}$
Fe-55	2.74	$2.53\ 10^{-1}$
Co-60	5.27	$1.32\ 10^{-1}$
Ni-59	$1.01\ 10^{5}$	$6.86\ 10^{-6}$
Ni-63	100	$6.92\ 10^{-3}$
Zn-65	0.668	$1.04\ 10^{0}$
Se-79	$2.95\ 10^{5}$	$2.35\ 10^{-6}$
Sr-90	28.8	$2.41\ 10^{-2}$
Mo-93	$4.0\ 10^{3}$	$1.73\ 10^{-4}$
Zr-93	$1.53\ 10^{6}$	$4.53\ 10^{-7}$
Nb-93m	16.1	$4.30\ 10^{-2}$
Nb-94	$2.03\ 10^{4}$	$3.41\ 10^{-5}$
Tc-99	$2.11\ 10^{5}$	$3.28\ 10^{-6}$
Ru-106	1.02	$6.78\ 10^{-1}$
Ag-108m	418	$1.66\ 10^{-3}$
Ag-110m	0.684	$1.01\ 10^{0}$
Cd-109	1.26	$5.49\ 10^{-1}$
Sb-125	2.76	$2.51\ 10^{-1}$
Sn-119m	0.802	$8.64\ 10^{-1}$
Sn-123	0.354	$1.96\ 10^{0}$
Sn-126	$2.30\ 10^{5}$	$3.01\ 10^{-6}$
Te-127m	0.290	$2.39\ 10^{0}$
I-129	$1.57\ 10^{7}$	$4.41\ 10^{-8}$
Ba-133	10.5	$6.59\ 10^{-2}$
Cs-134	2.07	$3.36\ 10^{-1}$
Cs-135	$2.30\ 10^{6}$	$3.01\ 10^{-7}$
Cs-137	30.2	$2.30\ 10^{-2}$
Ce-144	0.780	$8.89\ 10^{-1}$
Pm-147	2.62	$2.64\ 10^{-1}$

Radionuclide	Half-life (y)	Decay constant (y^{-1})
Sm-147	$1.06\ 10^{11}$	$6.54\ 10^{-12}$
Sm-151	90.0	$7.70\ 10^{-3}$
Eu-152	13.5	$5.12\ 10^{-2}$
Eu-154	8.59	$8.07\ 10^{-2}$
Eu-155	4.76	$1.46\ 10^{-1}$
Gd-153	0.658	$1.05\ 10^{0}$
Pb-210	22.2	$3.12\ 10^{-2}$
Po-210	0.379	$1.83\ 10^{0}$
Ra-226	$1.60\ 10^{3}$	$4.33\ 10^{-4}$
Ra-228	5.75	$1.21\ 10^{-1}$
Ac-227	21.8	$3.18\ 10^{-2}$
Th-228	1.91	$3.63\ 10^{-1}$
Th-229	$7.34\ 10^{3}$	$9.44\ 10^{-5}$
Th-230	$7.54\ 10^{4}$	$9.20\ 10^{-6}$
Th-232	$1.41\ 10^{10}$	$4.93\ 10^{-11}$
Pa-231	$3.28\ 10^{4}$	$2.12\ 10^{-5}$
U-232	68.9	$1.01\ 10^{-2}$
U-233	$1.59\ 10^{5}$	$4.35\ 10^{-6}$
U-234	$2.46\ 10^{5}$	$2.82\ 10^{-6}$
U-235	$7.04\ 10^{8}$	$9.85\ 10^{-10}$
U-236	$2.34\ 10^{7}$	$2.96\ 10^{-8}$
U-238	$4.47\ 10^{9}$	$1.55\ 10^{-10}$
Np-237	$2.14\ 10^{6}$	$3.23\ 10^{-7}$
Pu-238	87.7	$7.90\ 10^{-3}$
Pu-239	$2.41\ 10^{4}$	$2.87\ 10^{-5}$
Pu-240	$6.56\ 10^{3}$	$1.06\ 10^{-4}$
Pu-241	14.4	$4.83\ 10^{-2}$
Pu-242	$3.75\ 10^{5}$	$1.85\ 10^{-6}$
Pu-244	$8.0\ 10^{7}$	$8.66\ 10^{-9}$
Am-241	432	$1.60\ 10^{-3}$
Am-242m	141	$4.92\ 10^{-3}$
Am-243	$7.37\ 10^{3}$	$9.40\ 10^{-5}$
Cm-242	0.446	$1.56\ 10^{0}$
Cm-243	29.1	$2.38\ 10^{-2}$
Cm-244	18.1	$3.83\ 10^{-2}$
Cm-245	$8.50\ 10^{3}$	$8.15\ 10^{-5}$
Cm-246	$4.76\ 10^{3}$	$1.46\ 10^{-4}$
Cm-248	$3.48\ 10^{5}$	$1.99\ 10^{-6}$

E.9.2. Sorption Distribution coefficients

Table 199 Sorption distribution coefficients for the filling materials in the waste cells

Radionuclide	K_d soil (m^3 kg^{-1})	K_d clay (m^3 kg^{-1})
H-3	$1.00\ 10^{-4}$	$1.00\ 10^{-4}$
C-14	$1.00\ 10^{-1}$	$1.00\ 10^{-3}$
Cl-36	$1.50\ 10^{-2}$	$3.00\ 10^{-4}$
Ca-41	$9.00\ 10^{-3}$	$5.0\ 10^{-2}$
Mn-54	$4.90\ 10^{-2}$	$1.20\ 10^{0}$
Fe-55	$2.20\ 10^{-1}$	$1.60\ 10^{0}$
Co-60	$6.00\ 10^{-2}$	$4.90\ 10^{0}$
Ni-59	$4.00\ 10^{-1}$	$5.00\ 10^{-2}$
Ni-63	$4.00\ 10^{-1}$	$5.00\ 10^{-2}$
Zn-65	$2.00\ 10^{-1}$	$2.90\ 10^{0}$
Se-79	$1.50\ 10^{-1}$	$3.00\ 10^{-2}$
Sr-90	$1.30\ 10^{-2}$	$1.00\ 10^{-1}$
Mo-93	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$
Zr-93	$6.00\ 10^{-1}$	$2.00\ 10^{-1}$
Nb-93m	$1.60\ 10^{-1}$	$2.00\ 10^{0}$
Nb-94	$1.60\ 10^{-1}$	$2.00\ 10^{0}$
Tc-99	$1.40\ 10^{-4}$	$6.30\ 10^{-5}$
Ru-106	$5.50\ 10^{-2}$	$4.00\ 10^{-1}$
Ag-108m	$9.00\ 10^{-2}$	$1.00\ 10^{-1}$
Ag-110m	$9.00\ 10^{-2}$	$1.00\ 10^{-1}$
Cd-109	$1.50\ 10^{-1}$	$1.10\ 10^{-1}$
Sb-125	$4.50\ 10^{-2}$	$2.00\ 10^{-2}$
Sn-119m	$1.30\ 10^{-1}$	$3.00\ 10^{-1}$
Sn-123	$1.30\ 10^{-1}$	$3.00\ 10^{-1}$
Sn-126	$1.30\ 10^{-1}$	$3.00\ 10^{-1}$
Te-127m	$1.30\ 10^{-1}$	$4.80\ 10^{-1}$
I-129	$1.00\ 10^{-3}$	$7.00\ 10^{-3}$
Ba-133	$9.00\ 10^{0}$	$4.00\ 10^{-4}$
Cs-134	$2.70\ 10^{-1}$	$3.0\ 10^{-1}$
Cs-135	$2.70\ 10^{-1}$	$3.0\ 10^{-1}$
Cs-137	$2.70\ 10^{-1}$	$3.0\ 10^{-1}$
Ce-144	$4.90\ 10^{-1}$	$4.30\ 10^{1}$
Pm-147	$2.40\ 10^{-1}$	$4.50\ 10^{-1}$

Radionuclide	K_d soil (m^3 kg^{-1})	K_d clay (m^3 kg^{-1})
Sm-147	$2.40\ 10^{-1}$	$7.00\ 10^{-1}$
Sm-151	$2.40\ 10^{-1}$	$7.00\ 10^{-1}$
Eu-152	$2.40\ 10^{-1}$	$1.00\ 10^{0}$
Eu-154	$2.40\ 10^{-1}$	$1.00\ 10^{0}$
Eu-155	$2.40\ 10^{-1}$	$1.00\ 10^{0}$
Gd-153	$4.90\ 10^{-1}$	$4.30\ 10^{1}$
Pb-210	$2.70\ 10^{-1}$	$4.90\ 10^{0}$
Po-210	$1.50\ 10^{-1}$	$1.90\ 10^{-1}$
Ra-226	$4.90\ 10^{-1}$	$9.00\ 10^{0}$
Ra-228	$4.90\ 10^{-1}$	$9.00\ 10^{0}$
Ac-227	$4.50\ 10^{-1}$	$2.90\ 10^{0}$
Th-228	$3.00\ 10^{0}$	$1.43\ 10^{1}$
Th-229	$3.00\ 10^{0}$	$1.43\ 10^{1}$
Th-230	$3.00\ 10^{0}$	$1.43\ 10^{1}$
Th-232	$3.00\ 10^{0}$	$1.43\ 10^{1}$
Pa-231	$5.40\ 10^{-1}$	$5.70\ 10^{0}$
U-232	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
U-233	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
U-234	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
U-235	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
U-236	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
U-238	$3.30\ 10^{-2}$	$3.00\ 10^{-1}$
Np-237	$4.10\ 10^{-3}$	$3.00\ 10^{-2}$
Pu-238	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Pu-239	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Pu-240	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Pu-241	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Pu-242	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Pu-244	$5.40\ 10^{-1}$	$1.00\ 10^{0}$
Am-241	$2.00\ 10^{0}$	$2.90\ 10^{0}$
Am-242m	$2.00\ 10^{0}$	$2.90\ 10^{0}$
Am-243	$2.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-242	$4.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-243	$4.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-244	$4.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-245	$4.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-246	$4.00\ 10^{0}$	$2.90\ 10^{0}$
Cm-248	$4.00\ 10^{0}$	$2.90\ 10^{0}$

E.9.3. Dose Coefficients

Table 200 Radionuclide dose coefficients for ingestion and inhalation, and release fraction in a fire

Radionuclide	Ingestion - adult (Sv Bq^{-1})	Ingestion - child (Sv Bq^{-1})	Ingestion - infant (Sv Bq^{-1})	Inhalation - adult (Sv Bq^{-1})	Inhalation - child (Sv Bq^{-1})	Inhalation - infant (Sv Bq^{-1})	Release fraction (fire)
H-3	$1.80\ 10^{-11}$	$2.30\ 10^{-11}$	$4.80\ 10^{-11}$	$2.60\ 10^{-10}$	$3.80\ 10^{-10}$	$1.00\ 10^{-9}$	1.000
C-14	$5.80\ 10^{-10}$	$8.00\ 10^{-10}$	$1.60\ 10^{-9}$	$5.80\ 10^{-9}$	$7.40\ 10^{-9}$	$1.70\ 10^{-8}$	1.000
Cl-36	$9.30\ 10^{-10}$	$1.90\ 10^{-9}$	$6.30\ 10^{-9}$	$7.30\ 10^{-9}$	$1.00\ 10^{-8}$	$2.60\ 10^{-8}$	1.000
Ca-41	$1.90\ 10^{-10}$	$4.80\ 10^{-10}$	$5.20\ 10^{-10}$	$1.80\ 10^{-10}$	$3.30\ 10^{-10}$	$6.00\ 10^{-10}$	0.001
Mn-54	$7.10\ 10^{-10}$	$1.30\ 10^{-9}$	$3.10\ 10^{-9}$	$1.50\ 10^{-9}$	$2.40\ 10^{-9}$	$6.20\ 10^{-9}$	0.001
Fe-55	$3.30\ 10^{-10}$	$1.10\ 10^{-9}$	$2.40\ 10^{-9}$	$7.70\ 10^{-10}$	$1.40\ 10^{-9}$	$3.20\ 10^{-9}$	0.001
Co-60	$3.40\ 10^{-9}$	$1.10\ 10^{-8}$	$2.70\ 10^{-8}$	$3.10\ 10^{-8}$	$4.00\ 10^{-8}$	$8.60\ 10^{-8}$	0.001
Ni-59	$6.30\ 10^{-11}$	$1.10\ 10^{-10}$	$3.40\ 10^{-10}$	$4.40\ 10^{-10}$	$5.90\ 10^{-10}$	$1.50\ 10^{-9}$	0.001
Ni-63	$1.50\ 10^{-10}$	$2.80\ 10^{-10}$	$8.40\ 10^{-10}$	$1.30\ 10^{-9}$	$1.70\ 10^{-9}$	$4.30\ 10^{-9}$	0.001
Zn-65	$3.90\ 10^{-9}$	$6.40\ 10^{-9}$	$1.60\ 10^{-8}$	$2.20\ 10^{-9}$	$3.80\ 10^{-9}$	$1.00\ 10^{-8}$	0.100
Se-79	$2.90\ 10^{-9}$	$1.40\ 10^{-8}$	$2.80\ 10^{-8}$	$6.80\ 10^{-9}$	$8.70\ 10^{-9}$	$2.00\ 10^{-8}$	1.000
Sr-90	$3.07\ 10^{-8}$	$6.59\ 10^{-8}$	$9.30\ 10^{-8}$	$1.62\ 10^{-7}$	$1.83\ 10^{-7}$	$4.09\ 10^{-7}$	0.001
Mo-93	$3.10\ 10^{-9}$	$4.00\ 10^{-9}$	$6.90\ 10^{-9}$	$2.30\ 10^{-9}$	$2.80\ 10^{-9}$	$5.80\ 10^{-9}$	0.001
Zr-93	$1.10\ 10^{-9}$	$5.80\ 10^{-10}$	$7.60\ 10^{-10}$	$2.50\ 10^{-8}$	$9.70\ 10^{-9}$	$6.40\ 10^{-9}$	0.001
Nb-93m	$1.20\ 10^{-10}$	$2.70\ 10^{-10}$	$9.10\ 10^{-10}$	$1.80\ 10^{-9}$	$2.50\ 10^{-9}$	$6.50\ 10^{-9}$	0.001
Nb-94	$1.70\ 10^{-9}$	$3.40\ 10^{-9}$	$9.70\ 10^{-9}$	$4.90\ 10^{-8}$	$5.80\ 10^{-8}$	$1.20\ 10^{-7}$	0.001
Tc-99	$6.40\ 10^{-10}$	$1.30\ 10^{-9}$	$4.80\ 10^{-9}$	$1.30\ 10^{-8}$	$1.70\ 10^{-8}$	$3.70\ 10^{-8}$	0.001
Ru-106	$7.00\ 10^{-9}$	$1.50\ 10^{-8}$	$4.90\ 10^{-8}$	$6.60\ 10^{-8}$	$9.10\ 10^{-8}$	$2.30\ 10^{-7}$	0.100
Ag-108m	$2.30\ 10^{-9}$	$4.30\ 10^{-9}$	$1.10\ 10^{-8}$	$3.70\ 10^{-8}$	$4.40\ 10^{-8}$	$8.70\ 10^{-8}$	0.010
Ag-110m	$2.80\ 10^{-9}$	$5.20\ 10^{-9}$	$1.40\ 10^{-8}$	$1.20\ 10^{-8}$	$1.80\ 10^{-8}$	$4.10\ 10^{-8}$	0.010
Cd-109	$2.00\ 10^{-9}$	$3.50\ 10^{-9}$	$9.50\ 10^{-9}$	$8.10\ 10^{-9}$	$1.40\ 10^{-8}$	$3.70\ 10^{-8}$	0.001
Sb-125	$1.30\ 10^{-9}$	$2.53\ 10^{-9}$	$7.54\ 10^{-9}$	$1.30\ 10^{-8}$	$1.73\ 10^{-8}$	$4.10\ 10^{-8}$	0.100
Sn-119m	$3.40\ 10^{-10}$	$7.50\ 10^{-10}$	$2.50\ 10^{-9}$	$2.20\ 10^{-9}$	$3.10\ 10^{-9}$	$7.90\ 10^{-9}$	0.001

Radionuclide	Ingestion - adult (Sv Bq^{-1})	Ingestion - child (Sv Bq^{-1})	Ingestion - infant (Sv Bq^{-1})	Inhalation - adult (Sv Bq^{-1})	Inhalation - child (Sv Bq^{-1})	Inhalation - infant (Sv Bq^{-1})	Release fraction (fire)
Sn-123	2.10 10^{-9}	4.60 10^{-9}	1.60 10^{-8}	8.10 10^{-9}	1.20 10^{-8}	3.10 10^{-8}	0.001
Sn-126	5.07 10^{-9}	1.06 10^{-8}	3.22 10^{-8}	2.85 10^{-8}	4.18 10^{-8}	1.02 10^{-7}	0.001
Te-127m	2.30 10^{-9}	5.20 10^{-9}	1.80 10^{-8}	9.80 10^{-9}	1.40 10^{-8}	3.30 10^{-8}	0.100
I-129	1.10 10^{-7}	1.90 10^{-7}	2.20 10^{-7}	3.60 10^{-8}	6.70 10^{-8}	8.60 10^{-8}	1.000
Ba-133	1.50 10^{-9}	4.60 10^{-9}	6.20 10^{-9}	1.00 10^{-8}	1.30 10^{-8}	2.90 10^{-8}	0.001
Cs-134	1.90 10^{-8}	1.40 10^{-8}	1.60 10^{-8}	2.00 10^{-8}	2.80 10^{-8}	6.30 10^{-8}	0.100
Cs-135	2.00 10^{-9}	1.70 10^{-9}	2.30 10^{-9}	8.60 10^{-9}	1.10 10^{-8}	2.40 10^{-8}	0.100
Cs-137	1.30 10^{-8}	1.00 10^{-8}	1.20 10^{-8}	3.90 10^{-8}	4.80 10^{-8}	1.00 10^{-7}	0.100
Ce-144	5.20 10^{-9}	1.10 10^{-8}	3.90 10^{-8}	5.30 10^{-8}	7.80 10^{-8}	2.70 10^{-7}	0.001
Pm-147	2.60 10^{-10}	5.70 10^{-10}	1.90 10^{-9}	5.00 10^{-9}	7.00 10^{-9}	1.80 10^{-8}	0.001
Sm-147	4.90 10^{-8}	6.40 10^{-8}	1.40 10^{-7}	9.60 10^{-6}	1.10 10^{-5}	2.30 10^{-5}	0.001
Sm-151	9.80 10^{-11}	2.00 10^{-10}	6.40 10^{-10}	4.00 10^{-9}	4.50 10^{-9}	1.00 10^{-8}	0.001
Eu-152	1.40 10^{-9}	2.60 10^{-9}	7.40 10^{-9}	4.20 10^{-8}	4.90 10^{-8}	1.00 10^{-7}	0.001
Eu-154	2.00 10^{-9}	4.10 10^{-9}	1.20 10^{-8}	5.30 10^{-8}	6.50 10^{-8}	1.50 10^{-7}	0.001
Eu-155	3.20 10^{-10}	6.80 10^{-10}	2.20 10^{-9}	6.90 10^{-9}	9.20 10^{-9}	2.30 10^{-8}	0.001
Gd-153	2.70 10^{-10}	5.80 10^{-10}	1.80 10^{-9}	2.10 10^{-9}	3.90 10^{-9}	1.20 10^{-8}	0.001
Pb-210	1.89 10^{-6}	4.50 10^{-6}	1.24 10^{-5}	9.99 10^{-6}	1.32 10^{-5}	3.23 10^{-5}	0.500
Po-210	1.20 10^{-6}	2.60 10^{-6}	8.80 10^{-6}	4.30 10^{-6}	5.90 10^{-6}	1.40 10^{-5}	0.001
Ra-226	2.17 10^{-6}	5.30 10^{-6}	1.34 10^{-5}	1.95 10^{-5}	2.53 10^{-5}	6.14 10^{-5}	0.001
Ra-228	8.34 10^{-7}	4.32 10^{-6}	6.80 10^{-6}	5.96 10^{-5}	7.98 10^{-5}	2.08 10^{-4}	0.001
Ac-227	1.21 10^{-6}	1.98 10^{-6}	4.29 10^{-6}	5.69 10^{-4}	7.45 10^{-4}	1.65 10^{-3}	0.001
Th-228	1.43 10^{-7}	4.21 10^{-7}	1.09 10^{-6}	4.36 10^{-5}	5.97 10^{-5}	1.60 10^{-4}	0.001
Th-229	6.13 10^{-7}	1.17 10^{-6}	2.38 10^{-6}	2.56 10^{-4}	3.11 10^{-4}	5.55 10^{-4}	0.001
Th-230	2.10 10^{-7}	2.40 10^{-7}	4.10 10^{-7}	1.00 10^{-4}	1.10 10^{-4}	2.00 10^{-4}	0.001
Th-232	1.06 10^{-6}	4.61 10^{-6}	7.25 10^{-6}	1.70 10^{-4}	2.10 10^{-4}	4.28 10^{-4}	0.001
Pa-231	7.10 10^{-7}	9.20 10^{-7}	1.30 10^{-6}	1.40 10^{-4}	1.50 10^{-4}	2.30 10^{-4}	0.001
U-232	4.73 10^{-7}	9.91 10^{-7}	1.91 10^{-6}	8.06 10^{-5}	1.03 10^{-4}	2.57 10^{-4}	0.001
U-233	5.10 10^{-8}	7.80 10^{-8}	1.40 10^{-7}	9.60 10^{-6}	1.20 10^{-5}	3.00 10^{-5}	0.001
U-234	4.90 10^{-8}	7.40 10^{-8}	1.30 10^{-7}	9.40 10^{-6}	1.20 10^{-5}	2.90 10^{-5}	0.001

Radionuclide	Ingestion - adult (Sv Bq^{-1})	Ingestion - child (Sv Bq^{-1})	Ingestion - infant (Sv Bq^{-1})	Inhalation - adult (Sv Bq^{-1})	Inhalation - child (Sv Bq^{-1})	Inhalation - infant (Sv Bq^{-1})	Release fraction (fire)
U-235	$4.73\ 10^{-8}$	$7.17\ 10^{-8}$	$1.33\ 10^{-7}$	$8.50\ 10^{-6}$	$1.10\ 10^{-5}$	$2.60\ 10^{-5}$	0.001
U-236	$4.70\ 10^{-8}$	$7.00\ 10^{-8}$	$1.30\ 10^{-7}$	$8.70\ 10^{-6}$	$1.10\ 10^{-5}$	$2.70\ 10^{-5}$	0.001
U-238	$4.84\ 10^{-8}$	$7.54\ 10^{-8}$	$1.45\ 10^{-7}$	$8.01\ 10^{-6}$	$1.00\ 10^{-5}$	$2.50\ 10^{-5}$	0.001
Np-237	$1.11\ 10^{-7}$	$1.12\ 10^{-7}$	$2.16\ 10^{-7}$	$5.00\ 10^{-5}$	$5.00\ 10^{-5}$	$9.30\ 10^{-5}$	0.001
Pu-238	$2.30\ 10^{-7}$	$2.40\ 10^{-7}$	$4.00\ 10^{-7}$	$1.10\ 10^{-4}$	$1.10\ 10^{-4}$	$1.90\ 10^{-4}$	0.001
Pu-239	$2.50\ 10^{-7}$	$2.70\ 10^{-7}$	$4.20\ 10^{-7}$	$1.20\ 10^{-4}$	$1.20\ 10^{-4}$	$2.00\ 10^{-4}$	0.001
Pu-240	$2.50\ 10^{-7}$	$2.70\ 10^{-7}$	$4.20\ 10^{-7}$	$1.20\ 10^{-4}$	$1.20\ 10^{-4}$	$2.00\ 10^{-4}$	0.001
Pu-241	$4.80\ 10^{-9}$	$5.10\ 10^{-9}$	$5.70\ 10^{-9}$	$2.30\ 10^{-6}$	$2.40\ 10^{-6}$	$2.90\ 10^{-6}$	0.001
Pu-242	$2.40\ 10^{-7}$	$2.60\ 10^{-7}$	$4.00\ 10^{-7}$	$1.10\ 10^{-4}$	$1.20\ 10^{-4}$	$1.90\ 10^{-4}$	0.001
Pu-244	$2.41\ 10^{-7}$	$2.62\ 10^{-7}$	$4.18\ 10^{-7}$	$1.10\ 10^{-4}$	$1.20\ 10^{-4}$	$1.90\ 10^{-4}$	0.001
Am-241	$2.00\ 10^{-7}$	$2.20\ 10^{-7}$	$3.70\ 10^{-7}$	$9.60\ 10^{-5}$	$1.00\ 10^{-4}$	$1.80\ 10^{-4}$	0.001
Am-242m	$2.42\ 10^{-7}$	$2.65\ 10^{-7}$	$4.34\ 10^{-7}$	$1.16\ 10^{-4}$	$1.21\ 10^{-4}$	$2.00\ 10^{-4}$	0.001
Am-243	$2.01\ 10^{-7}$	$2.22\ 10^{-7}$	$3.76\ 10^{-7}$	$9.60\ 10^{-5}$	$1.00\ 10^{-4}$	$1.70\ 10^{-4}$	0.001
Cm-242	$1.20\ 10^{-8}$	$2.40\ 10^{-8}$	$7.60\ 10^{-8}$	$5.90\ 10^{-6}$	$8.20\ 10^{-6}$	$2.10\ 10^{-5}$	0.001
Cm-243	$1.51\ 10^{-7}$	$1.61\ 10^{-7}$	$3.31\ 10^{-7}$	$6.93\ 10^{-5}$	$7.33\ 10^{-5}$	$1.50\ 10^{-4}$	0.001
Cm-244	$1.20\ 10^{-7}$	$1.40\ 10^{-7}$	$2.90\ 10^{-7}$	$5.70\ 10^{-5}$	$6.10\ 10^{-5}$	$1.30\ 10^{-4}$	0.001
Cm-245	$2.10\ 10^{-7}$	$2.30\ 10^{-7}$	$3.70\ 10^{-7}$	$9.90\ 10^{-5}$	$1.00\ 10^{-4}$	$1.80\ 10^{-4}$	0.001
Cm-246	$2.10\ 10^{-7}$	$2.20\ 10^{-7}$	$3.70\ 10^{-7}$	$9.80\ 10^{-5}$	$1.00\ 10^{-4}$	$1.80\ 10^{-4}$	0.001
Cm-248	$7.70\ 10^{-7}$	$8.40\ 10^{-7}$	$1.40\ 10^{-6}$	$3.60\ 10^{-4}$	$3.70\ 10^{-4}$	$6.50\ 10^{-4}$	0.001

Table 201 Radionuclide external dose coefficients and attenuation coefficients for soil

Radionuclide	External Irradiation from slab $ (Sv y^{-1} Bq^{-1} kg)	Gamma skin dose (7 mg cm^{-2}) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (4 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (40 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Attenuation coefficient (m^{-1})*
H-3	0	0	0	0	0
C-14	3.64 10^{-12}	0	9.02 10^{-7}	0	-5.59 10^{1}
Cl-36	6.46 10^{-10}	1.10 10^{-11}	2.51 10^{-6}	5.37 10^{-7}	-2.04 10^{1}
Ca-41	0	0	0	0	0
Mn-54	1.39 10^{-6}	6.10 10^{-8}	0	0	-1.36 10^{1}
Fe-55	0	1.60 10^{-8}	0	0	0
Co-60	4.38 10^{-6}	1.30 10^{-7}	1.83 10^{-6}	2.85 10^{-8}	-1.20 10^{1}
Ni-59	0	6.39 10^{-8}	0	0	0
Ni-63	0	0	1.83 10^{-8}	0	0
Zn-65	1.00 10^{-6}	5.00 10^{-8}	3.77 10^{-8}	1.14 10^{-9}	-1.26 10^{1}
Se-79	5.03 10^{-12}	0	1.14 10^{-6}	0	-5.40 10^{1}
Sr-90	6.65 10^{-9}	2.40 10^{-12}	5.14 10^{-6}	1.76 10^{-6}	-1.85 10^{1}
Mo-93	1.60 10^{-10}	2.13 10^{-8}	0	0	0
Zr-93	0	1.51 10^{-10}	4.80 10^{-7}	0	0
Nb-93m	2.81 10^{-11}	3.69 10^{-9}	0	0	0
Nb-94	2.62 10^{-6}	1.00 10^{-7}	2.17 10^{-6}	1.83 10^{-7}	-1.38 10^{1}
Tc-99	3.39 10^{-11}	3.49 10^{-14}	1.60 10^{-6}	1.37 10^{-8}	-3.85 10^{1}
Ru-106	3.49 10^{-7}	1.20 10^{-8}	2.85 10^{-6}	1.60 10^{-6}	-1.47 10^{1}
Ag-108m	2.61 10^{-6}	1.28 10^{-7}	2.76 10^{-7}	1.15 10^{-7}	-1.49 10^{1}
Ag-110m	4.64 10^{-6}	1.50 10^{-7}	8.24 10^{-7}	8.22 10^{-6}	-1.32 10^{1}
Cd-109	3.97 10^{-9}	1.70 10^{-8}	2.05 10^{-6}	0	-4.13 10^{1}
Sb-125	6.62 10^{-7}	3.51 10^{-8}	1.73 10^{-6}	8.45 10^{-8}	-1.54 10^{1}
Sn-119m	8.13 10^{-10}	7.20 10^{-9}	8.56 10^{-7}	0	-3.47 10^{2}
Sn-123	1.37 10^{-8}	0	0	0	-1.33 10^{1}
Sn-126	3.20 10^{-6}	1.33 10^{-7}	4.54 10^{-6}	1.43 10^{-6}	-1.46 10^{1}

Radionuclide	External Irradiation from slab $ (Sv y^{-1} Bq^{-1} kg)	Gamma skin dose (7 mg cm^{-2}) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (4 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (40 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Attenuation coefficient (m^{-1})*
Te-127m	$1.47\ 10^{-9}$	$6.72\ 10^{-10}$	$1.83\ 10^{-6}$	$1.07\ 10^{-8}$	$-3.05\ 10^{1}$
I-129	$3.50\ 10^{-9}$	$9.70\ 10^{-9}$	$6.51\ 10^{-7}$	0	$-1.31\ 10^{2}$
Ba-133	$5.35\ 10^{-7}$	0	0	0	$-1.79\ 10^{1}$
Cs-134	$2.56\ 10^{-6}$	$8.80\ 10^{-8}$	$1.83\ 10^{-6}$	$3.08\ 10^{-7}$	$-1.42\ 10^{1}$
Cs-135	$1.04\ 10^{-11}$	0	$1.10\ 10^{-6}$	$5.71\ 10^{-11}$	$-4.65\ 10^{1}$
Cs-137	$9.20\ 10^{-7}$	$3.31\ 10^{-8}$	$2.54\ 10^{-6}$	$3.92\ 10^{-7}$	$-1.45\ 10^{1}$
Ce-144	$1.94\ 10^{-8}$	$4.10\ 10^{-9}$	$4.45\ 10^{-6}$	$1.50\ 10^{-6}$	$-2.77\ 10^{1}$
Pm-147	$1.35\ 10^{-11}$	$4.90\ 10^{-13}$	$1.26\ 10^{-6}$	$4.11\ 10^{-10}$	$-3.73\ 10^{1}$
Sm-147	0	0	0	0	0
Sm-151	$2.66\ 10^{-13}$	$6.40\ 10^{-12}$	$2.85\ 10^{-8}$	0	$-4.66\ 10^{2}$
Eu-152	$1.89\ 10^{-6}$	$1.18\ 10^{-7}$	$1.60\ 10^{-6}$	$1.71\ 10^{-7}$	$-1.30\ 10^{1}$
Eu-154	$2.08\ 10^{-6}$	$9.02\ 10^{-8}$	$3.42\ 10^{-6}$	$3.77\ 10^{-7}$	$-1.29\ 10^{1}$
Eu-155	$4.92\ 10^{-8}$	$1.77\ 10^{-8}$	$8.68\ 10^{-7}$	$3.20\ 10^{-10}$	$-3.37\ 10^{1}$
Gd-153	$6.61\ 10^{-8}$	$6.30\ 10^{-9}$	$4.00\ 10^{-7}$	0	$-3.57\ 10^{1}$
Pb-210	$1.65\ 10^{-9}$	$8.30\ 10^{-9}$	$2.63\ 10^{-6}$	$8.45\ 10^{-7}$	$-1.39\ 10^{1}$
Po-210	$1.41\ 10^{-11}$	$4.80\ 10^{-13}$	0	0	$-1.39\ 10^{1}$
Ra-226	$3.03\ 10^{-6}$	$1.64\ 10^{-7}$	$8.53\ 10^{-6}$	$2.49\ 10^{-6}$	$-1.18\ 10^{1}$
Ra-228	$4.37\ 10^{-6}$	$5.78\ 10^{-8}$	$3.08\ 10^{-6}$	$7.19\ 10^{-7}$	$-1.03\ 10^{1}$
Ac-227	$4.44\ 10^{-7}$	$3.81\ 10^{-8}$	$6.59\ 10^{-6}$	$2.00\ 10^{-6}$	$-1.47\ 10^{1}$
Th-228	$2.76\ 10^{-6}$	$1.06\ 10^{-7}$	$6.34\ 10^{-6}$	$1.22\ 10^{-6}$	$-1.03\ 10^{1}$
Th-229	$4.29\ 10^{-7}$	$7.31\ 10^{-8}$	$8.56\ 10^{-6}$	$1.36\ 10^{-6}$	$-1.21\ 10^{1}$
Th-230	$3.27\ 10^{-10}$	$3.83\ 10^{-9}$	$1.04\ 10^{-7}$	0	$-2.93\ 10^{1}$
Th-232	$1.63\ 10^{-6}$	$1.65\ 10^{-7}$	$9.46\ 10^{-8}$	$1.94\ 10^{-6}$	$-1.03\ 10^{1}$
Pa-231	$5.15\ 10^{-8}$	$6.27\ 10^{-8}$	$1.48\ 10^{-7}$	$5.14\ 10^{-9}$	$-1.91\ 10^{1}$
U-232	$2.44\ 10^{-10}$	$9.36\ 10^{-8}$	$6.38\ 10^{-6}$	$1.22\ 10^{-6}$	$-1.03\ 10^{1}$
U-233	$3.78\ 10^{-10}$	$1.70\ 10^{-9}$	$5.25\ 10^{-7}$	0	$-2.29\ 10^{1}$
U-234	$1.09\ 10^{-10}$	$2.70\ 10^{-9}$	$7.42\ 10^{-9}$	0	$-3.58\ 10^{1}$

Radionuclide	External Irradiation from slab $ (Sv y^{-1} Bq^{-1} kg)	Gamma skin dose (7 mg cm^{-2}) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (4 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Beta skin dose (40 mg cm-2) (Sv h^{-1} Bq^{-1} cm^{2})	Attenuation coefficient (m^{-1})*
U-235	1.95 10^{-7}	5.31 10^{-8}	2.52 10^{-6}	1.09 10^{-8}	-2.32 10^{1}
U-236	5.81 10^{-11}	3.55 10^{-9}	4.57 10^{-9}	0	-3.16 10^{1}
U-238	2.79 10^{-11}	9.70 10^{-9}	3.82 10^{-6}	1.26 10^{-6}	-1.36 10^{1}
Np-237	2.11 10^{-8}	5.50 10^{-8}	3.46 10^{-6}	9.93 10^{-8}	-1.93 10^{1}
Pu-238	4.09 10^{-11}	2.70 10^{-9}	1.06 10^{-7}	0	-3.73 10^{1}
Pu-239	7.98 10^{-11}	1.00 10^{-9}	4.34 10^{-10}	0	-2.18 10^{1}
Pu-240	3.96 10^{-11}	2.60 10^{-9}	0	0	-4.44 10^{1}
Pu-241	1.60 10^{-12}	3.30 10^{-12}	0	0	-2.96 10^{1}
Pu-242	3.46 10^{-11}	3.07 10^{-9}	0	0	-5.58 10^{1}
Pu-244	2.04 10^{-11}	1.70 10^{-8}	2.64 10^{-6}	1.06 10^{-6}	-1.41 10^{1}
Am-241	1.18 10^{-8}	1.70 10^{-8}	5.48 10^{-8}	0	-5.37 10^{1}
Am-242m	4.56 10^{-10}	1.95 10^{-8}	1.94 10^{-6}	2.97 10^{-7}	-1.30 10^{1}
Am-243	3.84 10^{-8}	4.60 10^{-8}	4.24 10^{-6}	1.37 10^{-7}	-2.29 10^{1}
Cm-242	4.62 10^{-11}	2.40 10^{-9}	0	0	-3.16 10^{1}
Cm-243	1.58 10^{-7}	2.75 10^{-8}	1.94 10^{-6}	3.42 10^{-8}	-2.29 10^{1}
Cm-244	3.40 10^{-11}	2.20 10^{-9}	0	0	-3.52 10^{2}
Cm-245	9.19 10^{-8}	9.55 10^{-8}	9.82 10^{-7}	0	-2.79 10^{1}
Cm-246	3.14 10^{-11}	2.52 10^{-9}	0	0	-1.29 10^{2}
Cm-248	2.37 10^{-11}	2.32 10^{-9}	0	0	-3.26 10^{2}

*Attenuation coefficient for soil from (SNIFFER, 2006) taken from Hung (2000). The mass attenuation coefficient would be this value divided by the soil density.

$Dose from (US EPA, 1993) for adults applied to all age groups.

Table 202 External dose coefficients for different thicknesses of contamination in a semi-infinite slab

Radionuclide	Semi-infinite slab dose coefficients ($mSv\ h^{-1}$ per $Bq\ kg^{-1}$ at $1600\ kg/m^3$)			
	Top 1 cm	Top 5 cm	Top 15 cm	Uniform
Am-241	$6.62\ 10^{-10}$	$1.26\ 10^{-9}$	$1.35\ 10^{-9}$	$1.35\ 10^{-9}$
Pu-241	$5.53\ 10^{-14}$	$1.41\ 10^{-13}$	$1.81\ 10^{-13}$	$1.82\ 10^{-13}$
Pu-239	$3.23\ 10^{-12}$	$6.62\ 10^{-12}$	$8.76\ 10^{-12}$	$9.10\ 10^{-12}$
Eu-152	$4.05\ 10^{-8}$	$1.17\ 10^{-7}$	$1.85\ 10^{-7}$	$2.16\ 10^{-7}$
Cs-137+Ba-137m	$2.05\ 10^{-8}$	$5.93\ 10^{-8}$	$9.30\ 10^{-8}$	$1.05\ 10^{-7}$
Sr-90+Y-90	$1.86\ 10^{-10}$	$4.96\ 10^{-10}$	$7.13\ 10^{-10}$	$7.59\ 10^{-10}$
Ni-63	0	0	0	0
Co-60	$8.76\ 10^{-8}$	$2.56\ 10^{-7}$	$4.18\ 10^{-7}$	$5.00\ 10^{-7}$
Fe-55	0	0	0	0
C-14	$2.48\ 10^{-13}$	$3.89\ 10^{-13}$	$4.15\ 10^{-13}$	$4.15\ 10^{-13}$
H-3	0	0	0	0

E.9.4. Crop and animal transfer parameters

Table 203 Uptake factors for various crops ($Bq\ kg^{-1}$ fresh crop per $Bq\ kg^{-1}$ soil)

Element	Grain	Green Vegetables	Root Vegetables	Pasture
H	$1.00\ 10^{-2}$	$5.00\ 10^{0}$	$5.00\ 10^{0}$	$5.00\ 10^{0}$
C	$1.60\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$
Cl	$8.80\ 10^{-2}$	$5.00\ 10^{0}$	$5.00\ 10^{0}$	$5.00\ 10^{0}$
Ca	$8.00\ 10^{-2}$	$5.00\ 10^{-1}$	$5.00\ 10^{-1}$	$5.00\ 10^{-1}$
Mn	$5.00\ 10^{-1}$	$5.00\ 10^{-1}$	$5.00\ 10^{-1}$	$3.00\ 10^{-1}$
Fe	$1.00\ 10^{-1}$	$2.00\ 10^{-4}$	$3.00\ 10^{-4}$	$4.00\ 10^{-4}$
Co	$8.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$	$6.00\ 10^{-3}$
Ni	$1.60\ 10^{-1}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$	$2.00\ 10^{-2}$
Zn	$5.00\ 10^{-2}$	$3.30\ 10^{0}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$
Se	$1.00\ 10^{0}$	$1.00\ 10^{0}$	$1.00\ 10^{0}$	$1.00\ 10^{0}$
Sr	$1.20\ 10^{-1}$	$3.00\ 10^{0}$	$9.00\ 10^{-2}$	$3.00\ 10^{0}$
Mo	$8.00\ 10^{-1}$	$5.10\ 10^{-1}$	$3.20\ 10^{-1}$	$5.40\ 10^{0}$
Zr	$5.00\ 10^{-3}$	$5.00\ 10^{-3}$	$5.00\ 10^{-3}$	$5.00\ 10^{-3}$
Nb	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$
Tc	$1.00\ 10^{1}$	$1.00\ 10^{1}$	$1.00\ 10^{1}$	$1.00\ 10^{1}$
Ru	$1.00\ 10^{-1}$	$4.00\ 10^{-3}$	$1.00\ 10^{-2}$	$4.00\ 10^{-2}$
Ag	$8.80\ 10^{-2}$	$2.70\ 10^{-4}$	$1.30\ 10^{-3}$	$1.50\ 10^{-1}$
Cd	$8.80\ 10^{-1}$	$5.50\ 10^{-1}$	$1.50\ 10^{0}$	$2.10\ 10^{0}$
Sb	$1.50\ 10^{-1}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$
Sn	$2.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$2.00\ 10^{-1}$
Te	$3.60\ 10^{-3}$	$2.50\ 10^{-3}$	$6.00\ 10^{-4}$	$8.00\ 10^{0}$
I	$2.80\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$

Element	Grain	Green Vegetables	Root Vegetables	Pasture
Ba	$8.00\ 10^{-2}$	$5.00\ 10^{-3}$	$5.00\ 10^{-3}$	$2.00\ 10^{-2}$
Cs	$2.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$
Ce	$4.80\ 10^{-2}$	$1.00\ 10^{-3}$	$6.00\ 10^{-4}$	$2.00\ 10^{-2}$
Pm	$4.80\ 10^{-2}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$
Sm	$4.80\ 10^{-2}$	$2.00\ 10^{-3}$	$2.00\ 10^{-3}$	$2.00\ 10^{-3}$
Eu	$4.80\ 10^{-2}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$
Gd	$4.80\ 10^{-2}$	$1.00\ 10^{-3}$	$6.00\ 10^{-4}$	$2.00\ 10^{-2}$
Pb	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$
Po	$2.00\ 10^{-4}$	$2.00\ 10^{-4}$	$2.00\ 10^{-4}$	$2.00\ 10^{-4}$
Ra	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$
Ac	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$
Th	$5.00\ 10^{-4}$	$5.00\ 10^{-4}$	$5.00\ 10^{-4}$	$5.00\ 10^{-4}$
Pa	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$	$4.00\ 10^{-2}$
U	$1.00\ 10^{-4}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$
Np	$3.00\ 10^{-4}$	$1.00\ 10^{-2}$	$1.00\ 10^{-3}$	$5.00\ 10^{-3}$
Pu	$3.00\ 10^{-5}$	$1.00\ 10^{-4}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$
Am	$1.00\ 10^{-5}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$	$5.00\ 10^{-3}$
Cm	$1.00\ 10^{-5}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$	$5.00\ 10^{-3}$

Values from (Augean, 2009)

Table 204 Transfer factors for animal produce

	Transfer factor		
Element	Meat (d kg^{-1})	Milk (d kg^{-1})	Fish (m^3 kg^{-1})
H	$2.90\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-3}$
C	$1.20\ 10^{-1}$	$1.00\ 10^{-2}$	$9.00\ 10^{0}$
Cl	$4.30\ 10^{-2}$	$1.70\ 10^{-2}$	$5.00\ 10^{-2}$
Ca	$2.00\ 10^{-3}$	$3.00\ 10^{-3}$	$2.00\ 10^{-1}$
Mn	$5.00\ 10^{-4}$	$3.00\ 10^{-5}$	$4.00\ 10^{-1}$
Fe	$2.00\ 10^{-2}$	$3.00\ 10^{-5}$	$1.00\ 10^{-1}$
Co	$1.00\ 10^{-2}$	$3.00\ 10^{-4}$	$3.00\ 10^{-1}$
Ni	$5.00\ 10^{-3}$	$1.60\ 10^{-2}$	$1.00\ 10^{-1}$
Zn	$1.00\ 10^{-1}$	$1.00\ 10^{-2}$	$1.00\ 10^{0}$
Se	$7.00\ 10^{-3}$	$4.50\ 10^{-4}$	$8.00\ 10^{-1}$
Sr	$8.00\ 10^{-3}$	$3.00\ 10^{-3}$	$6.00\ 10^{-2}$
Mo	$1.00\ 10^{-3}$	$1.70\ 10^{-3}$	$1.00\ 10^{-2}$
Zr	$1.00\ 10^{-6}$	$5.50\ 10^{-7}$	$3.00\ 10^{-1}$
Nb	$3.00\ 10^{-7}$	$4.10\ 10^{-7}$	$3.00\ 10^{-1}$
Tc	$1.00\ 10^{-4}$	$2.30\ 10^{-5}$	$2.00\ 10^{-2}$
Ru	$5.00\ 10^{-2}$	$3.30\ 10^{-6}$	$1.00\ 10^{-2}$
Ag	$3.00\ 10^{-5}$	$5.00\ 10^{-5}$	$5.00\ 10^{-3}$
Cd	$5.80\ 10^{-3}$	$1.90\ 10^{-4}$	$1.00\ 10^{-1}$
Sb	$4.00\ 10^{-5}$	$2.50\ 10^{-5}$	$1.00\ 10^{-1}$
Sn	$1.90\ 10^{-3}$	$1.00\ 10^{-3}$	$1.00\ 10^{0}$
Te	$7.00\ 10^{-3}$	$4.50\ 10^{-4}$	$2.00\ 10^{-1}$
I	$4.00\ 10^{-2}$	$1.00\ 10^{-2}$	$3.00\ 10^{-2}$

	Transfer factor		
Element	Meat (d kg^{-1})	Milk (d kg^{-1})	Fish (m^3 kg^{-1})
Ba	$5.00\ 10^{-4}$	$2.80\ 10^{-4}$	$4.00\ 10^{-3}$
Cs	$5.00\ 10^{-2}$	$7.90\ 10^{-3}$	$2.00\ 10^{0}$
Ce	$2.00\ 10^{-5}$	$3.00\ 10^{-5}$	$3.00\ 10^{-2}$
Pm	$5.00\ 10^{-3}$	$2.00\ 10^{-5}$	$3.00\ 10^{-2}$
Sm	$5.10\ 10^{-4}$	$2.00\ 10^{-5}$	$3.00\ 10^{-2}$
Eu	$4.70\ 10^{-4}$	$5.00\ 10^{-5}$	$3.00\ 10^{-2}$
Gd	$2.00\ 10^{-5}$	$3.00\ 10^{-5}$	$3.00\ 10^{-2}$
Pb	$4.00\ 10^{-4}$	$3.00\ 10^{-4}$	$3.00\ 10^{-1}$
Po	$5.00\ 10^{-3}$	$3.40\ 10^{-4}$	$5.00\ 10^{-2}$
Ra	$9.00\ 10^{-4}$	$1.30\ 10^{-3}$	$5.00\ 10^{-2}$
Ac	$1.60\ 10^{-4}$	$4.00\ 10^{-7}$	$8.00\ 10^{-1}$
Th	$2.70\ 10^{-3}$	$5.00\ 10^{-6}$	$3.00\ 10^{-2}$
Pa	$5.00\ 10^{-5}$	$5.00\ 10^{-6}$	$1.00\ 10^{-2}$
U	$3.00\ 10^{-4}$	$4.00\ 10^{-4}$	$1.00\ 10^{-2}$
Np	$1.00\ 10^{-3}$	$5.00\ 10^{-6}$	$1.00\ 10^{-2}$
Pu	$1.00\ 10^{-5}$	$1.10\ 10^{-6}$	$4.00\ 10^{-3}$
Am	$4.00\ 10^{-4}$	$1.50\ 10^{-6}$	$3.00\ 10^{-2}$
Cm	$4.00\ 10^{-5}$	$1.50\ 10^{-6}$	$8.00\ 10^{-1}$

* Ac fish value from IAEA 472 (IAEA, 2010)
Other values from (Augean, 2009),

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