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APPENDIX ES11.1

ENVIRONMENTAL SAFETY CASE AND ADDENDUM TO THE ENVIRONMENTAL SAFETY CASE

COMMERCIAL

Environmental Safety Case:

Disposal of Low Activity Low Level Radioactive Waste at East Northants Resource Management Facility

Final: ENE-154/001

Main Report

July 2015

Eden
Nuclear and Environment

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Previous issues of this document shall be destroyed or marked **SUPERSEDED**

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List of Terms and Acronyms

ALARA	As low As Reasonably Achievable
AOD	Above Ordnance Datum
bgl	Below ground level
BGS	British Geological Survey
Biosphere	The part of the environment where living organisms exist, or which is capable of supporting life
BPM	Best Practicable Means
Cefas	Centre for Environment, Fisheries and Aquaculture Science
CLEA	Contaminated Land Exposure Assessment Model
CoRWM	Committee on Radioactive Waste Management
DfT	Department for Transport
EA	Environment Agency
EFEPs	External features, Events and Processes
EH&S	Environment, Health and Safety
EHS&QM	Environment, Health, Safety and Quality Manager
EIA	Environmental Impact Assessment
EMP	Environmental Management Plan
EMS	Environmental Management System
ESC	Environmental Safety Case
ESS	Environmental Safety Strategy
Far Field	The geosphere external to the engineered features of the disposal system.
FEPs	Features, Events and Processes
GDF	Geological Disposal Facility
Geosphere	The solid component of the earth (rock and soil etc.)
NS-GRA	Guidance on the requirements for authorisation for near-surface disposal facilities on land for solid radioactive wastes
PHE	Public Health England
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IPPC	Integrated Pollution Prevention and Control
LA-LLW	Lower Activity Low Level Waste
LLW	Low Level Waste
LOAEL	Lowest Observable Adverse Effects Level
MDI	Mean Daily Intake
NDA	Nuclear Decommissioning Authority
NE	Normal Evolution (a description of the reference case for the evolution of the disposal system)
Near Field	The wastes, waste packages and engineered barriers within the disposal system.
NHB	Non-Human Biota
NOAEL	No Observable Adverse Effects Level
PA	Performance Assessment
PEG	Potentially Exposed Group
QMS	Quality Management System
RWMD	Radioactive Waste Management Directorate (part of the NDA)
SEPA	Scottish Environment Protection Agency
SQEP	Suitably Qualified and Experienced Personnel
UKCP	UK Climate Projections
VLLW	Very Low Level Waste
WAC	Waste Acceptance Criteria
WASSC	Waste Safety Standards Committee (part of the IAEA Safety Standards Commission and Committees)

Units and Prefixes

SI units of radiation and radioactivity

Quantity	SI unit and abbreviation
Absorbed dose	Gray (Gy)
Effective Dose	Sievert (Sv)
Radioactivity	Becquerel (Bq)

Multiples and sub-multiples of SI units

Factor	Prefix and abbreviation	Factor	Prefix and abbreviation
10^{18}	exa (E)	10^{-3}	milli (m)
10^{15}	peta (P)	10^{-6}	micro (μ)
10^{12}	tera (T)	10^{-9}	nano (n)
10^{9}	giga (G)	10^{-12}	pico (p)
10^{6}	mega (M)	10^{-15}	femto (f)
10^{3}	kilo (k)	10^{-18}	atto (a)

Environmental Safety Case for the Disposal of Low-level Radioactive Waste at the East Northants Resource Management Facility: Non-technical Summary

This is the 'Non-technical Summary' of the Environmental Safety Case (ESC) for the Disposal of Low-level Radioactive Waste at the East Northants Resource Management Facility (ENRMF). The disposal of radioactive waste in England and Wales is regulated by the Environment Agency under the Environmental Permitting (England and Wales) Regulations 2010. In 2011 the Environment Agency issued Augean (the operator of the ENRMF) with a Permit for the disposal of radioactive wastes at the ENRMF. This ESC supports an application to the Environment Agency for a variation to the current Permit regarding the disposal of radioactive waste at the ENRMF.

The variation application primarily is made to include the additional landfill area in the west of the site for which Development Consent has been granted and to clarify the approaches necessary to take into account the uneven distribution of activity which is likely in the types of wastes received. The proposals the subject of this application do not change the acceptance limit of 200 Becquerel per gramme (Bq/g) nor do they change the dose criteria based on which the risks to humans and the environment are assessed. The application therefore does not result in an increase in the impact or risk to the public or the environment which even under conservative assumptions would be result in an annual dose of less than 1% of natural background levels of radiation present in the UK.

The ENRMF

Augean is the operator of the ENRMF, which comprises a hazardous waste treatment facility at which materials are recycled, recovered and hazardous properties reduced and a landfill at which a limited range of hazardous wastes and low activity radioactive waste is disposed. On 11th July 2013, the Secretary of State (The East Northamptonshire Resource Management Facility Order, 2013) approved the extension of the ENRMF to include an additional void of 1.2 million cubic metres) over an area of approximately 11 hectares and an increase in the annual capacity of the treatment facility to 150,000 tonnes per year. The Order permits disposal of 150,000 tonnes per year of hazardous and low level radioactive waste (LLW) direct to landfill. The Order states that radioactive waste, to a maximum specific activity of 200 Bq/g may be disposed in cells 4B, 5A, 5B and 6 to 11 with the total amount of LLW deposited in the site limited to a maximum of 448,000 tonnes. The application is for a variation of the current permit to extend the LLW disposal area to include the new phases (6 to 11) of the landfill. To take account of the extended disposal area and based on experience of operating for 3 years, the ESC and assessment calculations have been revised.

Low-level waste

Low-level radioactive wastes form the bulk of all the radioactive wastes in the United Kingdom. About 95 percent of the total physical volume of radioactive wastes is LLW; however, LLW only contains a small fraction of the total radioactivity in all the wastes, much less than one percent of the total. LLW contains a wide range of materials, including: paper, tissue, wood, resins, plastic, steels and other metals, graphite, building rubble, and soil. It includes radioactive wastes from the nuclear industry and from other sources including the oil industry, research facilities, remediation of contaminated sites and hospitals. Augean's proposal for a variation to the permit involves the continued disposal of radioactive waste with a specific activity (radionuclide activity concentration in a consignment) of up to 200 Bq/g. This specific activity is the same as that set out in Augean's current permit. It limits disposals to Low Level Waste that has a relatively low radionuclide content.

Protecting the Environment

When the ENRMF landfill is full and site restoration has been completed, the design minimises contact between infiltrating water and the waste; limiting any releases to the environment. However, it is recognised that over long timescales, small quantities of radioactivity may migrate to the environment. The main purpose of the objectives of the ESC is to show that the public and the environment are adequately protected from such releases. The approach follows guidance for assessing disposal sites prepared by the Environment Agency who regulate radioactive waste disposal in England (Environment Agencies, 2009). The amount of LLW that can be safely accepted at the ENRMF has been determined. The ESC demonstrates that for all reasonably foreseeable circumstances, doses or risks remain below the relevant dose and risk guidance levels that have been defined by the Environment Agency based on International criteria, both for humans and for biota. For humans, in the long term and for events that are expected to occur the Environment Agency requires that a radiation dose of no more than 0.02 mSv y^{-1} arises to members of the most exposed group.

Environmental Monitoring

Environmental monitoring during the period over which the site is managed will check that levels of radioactivity in environmental media will not give rise to emissions which could result in exposure which exceeds the design criteria set for the site. Environmental samples are taken on a regular basis and results are reported to the Environment Agency, who currently undertake an independent sampling programme. All these samples provide additional assurance that the site is performing as expected.

Design and Management

The facility has been designed so that it is consistent with best practice for landfills. It:

- has been in operation since 2002;
- is based on well tried and tested technologies;
- is robust and incorporates multiple engineered barriers and safety functions;
- is regularly reviewed for compliance with current standards as subsequent phases for developing disposal cells are planned;
- is subject to active management control; and,
- maximises the use of passive safety features.

This provides confidence that releases from the facility will be as low as reasonably achievable.

We recognise the importance of an effective management culture and safety procedures to ensure that wastes are transported and handled safely reducing the potential for dose impacts on the workforce and the risk of accidents. Augean has a sound Management

System, a positive safety culture, and is committed to high standards of environmental safety and quality.

Summary of the assessment

A new ESC has been produced to support an application for a permit variation that would enable radioactive waste disposal in a new phase of landfill development at the ENRMF.

The ESC contains a detailed radiological assessment of the dose to the public from disposals of low activity radioactive waste to the ENRMF. This radiological assessment looks at the behaviour of radionuclides in the landfill, considers ways that radionuclides can enter the local environment and has looked at the timescale over which this may occur. Particular attention has been given to the potential for movement of radionuclides in groundwater. The radiological assessment also takes into account the future of the site once it has been closed, examining different site uses and also potential situations that could arise in the future when active control of the site has ceased even the possibility of people digging into the waste or living on top of the site.

The results of the calculations are used to determine the quantity (total activity) of each radionuclide that would meet the health protection standards specified by the EA if it was disposed of at the ENRMF. These quantities are used to limit the disposal of low activity radioactive waste at the ENRMF and they will be specified in the revised Permit. The assessment approach is very conservative and inevitably overestimates the doses that may occur from disposal of each radionuclide. This means that using the conservative calculations will set a lower limiting quantity for the LLW that can be disposed of compared with calculations based on more realistic assumptions.

Low activity radioactive wastes can contain different mixtures of radionuclides. It is not possible to know now the exact mixture of radionuclides that will be contained in future radioactive wastes received at ENRMF: this will only be known when the wastes are generated and analysed. In order to maintain the flexibility to respond to future mixtures of radionuclides, an approach is used by which the total quantity that can be received is under continual review within the framework of an agreed limit set by the calculations in the ESC. This approach is referred to as the “sum of fractions” approach and it will be controlled through a clear condition of the permit. This approach is also used at other sites receiving low activity radioactive waste.

Each waste consignment will be evaluated to check that it meets the criteria set through the ‘sum of fractions’ approach. Each waste consignment will also be evaluated to check that it meets the limits on the total number of tonnes of radioactive waste and on the activity concentration that are specified in the 2013 Order. The limit on the total number of tonnes may be more restrictive than the ‘sum of fractions’ limit on the total activity for some radionuclides and some wastes.

The new ESC is consistent with the previous ESC. The following aspects are unchanged:

- The health standards applied by the EA are the same;
- The same set of possible future exposure situations are considered;
- The same radionuclides are considered;
- The vast majority of the models and data used in the radiological assessment are the same;

- The maximum activity concentration for a consignment of waste is unchanged at 200 Bq/g;
- The capacity of the site is given in terms of the total quantity of each radionuclide that would meet the EA standards for protection of health and the environment;
- Waste acceptance criteria are developed to ensure that wastes received at the ENRMF meet the EA standards for protection of health and the environment.

The differences between the new ESC and the previous ESC are:

- A larger landfill volume including the western extension is considered for radioactive waste disposal;
- More detailed modelling has been carried out of the movement of radionuclides to and in the groundwater;
- A longer time period is considered (up to a hundred thousand years);
- More detailed sensitivity analysis has been carried out to investigate uncertainties;
- Explicit consideration has been given to wastes that contain an uneven distribution of activity;
- A 'sum of fractions' approach has been used to determine which radioactive wastes meet the EA standards and the 2013 Order specification and are therefore acceptable for disposal;
- Additional waste acceptance criteria have been proposed specifying that wastes containing above 5 Bq/g of Ra-226 should be buried at least five metres below the restored land surface; and,
- The calculations include the contributions to emissions of radioactive waste that has already been disposed of in the ENRMF.

Disposal records up to June 2015 show that nearly 10,800 tonnes of low activity radioactive waste have been disposed of in the ENRMF. The total activity disposed of at the site is around 91 GBq (thousand million Bequerels) and the average activity concentration of this waste is below 10 Bq/g, well below the 200 Bq/g activity concentration limit specified in the Permit. The maximum dose from situations that are expected to occur is 0.04 µSv compared with the Environment Agency acceptable dose criteria from the site of 20 µSv per annum. The maximum dose from potential future situations where the waste is unintentionally brought to the surface is 0.5 µSv.

The current Permit limits the radiological capacity of the site to a maximum total of 17 TBq. The actual total activity of waste that could be disposed of at ENRMF under the proposed variation depends on the radionuclide mix in the waste it receives. Assuming that it all contains the same radionuclide mix as the waste that is already in the ENRMF, the maximum total activity that could be disposed of at the ENRMF would be 4 TBq. A different maximum total activity would be acceptable if the waste in the ENRMF contained a different radionuclide mix. The actual mixture of radionuclides is recorded for each load deposited and compared with a 'running total' of the activity deposited to date which ensures that the consented total activity is not exceeded regardless of the mixture of radionuclides which are actually deposited. This is how the sums of fractions approach is applied and regulated.

Careful control of the activity and quantities of waste disposed, use of best practice design, the existence of a sound environmental management culture, and ongoing environmental monitoring will provide confidence that any radioactive emissions will be low and consistent with the health protection standards specified by the Environment Agency.

The ENRMF will continue to be monitored and regulated to confirm that it is operating in compliance with all appropriate International, European and national health and safety standards. Environmental monitoring during the operational and aftercare phases while the site is managed will check that the levels of radiation in a range of potential exposure pathways such as landfill gas, air emissions, leachate, surface water, ground water and dust will not exceed the criteria that are set for the site. Samples are taken to an agreed programme specified in the Permit and follow protocols set by the EA, with the resulting monitoring data reported to it. The EA currently undertakes its own independent sampling programme. The monitoring regime provides assurance that the site is performing as expected and that the design, construction and operating standards of the site are effective in eliminating or controlling any exposure risks.

Augean will continue to engage with the local community through the KCLG. This has involved annual open days, a twice yearly newsletter and maintenance of a register of stakeholders. The KCLG has been kept up to date with the programme for this application to vary the radiological Environmental Permit.

Contents

Tables and Figures

1 Introduction

1. This document is an Environmental Safety Case (ESC) that supports a request for a variation to Environment Agency Permit number CD8503, for receipt and disposal of low level radioactive waste at the East Northants Resource Management Facility (ENRMF), Stamford Road, King's Cliffe, Northamptonshire, PE8 6XX, United Kingdom (the centre of the site lies approximately at OS Grid Reference TF 0084 0002, 52.5887^{o} N 0.5130^{o} W).

2. Augean South Limited (Augean) is the operator of the ENRMF which comprises a hazardous waste treatment facility at which materials are recycled, recovered and hazardous properties reduced and a landfill at which a range of hazardous wastes and low activity radioactive waste is disposed. The Environment Agency Permit number CD8503 covers disposal in cells 4B, 5A and 5B of the landfill. On 11th July 2013, the Secretary of State (The East Northamptonshire Resource Management Facility Order, 2013) approved the extension of the ENRMF to include an additional void of 1.2 10^{6} m^{3} (1.2 million cubic metres) over an area of approximately 11 ha (hectares) and an increase in the annual capacity of the treatment facility to 150,000 t y^{-1} (tonnes per year). The order permits disposal of 150,000 t y^{-1} of hazardous and low level radioactive waste (LLW) direct to landfill. It states that radioactive waste, to a maximum specific activity of 200 Bq g^{-1} (Becquerel per gramme) may be disposed in cells 4B, 5A and 5B and Phases 6 to 11 (see Figure 1). LLW input to the site is capped at 448,000 t (tonnes). The application this document supports is to extend the Environmental Permit for the LLW disposal area to include Phases 6 to 11 as well as cells 4B, 5A and 5B.

3. The guidance on requirements for authorisation of near-surface disposal facilities for solid radioactive wastes (NS-GRA) has been used as the basis for the analysis in this ESC (Environment Agencies, 2009). The NS-GRA contains fourteen requirements, of which Requirement 3 of the NS-GRA is for an ESC:

> *"An application under RSA 93 relating to a proposed disposal of solid radioactive waste should be supported by an environmental safety case." NS–GRA (Environment Agencies, 2009) para 6.2.1*

Document structure

4. An ESC provides a safety assessment and related safety arguments that bear on the acceptability of proposed disposals of radioactive waste at a facility and it is required to demonstrate that members of the public and the environment are adequately protected and be proportionate to the hazard presented by the waste. The section titles of this ESC indicate where each NS-GRA requirement is addressed, for example Section 4.1 has the title "Process by Agreement {R1}" indicating where Requirement 1 is addressed. The relevant sections, as numbered, are listed below:

- 4.1 Process by Agreement {R1}
- 4.2 Dialogue with Local Communities and Others {R2}
- 5.1 Environmental Safety Case {R3}
- 5.2 Environmental Safety Culture and Management System {R4}
- 6.1 Dose constraints during the period of authorisation {R5}
- 6.2 Risk guidance level after the period of authorisation {R6}

- 6.3 Human intrusion after the period of authorisation {R7}
- 6.4 Optimisation {R8}
- 6.5 Environmental radioactivity {R9}
- 7.1 Protection against non-radiological hazards {R10}
- 7.2 Site investigation {R11}
- 7.3 Use of site and facility design, construction, operation and closure {R12}
- 7.4 Waste acceptance criteria {R13}
- 7.5 Monitoring {R14}

5. The location of the ENRMF and the local environment are described in Section 2 of the ESC with waste characteristics detailed in Section 3. The contents of Sections 4 to 7 cover the NS-GRA requirements as listed above and Section 8 draws together the safety assessment and related safety arguments. The rest of this section provides background information on LLW management within the United Kingdom (UK), summarises existing site permits, describes ENRMF development plans and then briefly describes the proposed permit variation. The last part of this section describes the environmental safety strategy (ESS) set out in the ESC.

1.1 Background

6. Within the UK, LLW is defined by Government policy as:

> *"radioactive waste having a radioactive content not exceeding four gigabecquerels per tonne (GBq/te) of alpha or 12 GBq/te of beta/gamma activity". (Defra, DTI and the Devolved Administrations, 2007)*

7. There is a sub-classification of LLW referred to as high volume very low level radioactive waste (HV-VLLW) that is defined as:

> *"Radioactive waste with maximum concentrations of four megabecquerels per tonne (MBq/te) of total activity which can be disposed of to specified landfill sites. For waste containing hydrogen-3 (tritium), the concentration limit for tritium is 40MBq/te. Controls on disposal of this material, after removal from the premises where the wastes arose, will be necessary in a manner specified by the environmental regulators". (Defra, DTI and the Devolved Administrations, 2007)*

8. The previous permit application was for receipt and disposal of LLW including HV-LLW and reference to LLW throughout this document is assumed to include this lower activity waste classification.

9. The use of landfill is an established approach to the disposal of waste with low specific activity and is supported by Government policy (Defra, DTI and the Devolved Administrations, 2007). The UK strategy for the management of solid LLW from non-nuclear sources is presented in two parts; the first considers anthropogenic radionuclides (Defra, 2011b) and the second part (DECC, 2014) deals with naturally occurring radioactive materials (NORM). Disposal of LLW to landfill is authorised as a radioactive substances activity under the Environmental Permitting (England and Wales)

Regulations 2010 [EPR 2010; (UK Statutory Instrument, 2010)] using permits issued by the Environment Agency in England.

10. Disposal routes for LLW are limited in the UK. The majority of LLW continues to be sent to the Low Level Waste Repository (LLWR), located near the village of Drigg in Cumbria. The UK is predicted to generate significantly more LLW than the planned disposal capacity at the LLWR, resulting in a need for alternative ways to manage LLW, including the use of alternative disposal routes. The Nuclear Decommissioning Authority (NDA) strategy recognises that the use of the LLWR, given its limited capacity, is likely to need prioritising in order to maximise the lifetime of the facility (Nuclear Decommissioning Authority, 2011). This is consistent with the UK nuclear industry recognition that to meet all the LLW disposal requirements, alternative disposal options may be required for appropriate waste streams (Nuclear Decommissioning Authority, 2010).

11. The LLWR does not therefore have capacity for the expected volumes covering the full range of LLW (up to 4000 Bq g^{-1} alpha activity and 12,000 Bq g^{-1} beta/gamma activity) that will be generated by the nuclear industry (Nuclear Decommissioning Authority, 2013). The disposal of LLW at the lower end of the range of specific activity is not a sustainable use of the repository, which has been designed and engineered to a standard suitable for materials with a radioactive content at the higher end of the range for LLW. The strategic need for alternative fit for purpose disposal routes is established and detailed within the UK nuclear industry LLW strategy (Nuclear Decommissioning Authority, 2010) and for the non-nuclear industry in UK Government policy (Defra, DTI and the Devolved Administrations, 2007). This is reinforced by recent management strategies developed for waste generated by non-nuclear industries in the United Kingdom concerning anthropogenic radionuclides (DECC, 2011) and NORM (DECC, 2014).

12. The ENRMF is centrally located for the wastes arising at the locations of the major LLW waste producers in the south and east of the country (Figure 2). The location of the site is well placed to serve the producers of LLW from the nuclear and non-nuclear industries. For many of the LLW producers who dispose of their LLW currently at the LLWR near Drigg the ENRMF provides a closer and more convenient alternative.

13. The LLW that will be considered for disposal at the ENRMF contains very small amounts of radioactivity; with a specific activity less than or equal to 200 Bq g^{-1}. The waste can be handled safely by workers in a manner similar to other low hazard wastes. Although the material is radioactive waste by legal definition, the accepted waste will contain a fraction of the specific activity limits specified for LLW, with 200 Bq g^{-1} representing less than 5% of the limit for alpha activity and 1.7% of the limit for beta/gamma activity. These wastes do not need special security measures.

Figure 1. The current site layout and permit boundaries

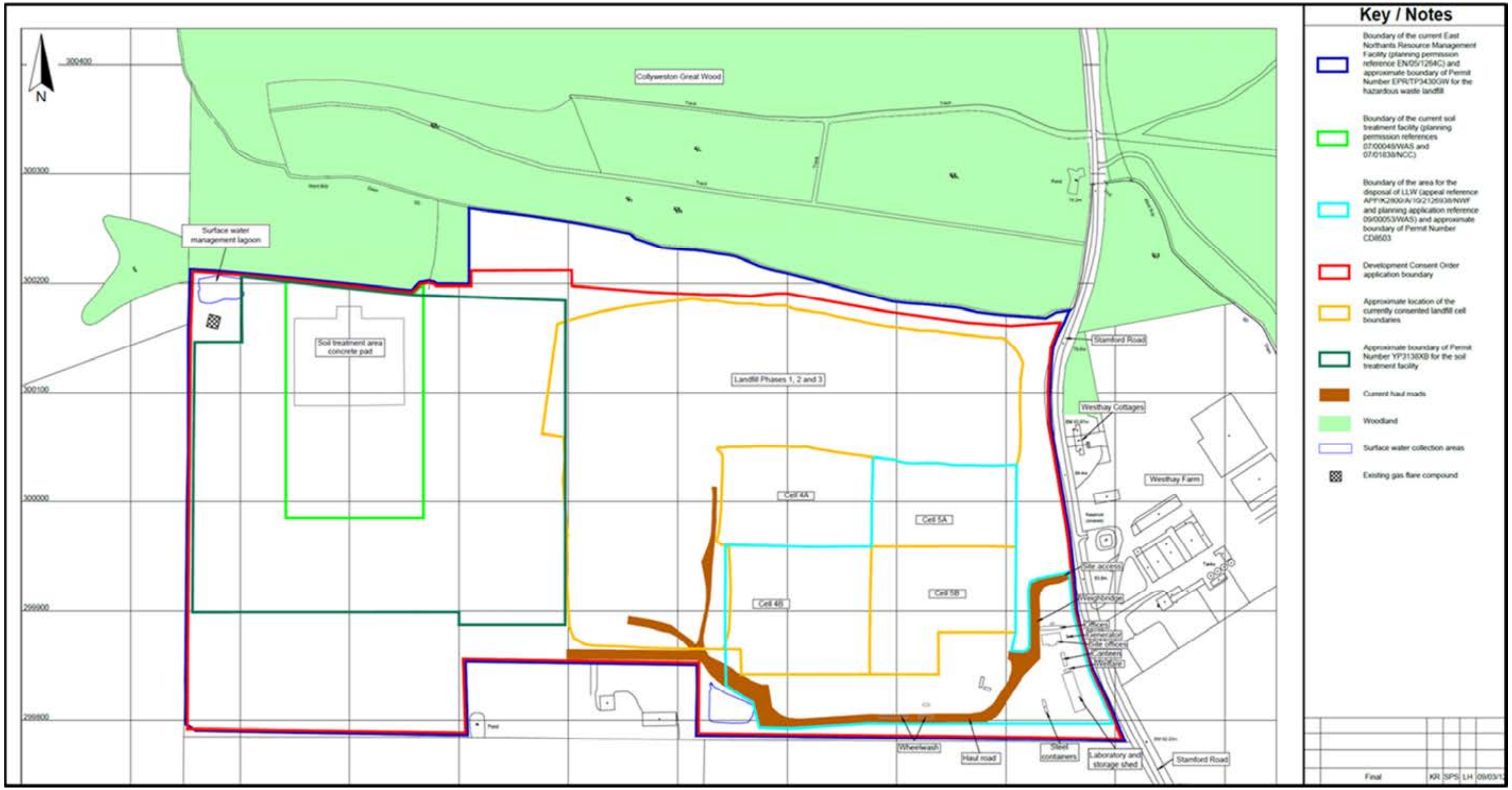
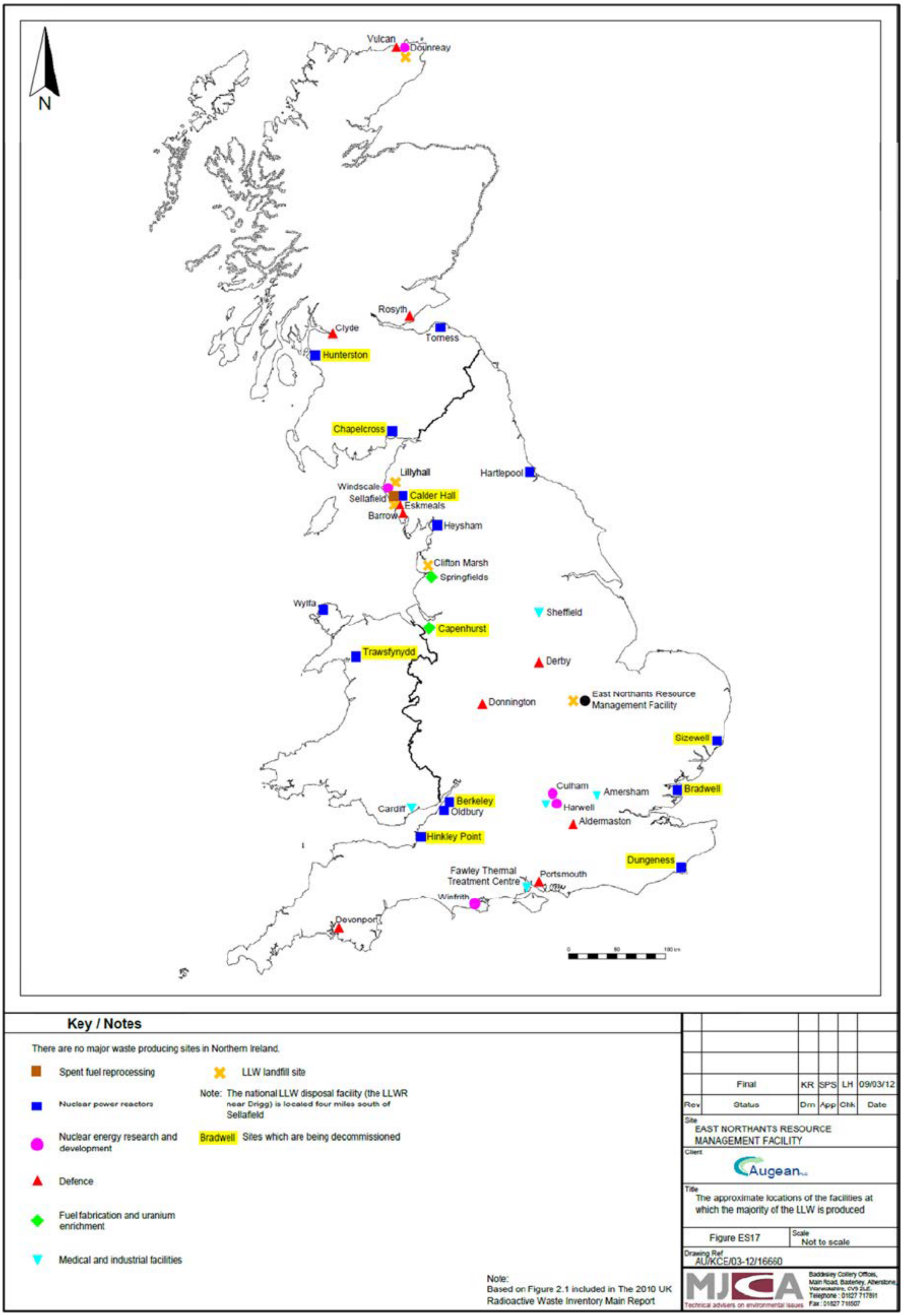


Figure 2. Approximate locations of the facilities at which the majority of the LLW is produced



1.2 Existing site status

Planning permission

14. The site is the subject of a Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013) made by the Secretary of State on 10th July 2013 and coming into force on the 31st July 2013. The Development Consent Order was made for the continuation and extension of the disposal to landfill of hazardous waste and low level radioactive waste and the treatment of hazardous waste. The consent was implemented on 2nd December 2013.

Environmental Permit - Hazardous Waste Landfill

15. The ENRMF landfill is operating under an Environmental Permit (TP3430GW) issued May 2009, for the disposal of hazardous waste. The site commenced operations in 2002 under a PPC Permit and was originally a co-disposal site for non-hazardous and hazardous wastes. Since the beginning of 2004, the site has received predominantly hazardous waste and the practice of co-disposal has ceased. The site is therefore now a hazardous waste and LLW landfill apart from the need for suitable cover materials. The permit boundary for hazardous waste covers an area of approximately 31 ha with some 13 ha currently permitted for landfill. A variation is being applied for to extend the permitted area for landfilling by approximately 11 ha. The disposal of LLW in this extended permit area is the subject of the application.

16. The non-radioactive wastes accepted at the ENRMF cover a broad spectrum of those defined as hazardous under the European Waste Catalogue and are subject to the hazardous waste acceptance criteria under the Landfill Directive (European Commission, 1999). These criteria in particular exclude explosive, flammable, corrosive and infectious materials.

Environmental Permitting – Hazardous Waste Treatment

17. The treatment facility is located in the north-west corner of the site. The treatment facility is the subject of Environmental Permit YP3138XB for stabilisation, soil washing and bioremediation. Currently the main activity at the treatment facility is stabilisation. Residues are disposed of to landfill.

Environmental Permitting – Low Level Waste

18. The disposal of low level radioactive waste up to 200 Bq g^{-1} in the hazardous landfill is the subject of Environmental Permit reference CD8503 issued in May 2011. Disposal commenced in December 2011. The disposal of LLW is permitted in Phases 4B, 5A and 5B.

1.3 Site development plans

19. The ENRMF landfill site is operated as a hazardous waste and low level radioactive waste disposal facility. The newly consented void space is approximately 1 10^6 m^3 of which up to approximately 20% is allocated for LLW. The planning consent requires landfill restoration to be completed by the end of 2026. The maximum consented hazardous waste and LLW tonnage accepted at the site is 250,000 t y^{-1}, with additional limits of 150,000 t y^{-1} direct disposal to landfill, a limit of 150,000 t y^{-1} hazardous waste processed at the treatment

facility and a total site limit for LLW of 448,000 t. The limit on LLW tonnage was based on a 20% cap on the proportion of the void space that could be used for LLW (The Planning Inspectorate, 2013).

20. The landfill is designed and constructed to a high level of containment engineering using low permeability clay and a high density polyethylene (HDPE) flexible membrane lining system (Augean, 2014), which meets the regulatory requirements under the Landfill Directive (Defra, 2010). The landfill is operated in a series of 11 phases. Phases 1 to 4 have been filled (see Figure 1). Current operations are in Phase 5. LLW has been disposed only in phases 4B, 5A and 5B. Phases 6 to 11 represent the extension area and are not included in the current Environmental Permit for radioactive substances. The permit application extends the disposal area to include Phases 6 to 11 as discussed below (see Section 1.4).

21. Each phase of operation is progressively restored under a defined scheme of capping and restoration. In accordance with the Development Consent Order the landfill site will be restored to grassland and woodland for ecological and amenity use.

22. Operating details for the site are not presented here and are available in the supporting documentation for the existing permitted operations (Augean, 2012a). There are about 110 separate operating procedures and risk assessments relating to the hazardous waste operations. The operating arrangements and culture at the site are consistent with the arrangements proposed for LLW disposal in the application.

1.4 The Proposal

23. In order to realise the benefits of the development consent it is necessary to vary the existing Environmental Permits. Accordingly Augean is seeking a variation to Environmental Permit number CD8503 for the receipt and disposal of low level radioactive waste at the landfill. The proposed boundary for LLW disposal under the permit variation is shown in Figure 1.

24. Other Environmental Permits will be varied under separate applications. An updated Hydrogeological Risk Assessment [HRA, (Augean, 2014)] has been produced.

25. A permit variation is sought to allow receipt and disposal of radioactive waste to the landfill extension (phases 6 to 11) in addition to the currently permitted cells (4B, 5A and 5B). We request that the current permit limitation allowing acceptance of LLW to a maximum specific activity of 200 Bq g^{-1} should be maintained. Disposed wastes will otherwise be compliant with Augean's Conditions For Acceptance (CFA) specified in site procedure LLW01 (see Section 7.4.3) relating to the non-radioactive properties of the waste (i.e. the proposal is for the disposal of radioactive wastes that would be classified as inert, non-hazardous or hazardous in terms of their content of non-radioactive materials). The radioactive waste disposals would not be segregated from other, non-radioactive wastes disposed in the ENRMF.

26. The approach presented here is based around a proposed maximum tonnage of LLW (448,000 t) and a specific activity limited to 200 Bq g^{-1}. The current permit does not include a maximum tonnage.

27. The proposed variation would involve changes to Table 1 of the current permit which lists 43 radionuclides and provides an absolute disposal limit in GBq (Giga Becquerel) for each.

A replacement table is proposed using the same radionuclides with new values inserted based on the assessments reported in this ESC. It is also intended that a condition of any new permit will require the operator to calculate, for each radionuclide or group of radionuclides listed, the ratio of the activity of the radioactive waste disposed of at the ENRMF to the relevant value in the new table. It will be a condition of any new permit that the sum of these ratios shall be less than 1. This sum of fractions approach (detailed in Section 7.4.2) allows the operator greater flexibility in determining the final radioactive waste inventory without compromising environmental safety. The sum of fractions approach has been used in other recent permits (e.g. CD7914 for the Lillyhall landfill site).

28. It is proposed that the specific activity of 200 Bq g^{-1} applies to a consignment. Based on records to the end of 2013, the waste streams consigned for disposal at the ENRMF have an average specific activity across all LLW consignments of less than 10 Bq g^{-1}. The current permit does not specify an averaging tonnage but the specific activity is recorded on a consignment basis.

29. The minimum depth of non-radioactive waste or material covering LLW and the constraining time periods for disposal or cover to be in place remain the same as in the current permit at 0.3 m (metre) and 8 h (hours), respectively. However, operating procedures have been updated and now include specifications on the depth of non-radioactive waste that will be placed at the base (2 m), sides (2 m) and top (1 m) of a landfill waste cell. An additional limitation is proposed for wastes containing a significant quantity of Ra-226 (Radium contaminated wastes) with a requirement to bury these wastes at least 5 m below the restored surface of the site. The proposed criterion for wastes containing a significant activity concentration of Ra-226 is waste containing >5 Bq g^{-1} Ra-226. The current permit does not specify where waste will be placed in waste cells.

30. The ESC assessments supporting the variation make specific reference to NORM and the impact of radionuclide distributions in waste forms.

1.5 Environmental Safety Strategy

"The Fundamental Protection Objective is to ensure that all disposals of solid radioactive waste to facilities on land are made in a way that protects the health and interests of people and the integrity of the environment, at the time of disposal and in the future, inspires public confidence and takes account of costs." (Environment Agencies, 2009) para 4.2.1

31. The objective is therefore to dispose of wastes to the ENRMF in such a way as to ensure that impacts to people and to the environment are maintained at levels, both in the short and long-term, which afford a high level of protection, based on current limits, targets and guidance, without any reliance on waste retrieval or other intervention measures.

32. This will be achieved through the use of both engineered and natural barriers to contain the disposed radionuclides for as long as reasonably practicable and thereafter limit the rate at which any radionuclides are released to the accessible environment.

33. The NS-GRA requires an environmental safety strategy that is supported by an ESC. Such a strategy should:

"… present a top level description of the fundamental approach taken to demonstrate the environmental safety of the disposal system. It should include a clear outline of the key

environmental safety arguments and say how the major lines of reasoning and underpinning evidence support these arguments." (Environment Agencies, 2009) para 7.2.2

34. The strategy to achieve the objective of low impacts at all times following waste disposal consists of disposing of wastes that represent a low inherent risk due to their relatively low specific activity and a restriction on the total quantity that can be disposed at the ENRMF. Such wastes will be disposed to a facility that:

- has been in operation since 2002;
- is based on well tried and tested technologies;
- is robust and incorporates multiple engineered barriers and safety functions;
- is regularly reviewed for compliance with current standards as subsequent phases for developing disposal cells are planned;
- is subject to active management control; and,
- maximises use of passive safety features.

35. The overall safety strategy for the disposal of LLW at the ENRMF involves both active (operational) management and the construction of passive barriers ensuring that wastes disposed of will give rise to low impacts, within the dose and risk guidance levels laid down in the NS-GRA (Environment Agencies, 2009). The following steps will be taken:

- limits will be set on the specific activity in each consignment and the total activity to be disposed (the total tonnage of LLW that can be accepted is already limited by the planning consent);
- Waste Acceptance Criteria (WAC) will be specified, covering radiological and non-radiological properties of the wastes, and a written specification of acceptable waste types will be provided to any person seeking to dispose of waste at the ENRMF (the CFA);
- waste inventory regulated using a sum of fractions approach;
- hazardous waste landfill design with fit-for-purpose disposal cells with basal and wall liners as well as a low permeability capping layer provide an engineered barrier, reducing leachate flow over periods of many decades or centuries;
- work management culture and safety procedures ensure that wastes are transported and handled safely reducing the potential for dose impact to the workforce and the risk of accidents leading to unplanned impacts on the environment;
- active collection of leachate during and following the operational period and use on site at the treatment facility or transported for discharge via a sewage treatment plant reduce the risk of contamination of groundwater in the vicinity of the disposal site;
- the wastes will be covered immediately to reduce dust suspension and hence the risk of impacts via the inhalation pathway during the operational period;
- cell caps will be constructed once disposal cells are full, eliminating dust resuspension and reducing water ingress, and hence reducing potential leachate generation;

- environmental monitoring during the period of authorisation will check the integrity of barriers and safety plans;
- scenarios involving exposure to waste during normal operations and expected site evolution have been considered ensuring doses or risks remain below the relevant dose and risk guidance levels
- a full range of scenarios involving unplanned exposure to waste have been considered, in order to ensure that for all reasonably foreseeable circumstances doses or risks remain below the relevant dose and risk guidance levels; and,
- the impact of uncertainty in estimated doses and risks has been considered to demonstrate that the ESC is robust in meeting all relevant dose and risk guidance levels.

36. Waste retrieval is not planned as this ESC relates to waste disposal (see NS-GRA (Environment Agencies, 2009), para 3.6.2). Nonetheless, retrieval would be feasible both in the short and longer term if required. This provides an assurance of last resort that, in the event that an unforeseen (and unacceptable) impact should occur, intervention to reduce or eliminate the impact could be undertaken. It is emphasised, however, that such an action does not form part of this ESC and it is considered that under all foreseeable circumstances it will not be necessary nor should it form any part of contingency planning.

2 Site Characteristics

2.1 Introduction

37. The NS-GRA (Environment Agencies, 2009) requires that the site characteristics including the geological environment and the biosphere are characterised, understood and capable of analysis to the extent necessary to support the ESC. Such characterisation has been undertaken (Augean, 2012a) and is the basis for the description set out in this section.

38. This description draws on the Environmental Statement presented in support of the development consent application (Augean, 2012b). This section presents a summary of the understanding of the characteristics of the site, including information on the physical setting, land use and hydrology, and of the regional and local geosphere including lithology, stratigraphy, resource potential, hydrogeology and geochemistry relevant to the assessment of the proposed disposal facility. Consideration of the potential for disruption under reasonably foreseeable future conditions is also presented.

2.2 Location

39. The ENRMF lies approximately 1.7 km (kilometre) east south east of Duddington and approximately 2.6 km north of King's Cliffe village in the East Northamptonshire district of Northamptonshire (Figure 3). The setting is generally rural with the majority of the land surrounding the site comprising open farmland or woodland. The site occupies approximately 31 ha and is within the boundary of the area which is the subject of the current planning consent (Figure 1). The land in the application area is owned by Augean. An aerial photograph of the site is presented as Figure 4.

Figure 3. The site location

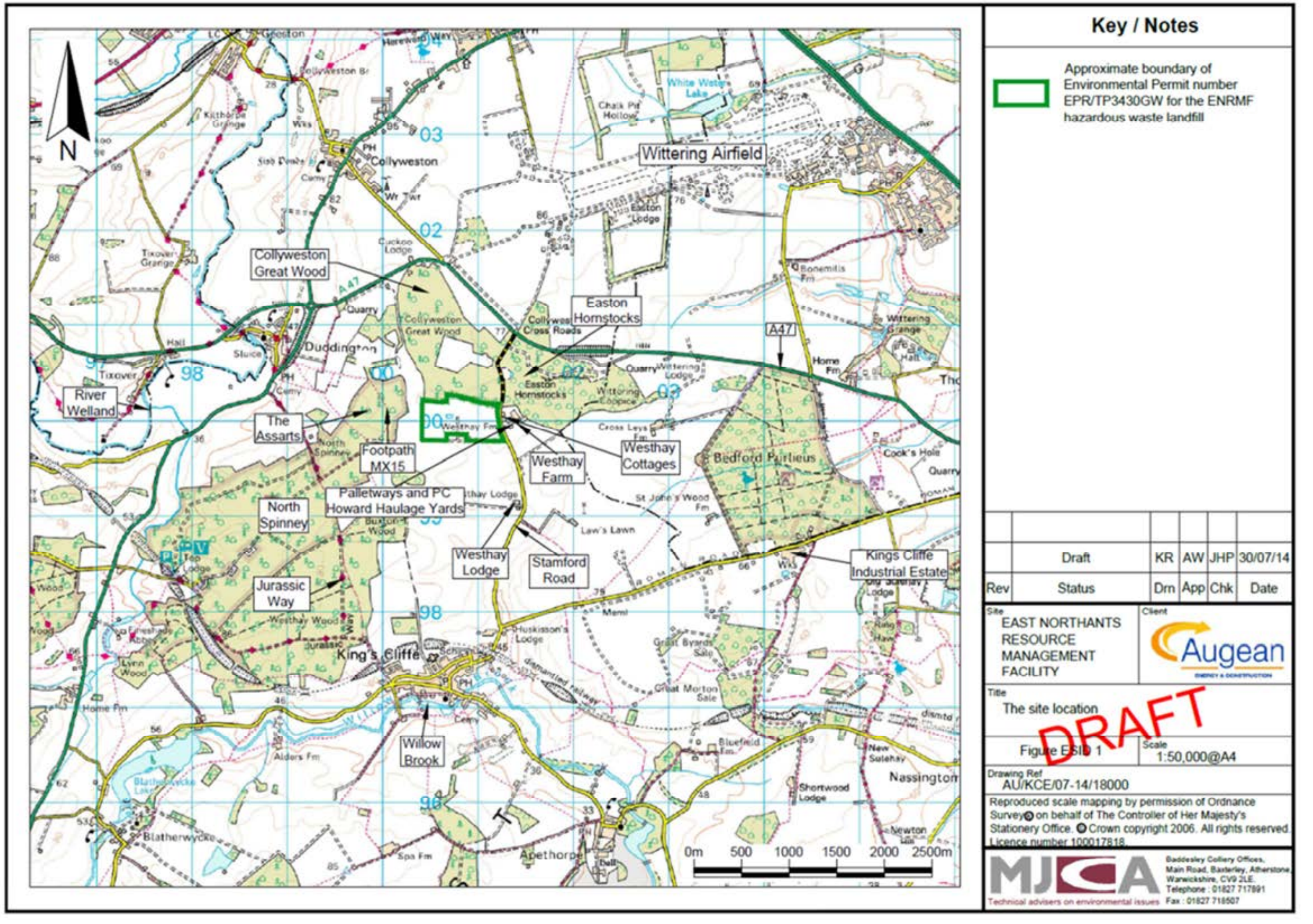


Figure 4. Aerial view of the site showing Development Consent Order boundary

2.3 Landfill History, Design and Use

40. The site commenced operations in 2002 under a PPC Permit and was originally a co-disposal site for non-hazardous and hazardous wastes. Since the beginning of 2004, the site has received predominantly hazardous waste and the practice of co-disposal has ceased. The permit boundary for hazardous waste covers an area of approximately 31 ha with some 13 ha currently permitted for landfill. A variation is being applied for to extend the permitted area for landfilling by approximately 11 ha.

41. The site comprises an operational hazardous waste and LLW landfill including restored and partially restored landfill areas together with a soil treatment and recycling facility. A surface water management facility and a landfill gas management compound including a flare stack are located in the north western corner of the site. Site infrastructure including the site access, waste reception facilities, car parking areas, site offices, welfare facilities, storage areas, laboratories together with wheel and vehicle body washing facilities are in place at the site (Figure 5).

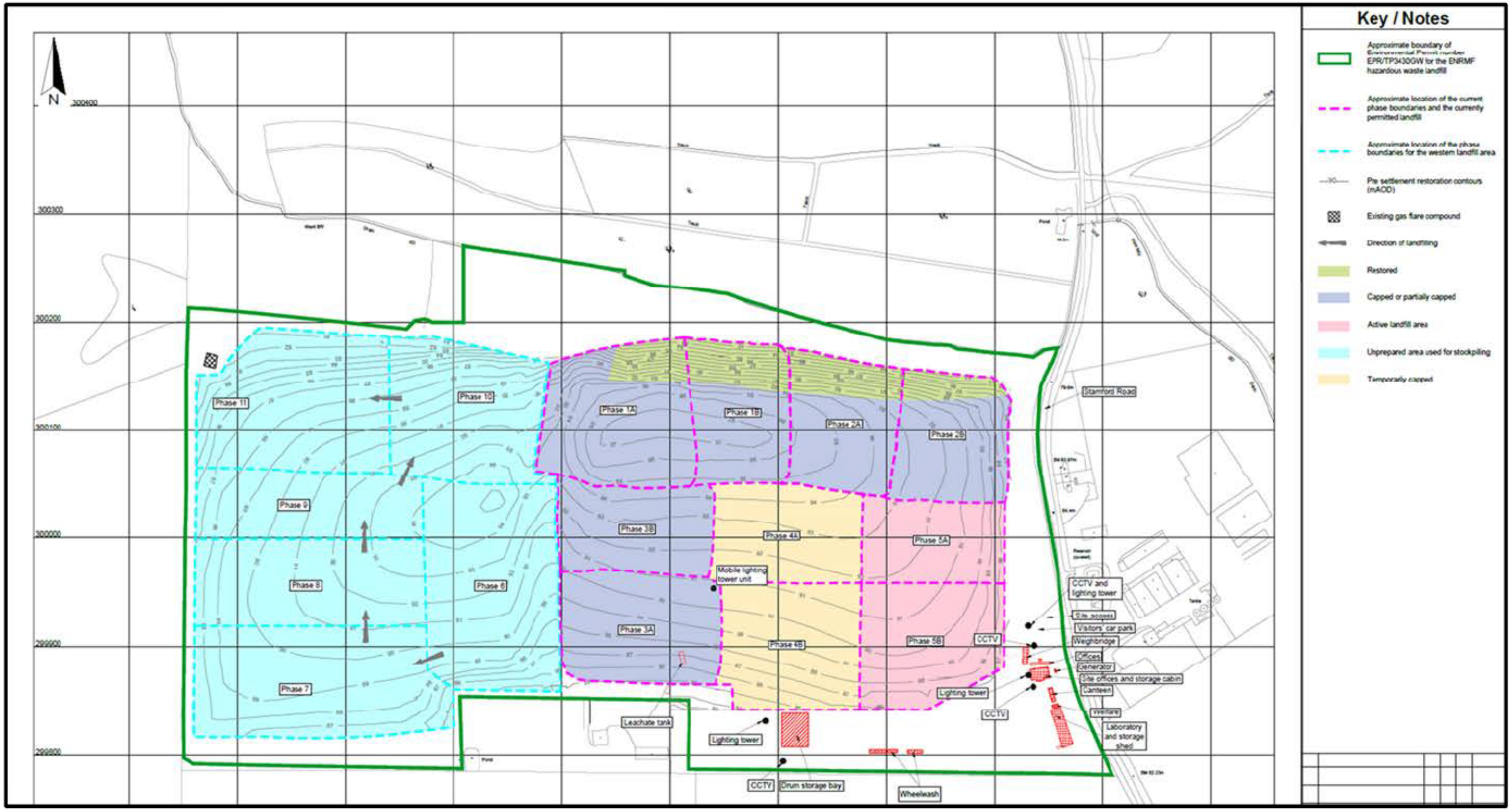
42. The current landfill comprises 5 phases of landfilling with each phase of landfilling subdivided into two cells. At the present time Phases 1, 2 and 3 in the current landfill have been filled, Phases 3A and 3B and parts of Phases 1A, 1B, 2A and 2B have been capped, Phases 4A and 4B are filled and covered with a temporary cap and Phase 5 is currently being filled. Soil has been stripped from the western area of the site to the west of the landfill area and this area is used currently for the soil treatment facility including a concrete pad and associated storage areas and for the storage of clay and overburden. The land to the west of the current landfill has the benefit of planning permission for landfill and this area is the subject of the application to vary the existing permit. The landfilling will continue in this area in a series of phases numbered 6 to 11 (Figure 5).

2.3.1 Design and Construction

43. The landfill site is designed and operated based on the principle of engineered containment with low permeability basal, perimeter and capping seals constructed to an engineering specification which is the subject of approval by the Environment Agency under the Environmental Permit for hazardous waste disposal and the Landfill Directive (European Commission, 1999). Clay is extracted during the development of the site and together with currently stockpiled clay is used in the construction of the containment system for the landfill cells.

44. The landfill site will continue to be operated on the principle of containment. This means that the cells will be lined with an engineered low permeability barriers designed to retain contaminants within the site. A series of cells will be filled, capped and restored progressively. To separate the wastes from the surface environment and to minimise the infiltration of rainfall the landfill will be capped with low permeability layers overlain with restoration materials.

Figure 5. Site facilities



45. Construction of Cell 5B is complete and comprises a void excavated to the top of the underlying limestone. A base of at least 1.5 m thickness of engineered low permeability clay with a hydraulic conductivity of less than 3 10^{-10} m s^{-1} (metres per second) has been constructed. The engineered clay is covered by an artificial sealing liner comprising a 2 mm (millimetre) thick HDPE geo-membrane with a leachate drainage layer of 500 mm of crushed aggregate or shredded tyres above. The liner specification for the currently consented landfill is agreed with the Environment Agency in accordance with the Environmental Permit for hazardous waste disposal.

46. For the western landfill area, in situ low permeability Rutland Formation clay 2 m thick will be left in place above the top of the Lincolnshire Limestone. The engineered basal liner will be constructed on top of the in situ clay. Prior to the construction of the engineered liner geophysical surveys will be carried out to identify potential solution features as is the case currently. The western landfill area will be prepared by removing all excess clay from the fill area. No stripping of soil will be necessary as there is no undisturbed soil in the western landfill area. Overburden that is not suitable for use as engineering clay will be stockpiled for use as cover material and during the restoration of the site or removed from the site for use elsewhere.

47. The western landfill area will be developed in 6 phases. The landfill will be constructed in accordance with the engineering specifications of the European Union (EU) Landfill Directive (European Commission, 1999) which are implemented in the UK through the EPR 2010 together with Environment Agency technical guidance (Environment Agency, 2011b). Cell construction in the western landfill area will comprise at least 1 m thickness of low permeability engineered clay, an artificial sealing liner comprising a 2 mm thick HDPE geo-membrane and a leachate drainage layer of crushed aggregate or shredded tyres above the basal low permeability seal.

48. The design of the low permeability capping layer at the site will be agreed with the Environment Agency and will comprise the following elements or alternative specification providing equivalent or greater protection: a composite cap consisting of a regulating layer of approximately 0.3 m over the top of the waste, a low permeability geo-synthetic clay liner, a low density polyethylene geo-membrane liner, a 300 mm granular drainage layer and 1 m to 1.5 m of restoration materials. A temporary cap is placed over filled cells prior to final capping.

49. The nature of the site containment including the basal and side wall lining system and the capping layer will be specified through the revised Environmental Permit for hazardous waste disposal. The landfill cells and capping layers in each phase will be constructed in accordance with the Environmental Permit and will be the subject of Construction Quality Assurance (CQA) Plans and protocols to ensure that the agreed specifications have been achieved. The final profile of the waste and capping layer is designed to form a stable slope which will encourage shedding of rainfall to minimise infiltration and as a consequence to minimise leachate generation.

2.3.2 Leachate Management

50. Leachate is formed as a result of the release of liquids entrained in deposited wastes and following the infiltration of rainfall through the waste. The engineered landfill containment system will include a leachate management system for the collection and extraction of leachate. A leachate drainage blanket and collection sumps will be constructed at the base

of the site immediately above the low permeability basal liner. The leachate levels will be controlled by pumping leachate from the leachate collection sumps or other extraction wells drilled as necessary. The level at which the leachate is maintained will be specified in the Environmental Permit.

51. The leachate generated at the site will not be used for dust suppression. The excess leachate will be pumped into a leachate storage tank and used in the on-site waste treatment facility in place of clean water. If the leachate is not needed in the on-site waste treatment facility it will be removed from site by tanker for treatment at a suitably authorised waste water treatment plant. The current location of the leachate storage tank is shown on Figure 5. Risk assessment in this ESC shows that even under conservative assumptions the level of activity that could accumulate in the leachate will not exceed relevant dose limits for workers or the public during treatment. Leachate is monitored for chemical and radiological characteristics to confirm that the contaminants remain below the levels specified in the risk assessment.

2.3.3 Landfill Gas Management

52. The waste types accepted prior to July 2004, which is when the limitation on the organic content of landfilled hazardous wastes was implemented, have the potential to generate significant quantities of landfill gas. The hazardous wastes that are currently and will continue to be deposited at the site will have a limited organic carbon content however there is residual potential for the generation of small quantities of landfill gas and volatile organic compound vapours at the site. The LLW wastes that will be disposed of at the site have a generally low level of organic matter and are only slowly degradable, if at all. Putrescible materials are not accepted. The levels of radioactivity in LLW are too low to give rise to a risk from radiolytic hydrogen gas evolution. As a precaution the site operates a gas management system that is able to manage any gas generated from the waste. It is unlikely that significant quantities of landfill gas will be generated from LLW that will be deposited at the site. If gas is generated by the hazardous waste and/or LLW, the gas will be collected in the gas management system and directed to the gas flare for combustion.

53. A dual system of migration control will continue to be operated at the site. The engineered low permeability basal and sidewall liners impede lateral gas and vapour migration and the low permeability cap reduces the emissions to the atmosphere. A pumped landfill gas extraction system will continue to be operated as necessary which prevents the accumulation of gas under elevated pressures in the landfill minimising further the risk of the migration of gas and the emissions of gas to the atmosphere. The collected gas will continue to be directed to the gas flare to the north west of the landfill and burnt in a high temperature flare. Combustion of the gas destroys potentially harmful and odorous components in the gas and minimises the release of methane. The location of the landfill gas pumping system and flare stack are shown on Figure 5 and are surrounded by 1.8 m high fencing. The maximum height of the flare stack is 10 m. The gas flare and pumping facility will remain at the site beyond the completion of landfilling.

2.3.4 Surface Water Management

54. Clean surface water that has not been in contact with waste will continue to be collected in a series of drainage ditches. The surface water management system is set out in a scheme which is developed in accordance with the Environmental Permit for the landfill and which has been approved by the Environment Agency. The surface water management system

will continue to be installed progressively as landfilling continues. The surface water management system comprises a series of ditches which drain to a surface water management pond in the north west corner, the pond near the southern boundary or to a proposed pond in the south east of the site.

55. Currently surface water is used in the soil treatment facility, for dust suppression and in the vehicle wheel wash. No surface water is currently discharged as it is all used on site. In the event that not all the surface water is used on site it will be discharged to a drainage ditch adjacent to Stamford Road in accordance with the conditions set by the Environment Agency. The Environmental Permit requires any discharges are monitored and subject to limits. The ditches and ponds at the site have been designed to provide sufficient capacity to manage a 1 in 100 year rainfall event including an allowance for increases in rainfall as a result of climate change.

2.4 Restoration and After-use

56. The proposed final restoration landform for the proposed development is shown in Figure 6. The landform design takes into account various factors arising from the current site and from best practice in terms of landfill restoration. The landform is designed to integrate with the wider landscape character. The proposed maximum height and shape of the landform is similar to that of the previously permitted landform. The land use and planting scheme is based on a number of overarching principles and objectives for the restoration of the site.

57. It is proposed that the site is returned to a mix of woodland, scrub and species rich neutral grassland. Hedges will be introduced on the boundary of the site with occasional hedgerow trees. A permissive footpath for public access is proposed through the site with the potential to link westwards to The Assarts and the Jurassic Way subject to landowner agreements. The footpath is located just inside the woodland edge rather than outside it making the path a green lane. The boundary hedges will be hedges on banks with hibernacula. The green lane will also provide a more sheltered movement corridor for bats and insects. A viewpoint is proposed at the woodland edge looking southwards over the landscape. Ponds with associated reptile and amphibian hibernacula are proposed along the southern edge of the site. The ponds will feature a range of gradients and planting cover to offer the most scope for colonisation.

58. Parts of the site are already designated as a Potential Wildlife Site. The proposed restoration and after-care will secure and extend the designation giving long term protection from development.

Figure 6. Landform and landscaping of the restored site

Figure 7. Designated sites in the vicinity of the ENRMF

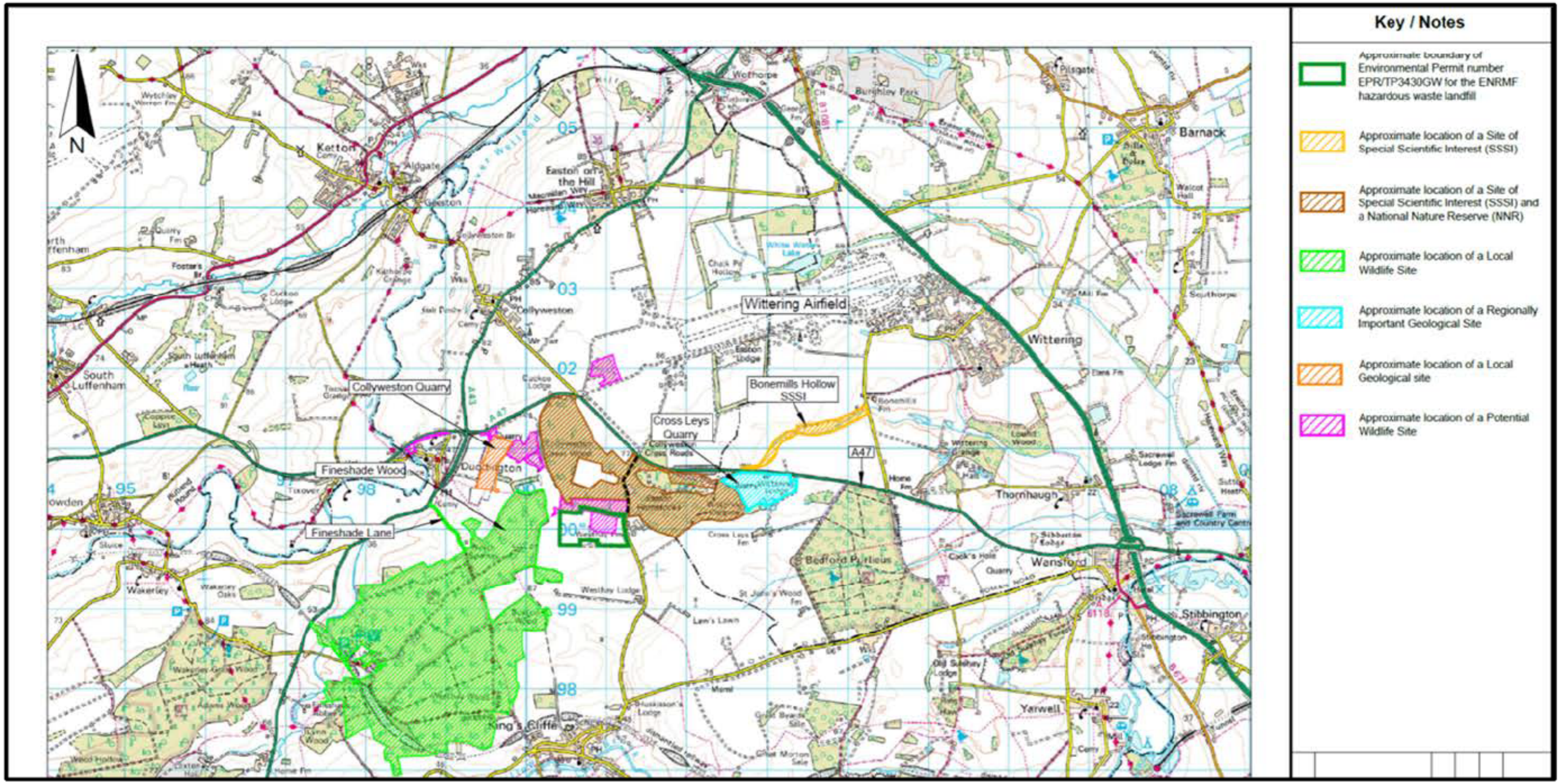
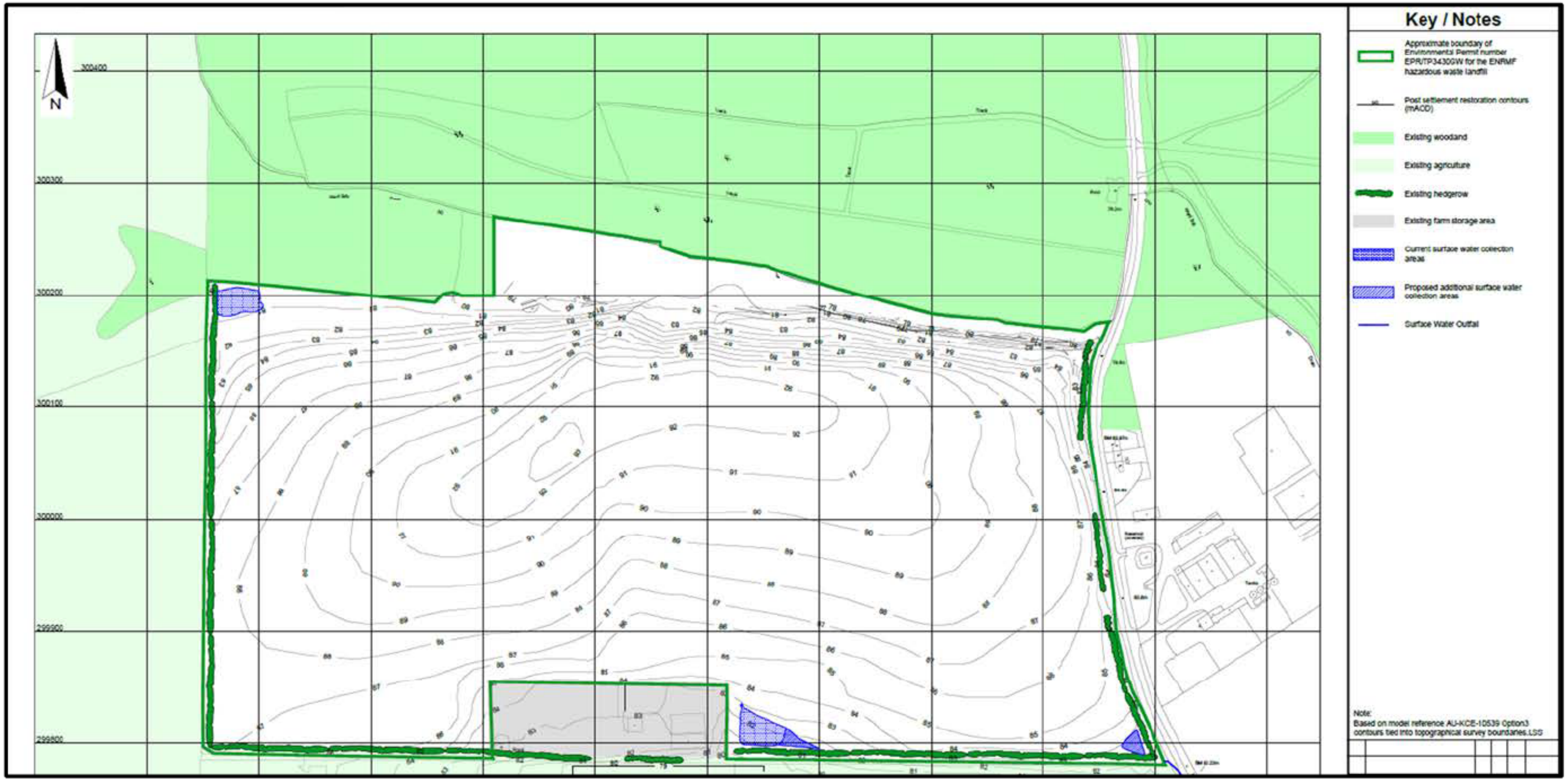


Figure 8. Surface water drainage plan for restored site



2.5 Local Environment

2.5.1 Site Perimeter

59. The ENRMF is bordered by a dense continuous thorn hedge on the eastern, western, southern and part of the northern boundaries. The remainder of the northern boundary is adjacent to extensive woodland. The soil treatment facility and the gas flare areas are fenced. A 1.8 m high fence is in place around the entire site boundary and there are gates at the site entrance which are locked outside operating hours.

2.5.2 Site Access

60. The current highway access to the ENRMF will continue to be used for the proposed development. The access is from Stamford Road which is a minor road that runs adjacent to the eastern boundary of the ENRMF from the A47 to the north and to King's Cliffe to the south. The access road enters the reception area adjacent to and south east of the landfill. The access is shown on Figure 3.

61. There are no public rights of way that cross or are adjacent to the application area. The public right of way closest to the application area (footpath MX15) is approximately 370 m west of the application boundary and passes through North Spinney Wood. The bridleway closest to the application area is located approximately 840 m south of the application boundary. The Jurassic Way footpath is approximately 1.3 km to the west of the site (Figure 3).

2.5.3 Settlements and Activities

62. The properties at Westhay Cottages are located approximately 25 m to the east of the application boundary and are the closest residential properties (Figure 3). Westhay Farm is located approximately 70 m east of the application boundary and is operated as a haulage yard and a farm with associated agricultural and commercial buildings. Westhay Lodge is located approximately 650 m south of the application boundary. A further residential property, Law's Lawn, is located approximately 1.2 km south east of the application area. To the west of the site there is open agricultural land and North Spinney Woods also known as The Assarts. The boundary of the airfield at RAF Wittering is located approximately 870 m to the north east of the application site.

2.5.4 Flora and Fauna

63. Adjacent to the northern boundary of the site is Collyweston Great Wood. To the east north east of the site is an area of woodland known as Easton Hornstocks. Parts of the Collyweston Great Wood and Easton Hornstocks comprise a Site of Special Scientific Interest (SSSI) and a National Nature Reserve (NNR). The north eastern part of the application site is designated as a Potential Wildlife Site (PWS). Figure 7 shows the designated sites in the vicinity of the ENRMF. The landfill lies within the Rockingham Forest/Lower Nene Valley Special Landscape Area, a local designation adopted by the County Council in 1974. This is an area of relatively level to gently undulating land at an elevation of approximately 85 m above Ordnance Datum. The predominant land uses within the immediate area of the site are agriculture and woodland.

2.6 Geology and Hydrogeology

64. A detailed description of the local geology and hydrogeology is given in the HRA (Augean, 2014) and detailed geological maps were produced in the 2004 HRA (Figure 2.9).

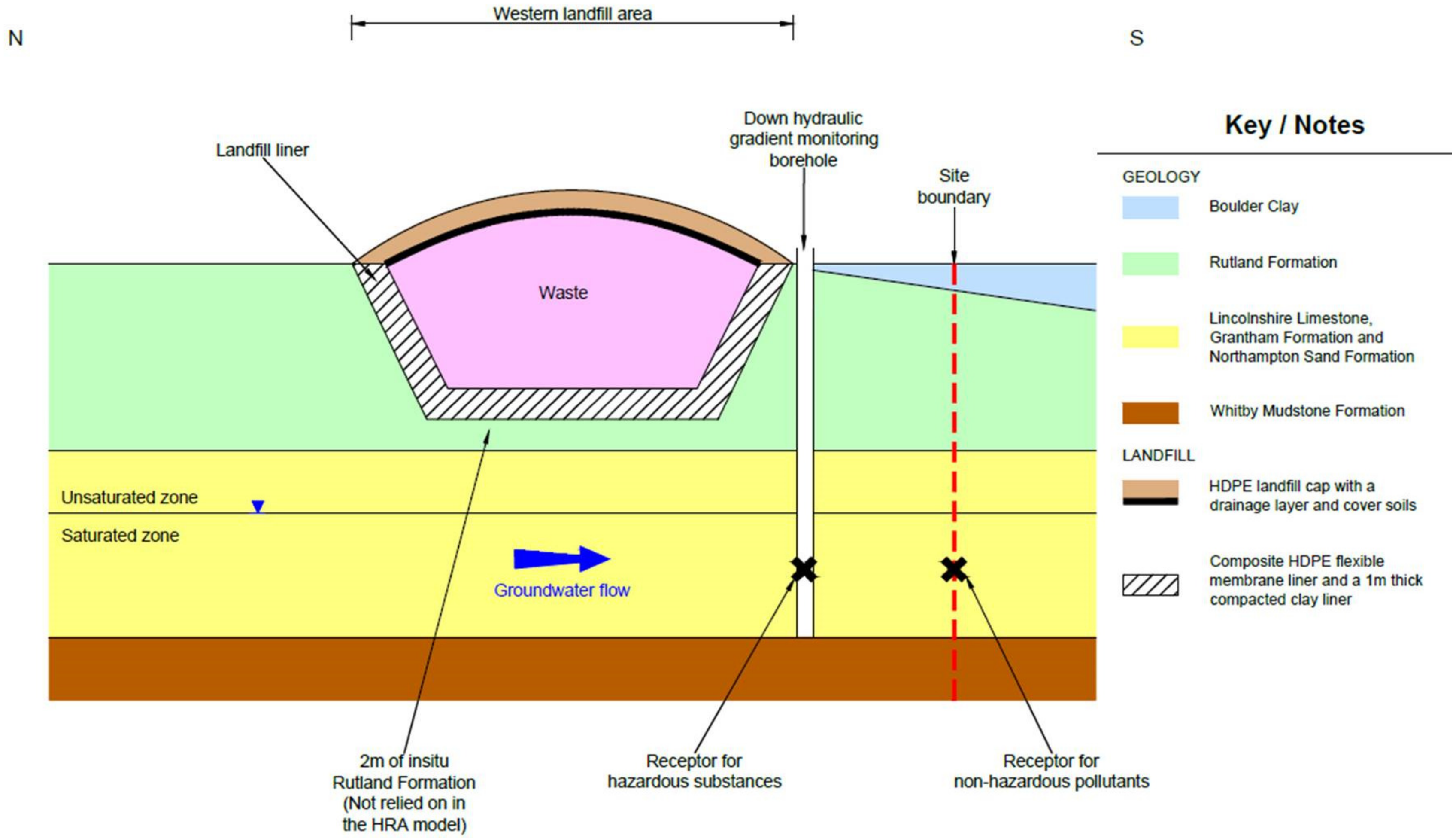
2.6.1 Geology

65. Drift deposits comprising boulder clay overlie the solid geology across parts of the ENRMF (the site). The solid geology comprises a thin layer of limestone comprising the Blisworth Limestone Formation in the south eastern corner of the site underlain by silty mudstone of the Rutland Formation (formerly referred to as the Upper Estuarine Series) of the Jurassic Great Oolite Series. The Rutland Formation overlies limestones of the Lincolnshire Limestone, sands, silts, clays and mudstones of the Grantham Formation (formerly the Lower Estuarine Series) and sandstones with subordinate limestones of the Northampton Sand Formation of the Jurassic Inferior Oolite Series. The sands, silts, clays and mudstones of the Grantham Formation are discontinuous locally and often the Lincolnshire Limestone is in direct contact with the Northampton Sand Formation.

66. Based on the results of a site investigation undertaken in and round the area of the western landfill area the Rutland Formation is between approximately 5.5 m and approximately 12 m thick beneath the Western Extension area of the site. In the vicinity of the site the Lincolnshire Limestone is between approximately 15 m and approximately 20.5 m thick.

67. A schematic cross-section of the Western Extension (Figure 9) shows a landfill cell in relation to the underlying geology, from (Augean, 2014).

Figure 9. Schematic cross-section for the Western Extension



2.6.2 Hydrology

68. The site is located in the catchment of the River Nene which flows generally eastwards, located approximately 6 km east south east of the site at the closest point.

69. Surface water management at the site is the subject of a surface water management plan which has been approved and is regulated by the Environment Agency as part of the Environmental Permit. The surface water management system is designed to drain to a ditch adjacent to the road at the south eastern corner of the site which flows generally to the south and after joining a small stream, outfalls to Willow Brook approximately 2.5 km south of the site. The Willow Brook joins the River Nene approximately 9 km south east of the site. The site is not located in an area of significant flood risk as designated by the Environment Agency and is not subject to flooding.

70. Based on information provided in 2010 by the Environment Agency, a third party information provider and the relevant Local Authorities there are no recorded surface water abstractions within a 3 km radius of the application area. There is an abstraction from the River Nene approximately 7 km east of the site where water is pumped to Rutland Water for public water supply. The abstraction is located approximately 8 km downstream of the confluence of the River Nene and the Willow Brook.

2.6.3 Hydrogeology

71. The boulder clay and the mudstones of the Rutland Formation have a low hydraulic conductivity and were not found to be water bearing during the drilling of boreholes at the site. The underlying limestones and sandstones of the Lincolnshire Limestone and the Northampton Sand Formation are water bearing. The Blisworth Limestone Formation at the site was not found to be water bearing during the drilling of boreholes at the site. It is likely that the Lincolnshire Limestone and Northampton Sand Formation are in hydraulic continuity. The Lincolnshire Limestone has a low to moderate primary hydraulic conductivity and a moderate to high secondary hydraulic conductivity due to the presence of fissures and fractures. Karst features such as swallow holes have been recorded in the vicinity of the site. A swallow hole has been observed in a field approximately 10 m to the north of the north western corner of the site. Groundwater levels have been recorded in the vicinity of the site between approximately 5 m below ground level at borehole K17 in the north west of the site and approximately 25 m below ground level at borehole K08 adjacent to and outside the south western corner of the site.

72. It is reported that the regional direction of groundwater flow in the Lincolnshire Limestone in the vicinity of the site is towards the east. Based on the groundwater elevation data for the area at and round the site, the direction of groundwater flow in the Lincolnshire Limestone local to the site is to the south and south east.

73. A number of springs are located within a 3 km radius of the site. The spring closest to the site is approximately 850 m south east of the site located approximately 400 m east of Westhay Lodge. Based on the general direction of groundwater flow in the vicinity of the site it is considered that the spring is down hydraulic gradient of the site. The spring to the south east of the site near Westhay Lodge feeds a tributary of Willow Brook.

74. The Rutland Formation is designated a Secondary B Aquifer. The Blisworth Limestone Formation and the Lincolnshire Limestone are designated as Principal Aquifers. The

Grantham Formation is designated a Secondary (undifferentiated) Aquifer and the Northampton Sand Formation is designated a Secondary A Aquifer.

75. One licensed groundwater abstraction which abstracts from two borehole locations, two deregulated groundwater abstractions and five private water supply groundwater abstractions, which are for agricultural, industrial and domestic use, are located within a 3 km radius of the site. The abstraction closest to and located down hydraulic gradient of the site with respect to the local groundwater flow direction is approximately 1.2 km south east of the site at Law's Lawn. This is a deregulated abstraction formerly licensed for general farming and domestic use. The abstraction at Law's Lawn is now registered with East Northamptonshire Council as a private water supply for domestic use.

2.7 Site Security

76. Site security is subject to control through the Environmental Permit. Actions have been agreed with the Environment Agency on the basis of risk. The entire operational landfill, reception area and site entrance will continue to be covered by 24 hour CCTV. The CCTV system includes night vision and motion sensing. The CCTVs will continue to be manned remotely. In the event of intrusion the police and site management will be called.

77. A review of the security of the site was undertaken by a Counter Terrorism Security Adviser (CTSA) from Northampton Police in 2011. The LLW accepted at the site is of such activity that it is highly unlikely to be the target of a terrorist attack due to the insignificant danger that the waste would pose to human health. LLW has no value so would not attract theft. As it is buried the material cannot be vandalised and trespassers would not be at risk due to the low activity of the waste and because it will be contained and covered. The CTSA has advised that: "to upgrade the fence to something more substantial around the whole perimeter would not be proportionate or commensurate to the perceived threat as it stands at this time".

78. Notwithstanding this comment in response to public concern on this issue and to reflect the terms of the current planning permission for the disposal of LLW in the current landfill area a 1.8 m high fence has been installed around the entire site boundary. An Emergency Plan is in place at the site which includes the actions which are necessary to inform the public in the highly unlikely event of an accident that has the potential for a significant effect beyond the site boundary. The Emergency Plan will be adapted and communicated as necessary depending on the operations permitted at the site.

79. The current site lighting comprises units fixed at a height of approximately 5 m directed towards the ground. The units operate on dusk to dawn optic sensors and all lighting is set up to minimise glare but to provide suitable light to ensure the effectiveness of the CCTV camera system. The lighting is located in key areas (see Figure 5) for both security and health and safety considerations and these locations are the site entrance and visitors' car park, the main site office to provide light to the staff car park and weighbridge area and around the laboratory and vehicle inspection area. Mobile lighting is provided on the landfill and down-facing lighting units are fixed to appropriate points on the soil treatment plant.

3 Waste Characteristics

3.1 Introduction

80. Hazardous waste that will be disposed at the site will be consistent with legislation and the Environmental Permit for the site. The waste types principally comprise treatment residues, contaminated materials including soils, and materials containing asbestos. Wastes that will not be accepted for disposal include liquid wastes, corrosive wastes, flammable wastes and wastes that are classified as oxidising. The non-radioactive hazardous wastes that are permitted for disposal are subject to a limit on their total organic carbon content and on the solubility of specified contaminants (subject to leaching tests).

81. LLW is waste that contains small amounts of radioactivity (up to 4000 Bq g^{-1} alpha activity and 12,000 Bq g^{-1} beta/gamma activity). NORM (naturally occurring radioactive material) waste contains radioactive substances that arise naturally in the environment and contain radionuclides of natural terrestrial and cosmic origin. NORM wastes generally fall into the LLW or very low level radioactive waste (VLLW) categories.

82. LLW typically comprises construction and demolition waste such as rubble, soils, crushed concrete, bricks and metals from the decommissioning of nuclear power plant buildings and infrastructure, lightly contaminated miscellaneous wastes from maintenance and monitoring at these facilities such as plastic, paper and metal, residues from plant at which LLW is incinerated and wastes from manufacturing activities, science and research facilities and hospitals where radioactive materials are used. NORM wastes are most commonly generated through processes that concentrate solid, liquid and gaseous NORM as a by-product (e.g. activities such as mining, the processing of minerals and earth materials, oil and gas operations, etc.). The physical, chemical and radiological characteristics of NORM wastes can vary markedly depending on the industrial process.

3.2 Radioactive Waste Inventory

83. The LLW that is and is expected to be disposed under the ENRMF Permit will arise from within the UK. The waste may arise from:

- **Non-nuclear industry sources** for example, waste derived from hospitals, universities, the oil industry or other non-nuclear users of radioactivity.

- **Nuclear industry sources** for example, wastes derived from decommissioning of nuclear power stations and research centres.

84. The LLW that is and is expected to be disposed at the ENRMF largely arises from the decommissioning and clean-up of nuclear industry sites and from the oil and gas industry.

85. The waste will conform to the landfill CFA which includes WAC established by any new permit and, where required, the consigning site will have an appropriate transfer permit. The radionuclides included in the current permit are listed in Table 1, along with their half-lives and daughters assumed to be in secular equilibrium (see paragraph Figure 10 below). The current permit includes an “Any other radionuclide” group to allow some flexibility for disposal of radionuclides that have not been listed explicitly. Ra-228 has been added to Table 1 as it is proposed to list it explicitly in the revised Permit.

86. When radionuclides decay they produce a daughter product that may be a stable atom, for example Po-210 has a half-life of 138 days and produces a stable daughter, Pb-206. In some cases the daughter product may also be radioactive and this can result in a sequence of radioactive daughters that is known as a decay chain. The uranium (U-238) and thorium (Th-232) series are the two most important decay chains. The longer lived radionuclides of these series are identified in Table 1 and Figure 11. The short-lived daughters are not treated explicitly in calculations of radiological impact although their hazard is assessed by including their doses with those of a longer lived parent.

87. In Table 1 and taking U-238 as an example, three daughters are listed (Th-234, Pa-234m, Pa-234) which do not appear in column 1 and any dose conversion factors used for U-238 are the sum of values for each of these radionuclides. The longest half-life of these three daughters is 24.1 days (Th-234). The last column indicates that there is a further daughter U-234, it has a long half-life of 245,500 years, but this is included in column 1 and will have its own dose conversion factors. The daughter of U-234 is Th-230 and because this also has a long half-life (75,380 years) it is considered explicitly in column 1. Dose conversion factors are taken from (ICRP, 1996), (European Commission, 1995), (European Commission, 1993) and (US EPA, 1993). Half-lives are taken from the LLWR radiological handbook (LLWR, 2011a) or from ICRP where radionuclides are not included in the LLWR assessment (ICRP, 1996).

Table 1 Radionuclides included in current permit

Radionuclide	Half-life (y)	Daughters assumed to be in secular equilibrium	Radioactive daughters considered explicitly
H-3	12.3		
C-14	$5.70\ 10^{3}$		
Cl-36	$3.01\ 10^{5}$		
Fe-55	2.74		
Co-60	5.27		
Ni-63	100.1		
Sr-90	28.8	Y-90	
Nb-94	$2.03\ 10^{4}$		
Tc-99	$2.11\ 10^{5}$		
Ru-106	1.02	Rh-106	
Ag-108m	418		
Sb-125	2.8		
Sn-126	$2.30\ 10^{5}$	Sb-126	
I-129	$1.57\ 10^{7}$		
Ba-133	10.5		
Cs-134	2.1		
Cs-137	30.2	Ba-137m	
Pm-147	2.6		
Eu-152	13.5		
Eu-154	8.6		
Eu-155	4.76		

Radionuclide	Half-life (y)	Daughters assumed to be in secular equilibrium	Radioactive daughters considered explicitly
Pb-210	22.2	Bi-210, Po-210	
Ra-226	$1.60\ 10^{3}$	Rn-222, Po-218, At-218, Pb-214, Bi-214, Po-214, Tl-210, Pb-210, Bi-210, Po-210	
Ra-228*	5.75	Ac-228	
Ac-227	21.8	Th-227, Fr-223, Ra-223, Rn-219, Po-215, Pb-211, Bi-211, Tl-207	
Th-229	$7.34\ 10^{3}$	Ra-225, Ac-225, Fr-221, Ra-221, Rn-217, At-217, Bi-213, Po-213, Tl-209, Pb-209	
Th-230	$7.54\ 10^{4}$		Ra-226
Th-232	$1.41\ 10^{10}$	Ra-228, Ac-228, Th-228, Ra-224, Rn-220, Po-216, Pb-212, Bi-212, Po-212, Tl-208	
Pa-231	$3.28\ 10^{4}$		Ac-227
U-232	68.9		
U-233	$1.59\ 10^{5}$		Th-229
U-234	$2.46\ 10^{5}$		Th-230
U-235	$7.04\ 10^{8}$	Th-231	Pa-231
U-236	$2.34\ 10^{7}$		Th-232
U-238	$4.47\ 10^{9}$	Th-234, Pa-234m, Pa-234	U-234
Np-237	$2.14\ 10^{6}$	Pa-233	U-233
Pu-238	87.7		U-234
Pu-239	$2.41\ 10^{4}$	U-235m	U-235
Pu-240	$6.56\ 10^{3}$		U-236
Pu-241	14.4		Am-241
Pu-242	$3.75\ 10^{5}$		U-238
Am-241	432		Np-237
Cm-243	29.1		Pu-239
Cm-244	18.1		Pu-240

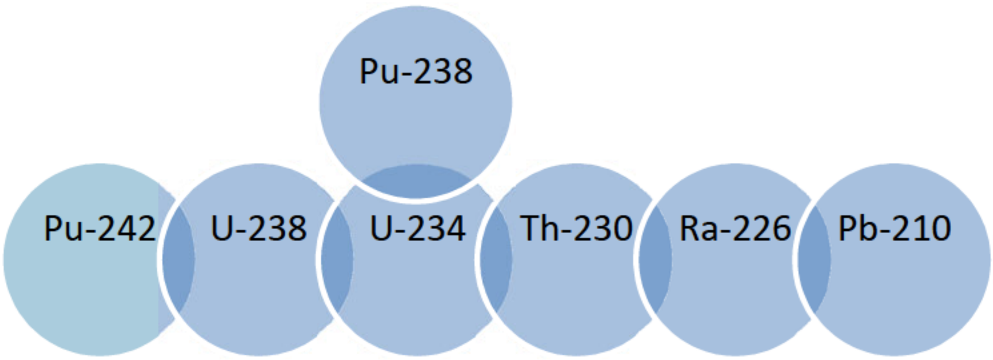
* Not considered in current Permit but proposed for revised Permit

88. Radionuclides with half-lives of less than one year or with half-lives significantly less than the parent radionuclide have not been explicitly assessed. Where such radionuclides arise from ingrowth, they are included through the assumption that they will be in secular equilibrium with the parent radionuclide, and the dose coefficients used are adjusted accordingly. The decay chains of coupled radionuclides are illustrated in Figure 11 through to Figure 13. Short-lived daughters that are assumed to be in secular equilibrium with a longer-lived parent radionuclide have been omitted from the figure. Note that Figure 11 lists Pb-210 as being considered explicitly, this applies only to the Goldsim groundwater

migration and radiological assessment models. In all other models Pb-210 is considered in secular equilibrium with the long-lived parent (Ra-226).

Figure 10. Decay system for Pu-241

Figure 11. Decay system for Pu-242 and Pu-238



89. Secular equilibrium describes the state that is achieved when each radionuclide in a chain decays at the same rate that it is produced. For example, as pure U-238 begins to decay to Th-234, the amount of thorium and its activity increase. Eventually the rate of thorium decay equals its production and its concentration then remains constant. As Th-234 decays to Pa-234m, the concentration of Pa-234m and its activity rise until its production and decay rates are equal. When the production and decay rates of each radionuclide in the decay chain are equal, the chain has reached secular equilibrium. Secular equilibrium between a long lived parent and a shorter lived daughter radionuclide is achieved after approximately five half-lives of the daughter radionuclide. Hence Ra-226 and Pb-210 would be in secular equilibrium after approximately 60 years.

Figure 12. Decay system for Cm-243

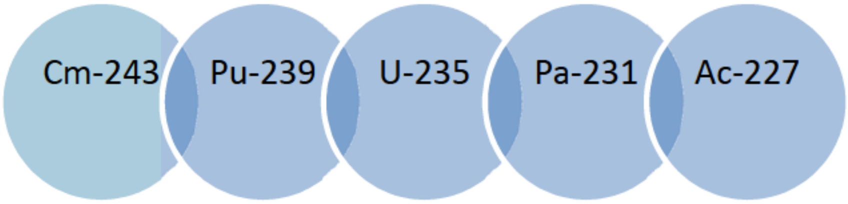


Figure 13. Decay system for Cm-244

90. In all of the assessment calculations, the quantities of long-lived daughters that have ingrown from specific parents or were directly disposed are distinguished. For example, the groundwater models consider four categories 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; and,
- U-234 ingrown from Pu-242.

91. The current inventory (Table 2) provides detail of the disposals at the site since the permit was granted in May 2011. The largest disposal as a fraction of the permit is Ra-226 at about 4% of the permitted amount, of which 95% is oil industry pipe scale that has been stabilised in cement.

Table 2 Radionuclides received at the ENRMF to June 2015

Radionuclide	Activity MBq	Permit CD8503 (MBq)
H-3	$2.38\ 10^{4}$	$3.23\ 10^{6}$
C-14	$2.24\ 10^{3}$	$1.70\ 10^{5}$
Cl-36	$3.10\ 10^{1}$	$8.50\ 10^{4}$
Fe-55	$4.58\ 10^{2}$	$5.95\ 10^{5}$
Co-60	$1.49\ 10^{3}$	$7.32\ 10^{5}$
Ni-63	$1.46\ 10^{3}$	$1.87\ 10^{5}$

Radionuclide	Activity MBq	Permit CD8503 (MBq)
Sr-90	$2.73\ 10^{3}$	$1.75\ 10^{6}$
Nb-94	$2.76\ 10^{-1}$	$8.50\ 10^{4}$
Tc-99	$1.71\ 10^{1}$	$3.74\ 10^{5}$
Ru-106	$1.80\ 10^{-2}$	$3.91\ 10^{5}$
Ag-108m	$3.75\ 10^{-1}$	$8.50\ 10^{4}$
Sb-125	1.15	$8.50\ 10^{4}$
Sn-126	0	$8.50\ 10^{4}$
I-129	1.80	$8.50\ 10^{4}$
Ba-133	$2.11\ 10^{1}$	$8.50\ 10^{4}$
Cs-134	2.23	$8.50\ 10^{4}$
Cs-137	$1.59\ 10^{4}$	$5.10\ 10^{6}$
Pm-147	5.08	$1.19\ 10^{5}$
Eu-152	$8.77\ 10^{2}$	$8.50\ 10^{4}$
Eu-154	$5.92\ 10^{1}$	$8.50\ 10^{4}$
Eu-155	9.04	$8.50\ 10^{4}$
Pb-210	$1.47\ 10^{4}$	$8.50\ 10^{4}$
Ra-226	$1.86\ 10^{4}$	$3.06\ 10^{5}$
Ra-228	Included in 'other radionuclide'	Included in 'other radionuclide'
Ac-227	4.43	$8.50\ 10^{4}$
Th-229	0	$8.50\ 10^{4}$
Th-230	$1.42\ 10^{2}$	$8.50\ 10^{4}$
Th-232	$3.98\ 10^{3}$	$8.50\ 10^{4}$
Pa-231	4.05	$8.50\ 10^{4}$
U-232	0	$8.50\ 10^{4}$
U-233	$2.70\ 10^{-2}$	$8.50\ 10^{4}$
U-234	$1.60\ 10^{2}$	$8.50\ 10^{4}$
U-235	6.58	$8.50\ 10^{4}$
U-236	$4.64\ 10^{-1}$	$8.50\ 10^{4}$
U-238	$3.18\ 10^{2}$	$8.50\ 10^{4}$
Np-237	0	$8.50\ 10^{4}$
Pu-238	$6.92\ 10^{1}$	$1.02\ 10^{5}$
Pu-239	$3.59\ 10^{2}$	$1.70\ 10^{5}$
Pu-240	$5.17\ 10^{2}$	$8.50\ 10^{4}$
Pu-241	$2.53\ 10^{3}$	$1.19\ 10^{6}$
Pu-242	$4.35\ 10^{-1}$	$8.50\ 10^{4}$
Am-241	$5.81\ 10^{2}$	$1.19\ 10^{5}$
Cm-243	1.51	$8.50\ 10^{4}$
Cm-244	$5.56\ 10^{1}$	$8.50\ 10^{4}$
Any other radionuclide	$8.49\ 10^{1}$	$8.50\ 10^{4}$

92. The future disposal inventory is not known in detail because waste streams for disposal will only be identified as a result of commercial agreements subsequent to receipt of the

revised permit. In view of this uncertainty estimates of radiological impact are given based on 'illustrative inventories' for waste streams that might be typical of those contributing to the total impact from disposals at the facility. These estimates are presented in Appendix G. In developing the safety case two illustrative inventories have been used, these are for wastes originally disposed to the Meashill trenches at Harwell (Augean, 2009a) and an illustrative NORM inventory based on the composition of a waste stream that has already been disposed at the ENRMF (consignment L12107400007, 2013).

93. These calculations do not show the total impact of the whole facility, this will be dependent on the waste that is actually received for disposal. However, the calculations illustrate the dose that would arise from waste streams typical of those that might be disposed to the ENRMF. The Meashill Trenches inventory has not been disposed of at the ENRMF.

Table 3 Inventories for calculation of contributions to the radiological impact

Radionuclide	Harwell Meashill trenches, 2010 (MBq)	ENRMF L12107400007, 2013 (Bq g^{-1})
H-3	3.25	
Co-60	8050	
Cs-137	952	
Ra-226	99.6	110.23
Th-232	40	16.30
U-234	500	
U-235	24	
U-238	500	
Pu-238	37	
Pu-239	400	
Pu-240	400	
Pu-241	38.2	
Am-241	99.2	

94. As stated above it is not possible prior to near the time of receipt of the wastes to describe the specific form, amounts or types of wastes. Most commonly waste from the nuclear industry is rubble, soils, crushed concrete, bricks and metals that arise from demolition of buildings that were previously used for nuclear research or power generation. A large programme of work to decommission the nuclear legacy sites created since the 1940's is currently underway in the UK that will generate significant volumes of LLW. The UK Nuclear Industry LLW strategy (Nuclear Decommissioning Authority, 2010) and supporting inventories (Nuclear Decommissioning Authority, 2013) provide detailed information on the potential types and nature of the wastes. During decommissioning, the hazards with the highest radioactivity are removed prior to demolition of structures. What remains after decommissioning is a mixture of construction materials/soils that can either be proven clean or which sometimes contain trace levels of radioactivity. Efforts are made to separate out radioactivity, to sort wastes, to recycle materials and to reuse materials. The wastes that remain with trace levels of radioactivity after these processes are typical of the wastes accepted at the ENRMF.

95. NORM waste contains radioactive substances that arise naturally in the environment and contain radionuclides of natural terrestrial and cosmic origin. NORM wastes are most commonly generated through processes that concentrate solid, liquid and gaseous NORM as a by-product (e.g. activities such as mining, the processing of minerals and earth materials, oil and gas operations, etc. see Table 4). The physical, chemical and radiological characteristics of NORM wastes can vary markedly depending on the industrial process. NORM wastes generally fall into the LLW or very low level radioactive waste (VLLW) categories. The UK strategy for the management of NORM was published recently (DECC, 2014) and included data on the types of waste, tonnage and activity concentrations produced. Those waste requiring specialist disposal are listed in Table 4.

Table 4 Types of solid NORM waste produced in the UK requiring specialist disposal

Industry	Waste type	Approximate quantity in tonnes per year	Approximate total activity per year
Oil and gas – offshore	Scales and sludge. May be hazardous due to heavy metal and hydrocarbon content	~ 160	~ 4 GBq Ra-226, ~ 2 GBq Ra-228, ~ 0.3 GBq Pb-210
Oil and gas – onshore	Scales and sludge, May be hazardous due to heavy metal and hydrocarbon content	< 20	< 0.05 GBq Ra-226, < 1 GBq Pb-210+
Titanium dioxide	Filter cloths	~ 10	~ 1 GBq Ra-226
China clay	Scale		
Zirconia industry	magnesium dross	~ 0.04	~232 MBq Th-232
Thorium coated lens manufacturer	Mixed solids	~ 1	~ 0.05 GBq Th-232
Contaminated land	Soil, building rubble, discrete items	Very variable	Very variable but anticipated to be less than 1 GBq Ra-226
Total		< 300 tonnes	< 6 GBq Ra-226, ~ 2 GBq Ra-228, ~ 1 GBq Pb-210, ~ 232 MBq Th-232

From (DECC, 2014)

96. Under the EPR (UK Statutory Instrument, 2010) a consignor can dispose of 5 10^{10} Bq y^{-1} of NORM waste containing up to 5 Bq g^{-1} to landfill without requiring a Permit (i.e.10,000 t y^{-1} of NORM at 5 Bq g^{-1}). There are also provisions for disposal of NORM waste containing up to 10 Bq g^{-1} at a landfill site without the need for a Permit, subject to the prior submission of an ESC to the EA and the receipt of no objections from EA.

97. The radioactive waste consignments received under the current permit during 2013 fall under the following broad groupings:

- Concrete, bricks, rubble, soil, sediments, metals (in various combinations);
- Concrete blocks;

- Plastics;
- NORM in drilling mud, sediments and descaling residues (from pipes and kilns);
- Hazardous waste (heavy metals, asbestos); and,
- Laboratory items, luminising material, clinker, incinerator filter cake, radiochemistry residues.

98. The general nature of the waste inventory is described in the national inventories for radioactive waste (Nuclear Decommissioning Authority, 2013). If the consigning site has established that disposal to landfill is the Best Available Technique (BAT) for the waste and it meets the CFA for the ENRMF, then the waste is acceptable. This would include wastes that if they were not radioactive would be classified as Inert, Non-Hazardous or Hazardous.

99. Subject to ensuring that the high levels of environmental protection afforded by the site are not compromised and the demonstration by the consignor that disposal to landfill is consistent with BAT, radioactive wastes with elevated levels of total organic carbon content and the specified soluble contaminants will be accepted at the site for disposal in accordance with the CFA

100. It is recognised that many disposed wastes are heterogeneous in terms of the distribution of activity within packaged material. For waste that remains in a waste cell the safety case can be based on the assumption that the wastes are broadly homogeneous. Where intrusion occurs the safety case needs to consider radionuclides that may be distributed heterogeneously in some waste materials. Consideration has therefore been given to the potential impact of variable activity within a waste package (see Section 6.6).

4 Authorisation of Disposal

4.1 Process by Agreement {R1}

101. The NS-GRA suggests that a developer is expected to enter into a voluntary agreement with the environment agencies to discuss a proposed facility and subsequent development (Requirement 1):

> *"The developer should follow a process by agreement for developing a disposal facility for solid radioactive waste." (NS–GRA (Environment Agencies, 2009) para 5.2.3)*

102. Early dialogue with the Environment Agency has been conducted at each stage of the development of the site. Discussions with the Environment Agency regarding the acceptance of LLW at the site date back to July 2008 and regular meetings have occurred.

103. The Environment Agency was consulted by Augean in respect of the landfill extension and the Environment Agency was also involved in the statutory process for the Nationally Significant Infrastructure Project application. The Agency took a direct role in late 2012 in the examination and hearings relating to the application.

104. Following the decision of the Secretary of State to grant the Development Consent Order in July 2013 Augean has engaged with the Agency in correspondence and at meetings to discuss the radiological proposals for the extension and to agree the approach to be taken by Augean for the Environmental Safety Case. Specifically meetings were held on the 11th November 2013 and the 10th June 2014 at which Augean set out the principles of their approach and the programme for the application.

4.2 Dialogue with Local Communities and Others {R2}

105. The NS-GRA expects the developer to engage widely in discussion of the developing ESC (Requirement 2):

> *"The developer should engage in dialogue with the planning authority, local community, other interested parties and the general public on its developing environmental safety case." (NS–GRA (Environment Agencies, 2009) para 5.7.1)*

106. Since 2009 Augean has conducted extensive dialogue with stakeholders including the planning authority and the local community. The consultations that have been conducted are summarised and listed in Appendix C**.**

107. The report by Jonathan Green on the ENRMF (The Planning Inspectorate, 2013) considered that the consultations that had been carried out covered all aspects of the proposed development including the disposal of LLW. The inspector concluded that the local community has had extensive engagement with Augean on this issue over several years, including public meetings, open days at the site, provision of written information, the opportunity to make written submissions and engagement with the public inquiry. The inspector was satisfied that the consultation requirements of the national policy for LLW management had been met.

108. Following the decision of the Secretary of State to grant the Development Consent Order Augean has continued to engage with the local community through the King's Cliffe Liaison

Group (KCLG) and the Thornhaugh Liaison Group (TLG). This has involved annual open days, a periodic newsletter and maintenance of a register of stakeholders. The KCLG has been kept up to date with the programme for the application to vary the radiological Environmental Permit and is aware that the application is scheduled for July 2015.

109. On submission of the application for the permit variation Augean will inform the local community representatives of the submission. Augean will also prepare a non-technical summary of the application proposals for circulation in the community. A site open day will be organised in October 2015 at which the community can discuss the application with Augean and the company's expert advisors. It is understood that the Environment Agency will take part in this event.

5 Management Requirements

5.1 Environmental Safety Case {R3}

110. This document has been designed to fulfil the requirement for an environmental safety case that is proportionate to the level of risk represented by the waste disposed at the ENRMF. The supporting technical basis for the radiological assessments used to support the ESC is presented in Appendix E. The safety assessments and related safety arguments presented throughout the document are drawn together in the summary (see Section 8).

5.2 Environmental Safety Culture and Management System {R4}

111. The NS-GRA outlines a requirement for a positive environmental safety culture supported by appropriate organisational structure and management systems (Requirement 4):

> *"The developer/operator of a disposal facility for solid radioactive waste should foster and nurture a positive environmental safety culture at all times and should have a management system, organisational structure and resources sufficient to provide the following functions: (a) planning and control of work; (b) the application of sound science and good engineering practice; (c) provision of information; (d) documentation and record-keeping; (e) quality management." NS–GRA (Environment Agencies, 2009) para 6.2.5*

112. Augean has an established effective management system and safety culture. The system ensures:

- Effective planning and control of work;
- Application of sound science and engineering practice;
- Safe acceptance and handling of waste;
- Maintenance and availability of comprehensive records and information; and,
- Quality management.

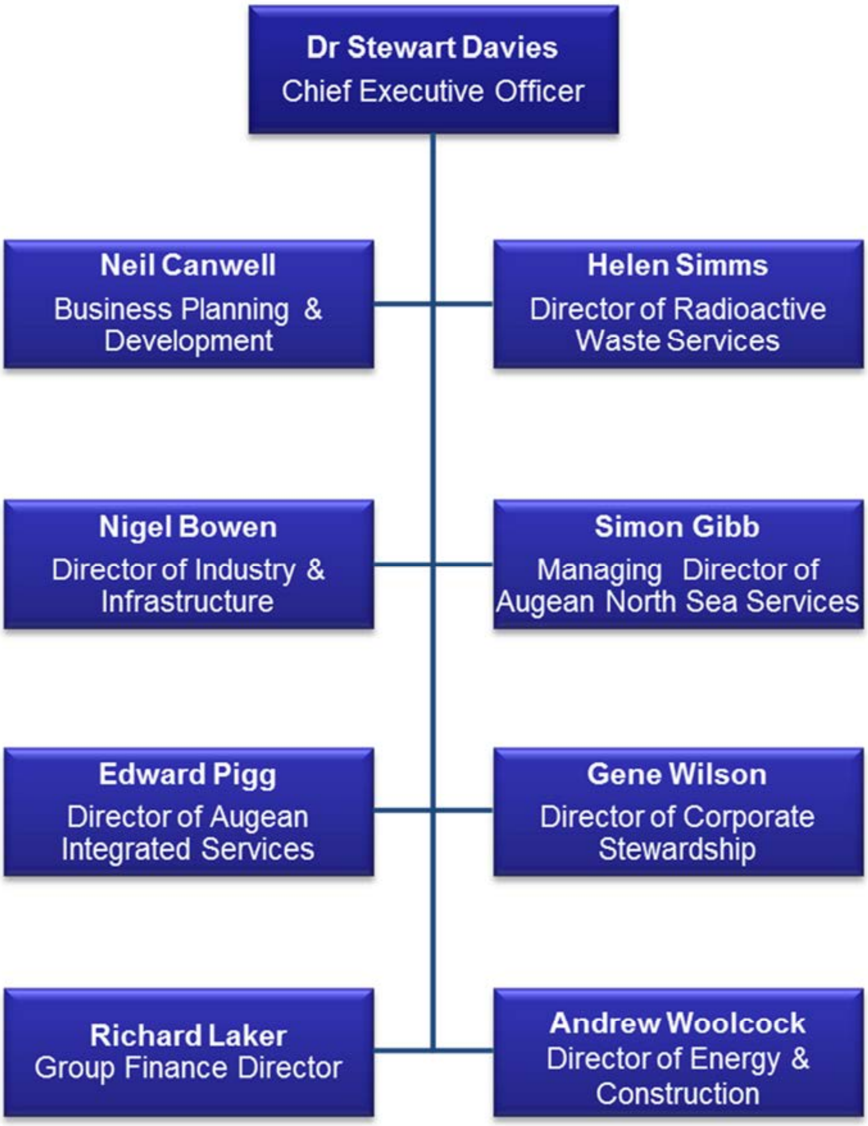
113. This system is subject to regular audit and inspection by internal independent compliance teams, external auditors including Public Health England (PHE), the British Standards Institute and customers, together with the Environment Agency. Augean has demonstrated that it is fully capable to assure environmental safety through its organisational structure, strong leadership and appropriate resourcing, competencies and culture. The proposed variation sets out a proposal that is a continuation of existing practice and does not require change to these systems. A summary of the business structure and management systems is provided below.

5.2.1 The Augean Business and Culture

114. Augean PLC, formed in 2004, is a UK-based specialist waste and resource management group. The group provides a wide range of services for difficult, hazardous and radioactive wastes through its treatment, transfer, landfill disposal and recycling operations. Over the past seven years the business has developed through a series of stages of acquisition, planning and development to establish a waste business operating to modern standards and responding to regulatory change.

115. The structure of the management board and areas of responsibility is shown in Figure 14.

Figure 14. Augean management board



116. Augean is committed to Corporate Social Responsibility (CSR) as demonstrated through the publication since 2006 annually of a CSR Report which measures their performance in respect of business, health and safety, their employees, their neighbours and the environment.

117. The Augean CSR Report is a record of company performance and how they are working together to improve that performance in respect of business values, health and safety, the environment and within our local communities. This annual exercise is a valuable discipline to help them demonstrate their commitment to responsible care, evaluate their performance against stated objectives and provide focus on their aspirations for the year ahead.

118. An essential element of their approach to business is their core business values supported by business principles.

"Augean's core business values are:

- *Respect – we show we value our people and others we work with;*
- *Integrity – we demonstrate we can be trusted;*
- *Teamwork – we work better together; and,*
- *Excellence – we strive to achieve our ambition.*

Based on these values Augean operate on the following business principles:

- *Priorities – we take action according to the priority: Safety, Compliance, Profit;*
- *Safety – we stop the job if we are not sure it is safe;*
- *Environmental responsibility – we respect the environment and take a planned approach to protecting it;*
- *Social and community responsibility – we invest time to build constructive relations with the communities in which we operate;*
- *Technical excellence – we value the expertise of our staff and use up-to-date techniques and equipment; and,*
- *Transparency – we are open and transparent in all that we do."*

5.2.2 Management systems

119. Operational performance is maintained through a certified Integrated Management System (IMS) delivering protection of health and safety, both internally and externally, and the management, protection and improvement of the environment for nature and our local communities. The IMS is certified by the British Standards Institute to the following standards:

- IS0 9001 Quality management system;
- ISO 14001 Environmental management system;
- OHSAS 18001 Health and safety management system; and,
- PAS 99 Integrated management system.

120. Central to the IMS is the Health, Safety, Quality and Environment Policy statement which is presented at Appendix D.

121. Delivery of the policy objectives is set out in the Augean Business Manual which:

- Defines roles of key positions in the organisation and provision of appropriate resources. This is further supported by specific job descriptions.
- Identifies the importance of training and competence which is supported by Corporate training requirements procedure and lead by the Group Training Manager.

- Identifies the provision of operational procedures.
- Describes the approach to incidents and accidents by the provision of site-specific emergency plans.
- Sets out the need for document control including record keeping.
- Describes auditing of compliance with the IMS which is supplemented by monthly compliance inspection at all sites.
- Includes systems for corrective and preventative action in the case of non-conformance.

122. The IMS provides a framework that considers the different aspects of the business and determines the impact of business activities on the workforce and the environment. Risk assessments have been conducted for all operational activities and where necessary to ensure adequate operational control procedures have been developed and implemented. Appendix D shows an overview of the IMS and lists the main corporate procedures within the system.

5.2.3 Corporate Reporting and Communication

123. The business has a range of mechanisms for developing policy, decision making and communication. Policy is usually determined at Management Board level. Policy decisions are communicated directly through the corporate structure and through a wide range of other mechanisms including Director Engagement Visits and presentations, training, safety campaigns and the monthly publication of Augean Update.

124. The outcome of auditing and inspection, near miss and safe act reporting, incident investigation and training are all reported to the Management Board on a monthly basis in a Compliance Report. The Compliance Report is reviewed each month at a Performance and Risk Board meeting. More strategic and policy matters together with serious near miss and incidents are reviewed at the Quarterly Compliance Review meeting attended by the Management Board, the Technical Team and invited Site Managers.

125. A series of operational fora operate within the business to develop and share best practice and to advise the Management Board on technical issues. These include:

- Best Practice Forum;
- Radiation Safety Group;
- Process Safety Group; and,
- Transport Managers Group.

5.2.4 Site organisation

126. The ENRMF Site Manager is responsible for the quality, health and safety and environmental performance of the sites. The Site Manager reports directly to the Management Board which is ultimately responsible for performance. The Site Manager at the ENRMF is a holder of a Certificate of Technical Competence for the management of a hazardous landfill. The Site Manager and Assistant Managers are trained Radiation Protection Supervisors (RPSs). The entire operating team has received radiation

awareness training and specific training in the operating procedures relevant to their function.

127. Operational meetings are held weekly. Health and safety meetings are held quarterly and include all staff present on site. There are Health and Safety Representatives in the landfill, treatment and administrative areas of the site.

128. Augean employs a range of highly qualified professionals with expertise in environmental and health and safety legislation, environmental management, chemistry, ecology, planning, engineering and waste management. As necessary, expertise is outsourced from external consultants. The Company maintains a list of approved consultants who are selected on the basis of qualification and experience and whose place on the list is dependent on good service.

129. Technical support and expertise is provided by the Technical Team specifically the Technical Manager who deals with Permitting issues and legislative compliance, the monitoring team that monitors the environmental impact of the site in all media and the site chemists who provide laboratory facilities and determine the suitability of waste for acceptance at the site. The Technical Team undertakes monthly inspections of the site including compliance with Environmental and Radiological Permits. Periodic audits of procedures are undertaken in accordance with the IMS the frequency of which is determined on a risk basis. The Technical Team reports all inspections to the Director of Corporate Stewardship who is a member of the Management Board and advises the Board on health and safety and environment issues. All members of the Technical Team have received radiological training relevant to the operation of the Augean sites and are qualified RPSs.

130. Augean employs a dedicated Technical Assessment Team providing a centralised service to the business. The team comprises three experienced professionals and one graduate trainee. The purpose of this team is to assess waste streams, determine how the waste can be managed in accordance with the waste hierarchy and the suitability of the waste for acceptance at a specified site. The team tracks and monitors waste inputs, including radiological capacity, to site through computer software. Specifically in respect of radioactive waste the company employs a qualified radioactive waste advisor and a specialist Technical Assessor qualified as an RPS who are further supported on a consultancy basis by Active Collection Bureau, Abbot Consulting Ltd and Loughborough University. The assessment team is independent of the operational team and based at the Company Headquarters at Wetherby. The Technical Assessor collates waste characterisation information and undertakes the initial chemical and radiological evaluation of the suitability of waste for disposal at the site. The final approval for booking of the waste to the site is given by the Site Manager. The process for acceptance of waste is set out in the Pre-acceptance and Acceptance procedures.

131. To support the site and in accordance with the Ionising Radiation Regulations and to provide staff training as necessary Augean will retain the services of PHE or other suitably qualified organisations as Radioactive Waste Advisor and Radiation Protection Advisor. The main scope of the support provided by the PHE is:

- Support during Permit transfer and variation;
- Preparing a comprehensive Radiation Risk Assessment of the impact on employees at the site;
- Local rules and procedures;

- Training site staff; and,
- Four site visits per annum to audit the waste handling operation, records and undertake additional monitoring.

5.2.5 Arrangements Specific to LLW Disposal Operations

132. The following arrangements are incorporated into the management system specific to LLW disposal operations:

- A radiation protection plan and risk assessment as required by the Ionising Radiations Regulations, prepared by the site Radiological Protection Advisor (currently PHE) (see Appendix H). Local rules in accordance with the Ionising Radiations Regulations and the conditions of the Environmental Permit. Defined roles and responsibilities include the following:
  - Radiation Protection Advisor,
  - Radioactive Waste Advisor (PHE),
  - Radiation Protection Supervisor(s), and,
  - Dangerous Goods Safety Advisor (Class 7).
- A procedure for the pre-acceptance of waste including the conditions for acceptance for LLW for use in contractual arrangements with consignors (LLW01, the CFA).
- A procedure for the pre-acceptance of waste by the central technical team (LLW02).
- A procedure for the receipt of waste, assay, waste emplacement, coverage, record keeping and general LLW disposal operations (LLW03).
- A procedure for the quarantine of non-compliant waste packages received at the ENRMF (LLW04).
- A procedure for monitoring employee doses and instructions for measuring X-Ray and Gamma Radiation dose rates during acceptance of LLW waste at the ENRMF (LLW05).
- A procedure for routine and periodic health surveillance monitoring for contamination and exposure.
- An emergency plan including response arrangements to identified fault scenarios including:
  i. Dropped load.
  ii. Contamination discovery.
  iii. Non-compliant load.
  iv. Dose above threshold discovery.
  v. Potentially contaminated person or wound.
- Procedures for environmental monitoring incorporated into the Monitoring and Action Plans (MAPs).
- A procedure for handling asbestos bearing packages.

- A procedure outlining actions to be taken if consignments are unable to reach the site entrance in order to minimise risks to staff, the site and wider community (LLW06).

5.2.6 Principles that would be applied to waste retrieval

133. Waste retrieval is not planned following emplacement and is not expected under all foreseeable circumstances. The Environment Agency has requested consideration of the principles that would be applied should a package of unsuitable waste be inadvertently deposited at the site.

134. Given the robustness of the packaging and the method of placement it is considered that the containers will remain intact in the landfill for an extended period. The placement of the waste in robust containers and in accurately located containers will facilitate recovery of waste if it is considered necessary. Detailed risk assessments would be undertaken and methods would be developed and agreed with the Environment Agency and the Radiation Protection Supervisor in advance of the exercise taking into account the specific circumstances of the removal but in principle the following approach would be taken:

- Identification of the location of the waste from the GPS records - this information also includes details of the types of hazardous waste deposited in the locality;
- Determination from GPS records the quantity and characteristics of waste that would need to be excavated to access the specific waste that must be removed;
- Identification of stockpiling areas for excavated material and standards for stocking;
- Consider the need for undertaking the operation under cover;
- Removal of the majority of soil and/or waste covering by machine and by hand where necessary;
- Monitor the emissions from the packaged waste to confirm that they remain less than 10μSv/hr at a distance of 1m from the package (i.e. measure to confirm before it is moved);
- In respect of bags locating of the carrying straps and then lifting out of the waste bag using the forks of a forklift truck;
- In respect of drums use of drum handler attachments on a forklift truck to remove the waste drum;
- If necessary the containers would be brushed down to remove extraneous adhered material;
- In the unlikely event that any of the containers are compromised they would be repacked or over packed at the excavation area;
- The containers would be loaded onto a lorry in the working area;
- Suitable personal protective equipment would be specified based on risk assessment and potential exposure would be monitored;
- Removal of the material from the site in accordance with the relevant Transportation Regulations; and,

- Replacement of wastes into the excavation using suitable cover material to infill interstices.

6 Radiological Requirements

135. The NS-GRA specifies dose constraints to members of the public that may arise from the ENRMF during the period of authorisation, a risk guidance level after the period of authorisation and dose constraints for human intrusion. This section summarises the dose assessments that have been undertaken to support the ESC (detailed in Appendix E). The results are presented as effective doses (µSv y^{-1} or mSv y^{-1}) and a maximum inventory (MBq) of each radionuclide.

136. The radiological capacity (also called the relevant value in this report) is the radionuclide inventory of each radionuclide that can be disposed at the ENRMF that would not result in a dose greater than the relevant dose criterion from any of the exposure scenarios. It is therefore the minimum of the values calculated for each exposure scenario (see Appendix E). All calculations detailed in Appendix E are inherently cautious ensuring that the prospective dose is overestimated and, because the radiological capacity is inversely proportional to the dose, the radiological capacity is therefore minimised. The radiological capacity of the ENRMF for each radionuclide is presented in Section 7.4.2 and these values, together with the sum of fractions approach, are used to control disposals. Calculating the fraction of the radiological capacity that has been used by each disposed radionuclide in turn and ensuring that the sum of fractions is ≤1.0 will ensure that the dose from all disposed radionuclides does not exceed the relevant dose criterion. Hence, the sum of fractions approach ensures that the dose criteria are not exceeded if a mix of radionuclides is disposed of. The ‘relevant values’ presented in Table 26 (Schedule 3 of the proposed Permit) are these radiological capacity values based on the dose criteria.

137. The site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013) restricts LLW disposal at the ENRMF to 448,000 t at a maximum specific activity of 200 Bq g^{-1}. This constrains disposal of LLW at the ENRMF to a maximum total of 89.6 TBq (8.96 10^{7} MBq). The maximum inventory that could be disposed of in the site for each radionuclide is therefore the minimum of 89.6 TBq and the radiological capacity and is therefore not necessarily the same as the radiological capacity. The results of the dose assessments presented in Sections 6.1, 6.2 and 6.3 show the maximum inventory that could be disposed of each radionuclide based on these two constraints. The maximum inventory values are not appropriate for use as ‘relevant values’ for the proposed Permit as they would overestimate the fraction of the radiological capacity for radionuclides with radiological capacities above 89.6 TBq.

138. 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 G.

6.1 Dose constraints during the period of authorisation {R5}

139. The NS-GRA specifies dose constraints for members of the public for the period of authorisation (Requirement 5):

“During the period of authorisation of a disposal facility for solid radioactive waste, the effective dose from the facility to a representative member of the critical group should not exceed a source-related dose constraint and a site related dose constraint.

The UK Government and Devolved Administrations have directed the environment agencies to have regard to the following maximum doses to individuals which may result from a defined source, for use at the planning stage in radiation protection:

- *0.3 mSv/y from any source from which radioactive discharges are made; or,*
- *0.5 mSv/y from the discharges from any single site."*

(Environment Agencies, 2009), para 6.3.1 and 6.3.2

140. For the purpose of the assessments reported here the ENRMF is considered to be a source from which radioactive discharges occur.

141. PHE recommends a lower annual dose constraint for members of the public of 0.15 mSv (milli Sievert) for a new disposal facility (HPA, 2009). The ENRMF is an existing disposal facility and therefore this constraint does not apply.

142. In supplementary guidance issued by the Environment Agency (Environment Agency, 2012b) for the implementation of the Groundwater Directive it is an additional requirement that:

"The radiation dose to members of the public through the groundwater pathway during the period of authorisation of the facility is consistent with, or lower than a dose guidance level of 20 µSv y^{-1}."

143. A dose guidance level of 20 µSv y^{-1} (micro Sievert per year; 0.02 mSv y^{-1}) is therefore applied in this ESC for public exposure through the groundwater pathway during the period of authorisation.

144. For workers the legal dose limit is 20 mSv/year, and the criterion used for the safety case is 1 mSv y^{-1}, which is the same as the current legal limit for the public. This is an operational criterion and is not used to set the radiological capacity of the landfill because the exposure arises in a manner unrelated to the total capacity of the site. This criterion does affect some of the authorisation conditions, in particular external dose limits on packages. This criterion will be used for radiation protection purposes during operation of the facility.

6.1.1 Dose assessments for the period of authorisation

145. Doses and risks need to be assessed for a range of hypothetical exposure groups in order to identify those at greatest risks at a given time. The present-day landuse can be used to inform calculations of the impact during the period of authorisation. Throughout this report the term "scenario" is used to describe a situation or class of situations leading to future exposures.

146. The radiological assessment has considered a range of potential scenarios. A review of generic guidance and existing publicly available ESCs identified a set of scenarios that are discussed in detail in Appendix E and those considered for the ENRMF for the period of authorisation are summarised in Table 5. In cases where a scenario has not been assessed, because it will not or is very unlikely to occur at the ENRMF, the reasons for this are discussed. The scenarios discussed below consider both workers and members of the public during the period of authorisation and these are divided into two broad categories – those that are expected to occur and those which have a low likelihood of occurrence.

None of these scenarios constrain the amount of radioactivity that can be disposed of at the ENRMF since this is constrained by calculations relating to the period after authorisation.

Table 5 Summary of radiological assessment scenarios; during in the period of authorisation

Scenario	Exposed group
Period of Authorisation – expected to occur	
Direct exposure	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 – not certain to occur	
Leachate spillage	Farming family
Dropped load	Worker
Aircraft impact	Member of public
Barrier failure	Member of public
Wound exposure	Worker
Exposure due to fire	Member of public

147. The detailed results of the assessments for the period of authorisation are presented in Appendix E, Section E.3. The ESC uses the term "period of authorisation" to cover the time when active management controls are maintained and the Permit remains in force. This period is assumed to last until 2086 in these assessments. Post-closure or after the period of authorisation refers to the time when the permit has been revoked and there is no active management or control at the site (2086 onwards is assumed in these assessments although the period of authorisation may be much longer).

148. The radiological capacity for individual radionuclides present in the LLW is obtained from the results of the environmental safety case and depends on the radiological characteristics of the radionuclide. Both the radiological capacity and the maximum inventory are calculated on the basis that the LLW only contains this one radionuclide. Actual waste disposal will be controlled using a sum of fractions approach (see paragraph 308).

6.1.2 Direct exposure from waste handling and emplacement

6.1.2.1 Waste handling

149. Radiation risks to employees from normal operations were reviewed by the HPA [Annex C, (Augean, 2009a)], and the assessment is included here as Appendix F. A conservative estimate of the dose to workers as a result of three work activities suggests an annual dose of about 1.1 mSv if the same worker undertook waste receipt, monitoring, transfer and placement in the landfill and worked in the covered waste area. HPA considered it unlikely that the same person would be exposed during all the listed work activities. An assessment

of exposure resulting from a wound concluded that internal doses from a contaminated wound would be very unlikely to exceed 1 mSv in practice.

150. The external radiation exposure to workers from their occupancy near to a waste package prior to disposal was also assessed by the UKAEA [Annex D of (Augean, 2009a)] reproduced here as Appendix I.

151. Appendix I considers the external radiation dose for a series of cases and package types. The hypothetical worst case is identified to be a flexible type waste container with 200 Bq g^{-1} of Co-60. This is an unlikely case and another case is included in Appendix I to illustrate more typical exposures. The hypothetical worst case dose identified in Appendix I is 14.5 µSv h^{-1} measured at a distance of 1 m from the package face. However, the radiation protection advisor (Appendix F) has advised that the maximum dose at 1 m from a package should be less than 10 µSv h^{-1} in order to ensure the occupational dose is considerably less than the dose criterion of 1 mSv y^{-1}. Thus 10 µSv h^{-1} is used as an acceptance criterion and constrains the contents of the package to this limit.

152. The adopted CFA is that the dose at 1 m from the package face must be less than 10 µSv h^{-1}. This is measured by the consignor prior to sending the package and is checked upon arrival of the package at the ENRMF. This dose is specific to workers during the operational phase and is managed through occupational radiation dose protection practices, hence it is not used to constrain overall radiological capacity.

153. Assessments have been presented (Augean, 2009a), showing the dose to a member of the public standing at a distance in direct line of sight of a waste package/shipment. The maximum dose rate at 50 metres is estimated to be 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}; the corresponding dose at a distance of 100 m would be 3 µSv y^{-1}. These are low doses and the calculations are very conservative. The estimates of dose do not take into account the significant shielding afforded by the soil screen bund at the boundary of the site.

6.1.2.2 Waste Emplacement and cell excavation

154. The external radiation exposure of workers in the vicinity of the waste emplaced in the landfill after it has been covered is assessed by the UKAEA [Annex H of (Augean, 2009a)] reproduced here as Appendix J. This illustrates the dose rate for varying cover thicknesses using two illustrative cases, one of which is a worst case. The advice of the radiation protection advisor (Appendix F) is that the maximum radiation dose 1 m above the covered waste should be less than 2 µSv h^{-1} in order to ensure the occupational dose is considerably less than the dose criterion of 1 mSv y^{-1}.

155. The assessment demonstrates that for most cases a 0.3 m thick cover layer will more than achieve the specified dose rate. For the worst case of waste containing Co-60, at 200 Bq g^{-1}, a cover layer of 0.7 m is required to reduce the dose rate.

156. The existing Permit contains the condition that a minimum cover layer of 0.3 m be utilised and that if the dose rate 1 m above the waste is still greater than 2 µSv h^{-1} then further cover will be added in order to achieve the dose rate. The minimum cover layer of 0.3 m is adequate to ensure daily physical protection of the waste. This condition is specified in the site operating procedure and it is proposed that this condition is retained.

157. Additional ALARA precautions are that all wastes are handled by machines. The only people on foot are those unstrapping loads and undertaking health physics monitoring. Workplace monitoring will confirm actual doses and enable dose limitation to be managed. Workplace monitoring to date has shown no measurable doses.

158. Cell excavations have not been assessed in the ESC. Any excavations will be undertaken with full knowledge of where waste is placed within each cell and appropriate precautions will be taken. Installation of the landfill cap requires landfill workers to locate the side liner of a waste cell. Operating procedures at the ENRMF require at least 2 m of non-radioactive waste to be placed between the side liner and LLW to make certain that workers do not come into contact with LLW packages when the landfill is permanently capped.

159. The external dose to workers during the operational phase will 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.

6.1.3 Impact due to leachate treatment

160. The permit variation application involves no specific authorised liquid discharge routes. Leachate is currently used at the on-site soil treatment facility or sometimes treated off-site.

161. Under normal circumstances leachate generated in the landfill is treated on site through the stabilisation plant. This process binds the leachate in the stabilisation matrix. The stabilised material is then disposed of in the landfill. In the event that the capacity of the stabilisation plant is insufficient to accommodate the amount of leachate that must be removed from the landfill the excess leachate is sent to a suitable treatment works which currently is the Augean Avonmouth Treatment Works. Under normal operating circumstances it is necessary to send leachate to the treatment works approximately once per month although this may be more frequent in the winter depending on the amount of precipitation.

162. Use of leachate at the on-site soil treatment facility is covered by the local assessment for the treatment facility, for compliance with the IRR, and is therefore not addressed in the ESC. However, an assessment has been undertaken to determine the potential impact of off-site leachate management. The ESC therefore considers the treatment of contaminated leachate at an off-site hazardous waste water treatment facility, secondary treatment at a sewage treatment works followed by discharge to an estuary and assesses the impact on workers at the treatment facilities, anglers fishing in an estuary into which the sewage treatment works discharge and a farming family assumed to grow crops on land fertilised with sludge from the works. Output from the GoldSim 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 within waste materials whereas in reality the waste received at the ENRMF is likely to provide sorption sites within waste cells

163. The main radionuclide specific doses arising from disposing of the maximum inventory (448,000 t at 200 Bq g^{-1} or the radionuclide radiological capacity if the calculated radiological capacity is lower than 89.6 TBq) are shown in Table 6. For the majority of radionuclides the highest dose is to the treatment facility worker. Doses to fishermen are much lower than doses to other exposed groups and are not significant. The highest dose is to workers at the off-site treatment facility from Co-60 (86 µSv y^{-1}), and this dose would only occur if Co-60 was the only radionuclide disposed of at the ENRMF and 89.6 TBq was

disposed of. As a percentage of the national inventory Co-60 accounts for about 7% of low level waste (Nuclear Decommissioning Authority, 2013) and it comprised less than 2% of the activity in radioactive waste disposed to June 2015 at the ENRMF. Hence it is expected that the dose to off-site treatment workers from disposed Co-60 at the ENRMF would be significantly lower (e.g. by an order of magnitude or more) if it received wastes with radionuclide compositions similar to that of the national inventory. The second highest dose is from Sb-125 to workers at the off-site treatment facility (14.9 µSv y^{-1}). As a percentage of the national inventory Sb-125 accounts for less than 0.1% of low level waste (Nuclear Decommissioning Authority, 2013), and about 1.1 MBq has been disposed in the ENRMF to June 2015. Hence it is expected that the dose to off-site treatment workers from disposed Sb-125 at the ENRMF would be significantly lower than that given in Table 6. Results for the top 10 radionuclides are given below and a full set of assessed doses is in Appendix E, Section E.3.5.3.

164. The calculations of doses from leachate treatment are very conservative and it is expected that the dose from off-site treatment of leachate would not exceed 0.3 mSv y^{-1} for Co-60 since the activity in the ENRMF is controlled using the sum of fractions approach: Co-60 comprises less than 7% of the activity in LLW and all other radionuclides give much lower doses.

Table 6 Dose estimated for exposure from the off-site treatment of leachate

Radionuclide	Maximum inventory (MBq)	Dose from disposal of maximum inventory		
		Treatment facility worker (µSv y^{-1})	Farming family - adult (µSv y^{-1})	Fisherman - adult (µSv y^{-1})
Co-60*	$8.96\ 10^{7}$	$8.60\ 10^{1}$	$4.80\ 10^{-1}$	$4.76\ 10^{-7}$
Sb-125*	$8.96\ 10^{7}$	$1.49\ 10^{1}$	$6.77\ 10^{-2}$	$5.83\ 10^{-9}$
Ra-226	$8.96\ 10^{7}$	5.56	$7.70\ 10^{-2}$	$3.14\ 10^{-6}$
Eu-152	$8.96\ 10^{7}$	6.54	$2.76\ 10^{-2}$	$2.57\ 10^{-7}$
Eu-154	$8.96\ 10^{7}$	6.86	$2.74\ 10^{-2}$	$2.27\ 10^{-7}$
Pb-210	$8.96\ 10^{7}$	$1.90\ 10^{-1}$	$5.36\ 10^{-2}$	$3.60\ 10^{-6}$
Cm-243	$8.96\ 10^{7}$	$7.10\ 10^{-1}$	$1.04\ 10^{-2}$	$4.19\ 10^{-9}$
Cs-137	$8.96\ 10^{7}$	1.77	$5.97\ 10^{-3}$	$2.18\ 10^{-8}$
Cs-134	$8.96\ 10^{7}$	3.55	$4.91\ 10^{-3}$	$1.35\ 10^{-8}$
Sr-90	$8.96\ 10^{7}$	$2.02\ 10^{-1}$	$8.57\ 10^{-3}$	$1.40\ 10^{-8}$
Tc-99	$8.96\ 10^{7}$	$5.37\ 10^{-2}$	1.72	$1.26\ 10^{-6}$

* These comprise less than 7% of the national LLW inventory so the doses are significantly overestimated, see text

165. The main contributors to dose from leachate treatment are likely to be Co-60, Cs-137, Sr-90 and Ra-226 when both the current inventory at the ENRMF and the composition of the national LLW inventory are considered. The projected dose to workers using the radionuclide proportions in the national waste inventory is 6.4 µSv y^{-1}, the dose based on the proportions currently disposed is 3.0 µSv y^{-1}, and the dose based on actual disposals is $3.0\ 10^{-3}$ µSv y^{-1}. Leachate monitoring at the ENRMF for these radionuclides would provide

an early indication as to whether the assessment is robust. The list should be reviewed as the inventory accumulates.

166. The workers at the off-site treatment facility would not be exposed as a result of undeclared radioactivity in the leachate sent for treatment. Radionuclide activity in leachate is monitored on a regular basis. Discharges from the ENRMF will be subject to permitting.

6.1.4 Impact due to atmospheric releases

167. The permit variation application involves no specific permitted gaseous discharge routes. However, the inadvertent release of gases during operations may expose landfill workers on the site and public exposure to gas may also occur but at some distance from the source (Appendix E, Section E.3.3). The gas pathway considers radioactive carbon, tritium and radon. The aim is to restrict chemical and biological processes occurring within the ENRMF once disposal has taken place. For example there are limits on the total organics in waste to reduce the prospect of C-14 and H-3 releases, no waste is accepted in liquid form, waste must not be corrosive, oxidising or flammable, it should not contain ion exchange resins or complexing agents and hazardous waste leaching criteria apply to non-radioactive content of LLW where practicable. These conditions reduce the likelihood that rapid gaseous release will occur and hence the assumptions used in the calculations are very conservative.

168. The calculations assume that waste is covered on a daily basis to a depth of 0.3 m, and covered again within 2 months, there is no radioactive decay and members of the public are always present in the downwind direction resulting in the highest dose (Appendix E, Section E.3.3.1). Similar assumptions are used for workers but they are assumed to be at the point of discharge with dilution by the average wind speed. The carbon-based gas release rates were calculated using a model of landfill gas evolution (GasSim) and doses are based on the peak rate of gas production following disposal of the inventory. The doses in Table 7 are from disposals of the maximum inventory that could be disposed of in the site, i.e. the minimum of 448,000 t at 200 Bq g^{-1} (89.6 TBq) and the radiological capacity.

Table 7 Dose estimated for exposure from gas released during operations

Radionuclide	Maximum inventory (MBq)	Dose* (μSv y^{-1})	
		Worker	Public
H-3	$8.96\ 10^{7}$	1.23	$4.55\ 10^{-1}$
C-14	$8.96\ 10^{7}$	$4.21\ 10^{1}$	$1.56\ 10^{1}$
Ra-226**	$8.96\ 10^{7}$	$3.02\ 10^{2}$	$1.87\ 10^{1}$

*Based on the peak release rate following disposal of the maximum inventory given in Column 1.

** Dose arises from radon gas.

169. Doses from exposure to gas when each radionuclide is disposed at the maximum inventory (see Table 7) are significantly below the site criterion for workers (1 mSv y^{-1}) and the public dose constraint (0.3 mSv y^{-1}). The dose estimates indicate that the highest doses are from Rn-222 exposure (following Ra-226 disposal) for both a worker and a member of the public. These dose estimates would still be below the relevant criteria even if 89.6 TBq of each radionuclide were disposed of. The projected peak dose to the public using the radionuclide proportions in the national waste inventory is 0.2 μSv y^{-1}, the dose based on the proportions currently disposed is 4.3 μSv y^{-1}, and the dose based on actual disposals is $4\ 10^{-3}$ μSv y^{-1}.

6.1.5 Impact due to leachate migration in groundwater during Period of Authorisation

170. Water abstraction from a hypothetical well located at the site boundary was modelled using GoldSim (see Appendix E Section E.3.4 and Appendix F) and doses were calculated to members of the public drinking contaminated well water and using the well water for irrigation. There is currently no well located at the site boundary but this scenario was chosen as the most conservative case. The calculated water activity concentrations at the site boundary are higher than those calculated for the existing abstraction point located about 1200 m from the ENRMF. The peak activity concentrations in the groundwater, at the location of the site boundary, during the period to the end of active management were used for the assessment (see Appendix E, Section E.3.4.5). These therefore correspond to the maximum dose that would be received in that time period. The radionuclide specific doses arising from disposing of the maximum inventory (448,000 t at 200 Bq g^{-1}, or the radiological capacity if the calculated radiological capacity is lower than 89.6 TBq), show maximum doses of about 0.4 µSv y^{-1} for Pb-210, 0.2 µSv y^{-1} for Ac-227 and 0.1 µSv y^{-1} for Cl-36; all other radionuclides give rise to lower maximum doses. The doses from this pathway during the period of authorisation are therefore all low and do not constrain landfill capacity.

171. The main contributors to dose are likely to be H-3, Cl-36, Sr-90, I-129 and Pb-210 when both the current inventory at the ENRMF and the composition of the national LLW inventory are considered. Groundwater monitoring for these radionuclides and comparison against background levels in groundwater (e.g. levels in groundwater extracted up-stream of the ENRMF) would provide an indication of releases into the environment through this pathway. The list should be reviewed as the inventory accumulates. Monitoring reports are prepared annually and published (http://www.augeanplc.com/Radiological). The post LLW radiological monitoring data shows that all analytical results were almost identical to the background data, with the majority of results showing that levels were below or equal to the Limit of Detection (LOD) of the test method used for the parameter listed at the time of analysis.

6.1.6 Doses from uncertain events during the period of authorisation

172. A number of events that are unlikely to occur during the period of authorisation have been considered (Table 5). Assessments have been undertaken for dropped waste containers, a leachate spillage during transport to the leachate treatment facility and an aircraft crash at the site. A fire in a waste cell and total barrier failure were considered too unlikely to warrant an explicit assessment (see discussion in Appendix E.3). The gradual deterioration of the HDPE liner is expected to occur and is considered in the groundwater risk assessments. Wound exposure is addressed in the operational safety case (see Section 6.1.2).

173. The maximum doses arising from a dropped container, an aircraft impact and leachate spillage are given in Table 8. In the first two cases the doses depend on the specific activity of waste (assumed to be 200 Bq g^{-1}) and for the leachate spillage the doses depend on the activity concentration in the leachate: this is based on the disposal of the maximum inventory (89.6 TBq or the calculated radiological capacity if it is lower). In the case of an aircraft impact 300 m^3 of waste are assumed to be displaced and the dose to a member of the public and a worker is assumed to be the same in the early stages of the response to the accident.

Table 8 Doses from a dropped container, aircraft impact and leachate spillage

Radionuclide	Dose due to dropped load*		Dose due to aircraft impact* (µSv)	Leachate spillage	
	Worker (µSv)	Public (µSv)		Dose to farming** family (µSv)	Maximum inventory (MBq)
Sr-90	$5.35\ 10^{-1}$	$1.82\ 10^{-3}$	$7.22\ 10^{-1}$	$1.11\ 10^{1}$	$8.96\ 10^{7}$
Tc-99	$4.29\ 10^{-2}$	$1.46\ 10^{-4}$	$5.79\ 10^{-2}$	8.54	$8.96\ 10^{7}$
Cs-134	$6.60\ 10^{-2}$	$2.24\ 10^{-4}$	$8.91\ 10^{-2}$	4.03	$8.96\ 10^{7}$
Pb-210	$3.30\ 10^{1}$	$1.12\ 10^{-1}$	$4.45\ 10^{1}$	$9.48\ 10^{1}$	$8.96\ 10^{7}$
Ra-226	$6.44\ 10^{1}$	$2.19\ 10^{-1}$	$8.69\ 10^{1}$	$1.96\ 10^{1}$	$8.96\ 10^{7}$
Ra-228	$1.97\ 10^{2}$	$6.69\ 10^{-1}$	$2.66\ 10^{2}$	$3.04\ 10^{1}$	$8.96\ 10^{7}$
Ac-227***	$1.88\ 10^{3}$	6.38	$2.53\ 10^{3}$	$3.04\ 10^{1}$	$8.96\ 10^{7}$
Th-229	$8.45\ 10^{2}$	2.87	$1.14\ 10^{3}$	$7.46\ 10^{-1}$	$8.96\ 10^{7}$
Th-230	$3.30\ 10^{2}$	1.12	$4.46\ 10^{2}$	$1.98\ 10^{-1}$	$6.93\ 10^{7}$
Th-232	$5.61\ 10^{2}$	1.91	$7.57\ 10^{2}$	1.03	$7.16\ 10^{7}$
Pa-231	$4.62\ 10^{2}$	1.57	$6.24\ 10^{2}$	$7.86\ 10^{-1}$	$1.86\ 10^{7}$
U-232	$1.22\ 10^{2}$	$4.15\ 10^{-1}$	$1.65\ 10^{2}$	$2.85\ 10^{1}$	$8.96\ 10^{7}$
U-236	$2.87\ 10^{1}$	$9.76\ 10^{-2}$	$3.88\ 10^{1}$	4.09	$8.96\ 10^{7}$
Pu-238	$3.63\ 10^{2}$	1.23	$4.90\ 10^{2}$	1.17	$8.96\ 10^{7}$
Pu-239	$3.96\ 10^{2}$	1.35	$5.35\ 10^{2}$	1.28	$8.96\ 10^{7}$
Pu-240	$3.96\ 10^{2}$	1.35	$5.35\ 10^{2}$	1.28	$8.96\ 10^{7}$
Pu-242	$3.63\ 10^{2}$	1.23	$4.90\ 10^{2}$	1.23	$8.96\ 10^{7}$

* Based on 200 Bq g^{-1}

** Based on the maximum leachate activity concentration during the period of authorisation that corresponds to disposal of the maximum inventory

*** National LLW inventory of Ac-227 is only 13 MBq, see text

Dropped load

174. The dropped load dose assessment meets the site criterion for workers for all radionuclides except Ac-227; all doses to the public are below 20 µSv. The national LLW inventory reports a total of 13 MBq of Ac-227 (Nuclear Decommissioning Authority, 2013), which is less than the total used in this assessment (200 MBq). Hence, Ac-227 is very unlikely to be present at 200 Bq g^{-1} in a single package given the low occurrence of this radionuclide and the maximum dose from a dropped load would be at least a factor of 10 smaller than that given in the table. The 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.

175. A key measure to mitigate dropped load dispersion events is to use waste containers that 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 and these regulations specify isotope specific limits designed to ensure public safety.

176. 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 the ENRMF.

Aircraft impact

177. The largest calculated dose following an aircraft impact on the site (approximately 3 mSv) arises from inhalation of dust containing Ac-227; inhalation of Th-229 gives about 1 mSv, the remaining alpha emitters about 0.5 mSv or less, and the beta and gamma emitters give much lower doses. As stated above there is very little Ac-227 reported in the national inventory of LLW (Nuclear Decommissioning Authority, 2013) so the inhalation dose from disposed Ac-227 at the site would be expected to be much lower than 3 mSv.

178. The assessment has not taken into account the depth of daily cover, has used a high resuspension factor and assumed that a large proportion of a waste package is very powdery. This calculation is therefore conservative and the complexity of an aircraft crash means that this calculation can only be considered as a scoping calculation. Nevertheless, the scoping calculations indicate that the 3 to 20 mSv y^{-1} dose guidance level for human intrusion events would not be exceeded by this very low probability event. This scenario has not been used to constrain the radiological capacity because it has a very low probability of occurrence and is independent of the total tonnage and total activity in the waste cells at the ENRMF.

Leachate spillage

179. It is expected that a spillage of landfill leachate will be subject to mitigation measures based on a detailed assessment of any ground contamination at the site. Doses to site workers would be kept within site constraints. However, leachate that enters water resources would become diluted and effective mitigation measures would be more difficult to achieve. The assessment of leachate spillage therefore focusses on pathways related to the use of water resources (drinking, irrigation, livestock and angling). The leachate activity concentration used in the calculations is the maximum observed during the period of authorisation based on output from the GoldSim model.

180. The radionuclide specific doses arising from disposing of 89.6 TBq or the radiological capacity if the calculated capacity is lower are presented in Table 8. The highest doses arise from spillage of leachate containing Pb-210, with a dose of about 95 µSv. The event has a low probability of occurring and clean-up actions would be taken to mitigate the event. The scenario does not constrain the radiological capacity even without mitigation measures.

6.2 Risk guidance level after the period of authorisation {R6}

181. The NS-GRA provides guidance on the level of risk to be applied after the period of authorisation (Requirement 6):

> *"After the period of authorisation, the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year)." (Environment Agencies, 2009), para 6.3.10*

182. Based on the recommended risk to dose conversion factor of 0.06 per Sv (HPA, 2009), and assuming that the event is certain to occur, the risk guidance level corresponds to a dose of approximately 20 µSv y^{-1}. For situations where the probability of receiving a dose is less than one, doses could be greater than 20 µSv y^{-1} while still maintaining consistency with the risk guidance level and, for situations where the probability is very much less than one, doses could be very much greater than 20 µSv y^{-1}. Where probability is less than 1 justification for any adopted value is required.

183. In a number of cases (the gas and groundwater pathways), we have assumed that the probability of an impact being received is unity. In some cases, this is cautious e.g. for the water abstraction well. In such circumstances the risk guidance level may be assumed to correspond to a dose guidance level of 20 µSv y^{-1}.

184. The NS-GRA does not lay down an absolute requirement for the risk guidance level to be met. The value of 10^{-6} y^{-1} (per year) is consistent with HSE advice that this is "a very low level of risk" above which people may be prepared to tolerate risks in order to secure benefits and below which risks are broadly accepted (HSE, 2001). The "risk guidance level" does not apply to human intrusion scenarios as these have a specific dose guidance level (see Section 6.3).

185. This ESC provides a quantitative assessment of the potential future effects of the contamination that can be compared with the risk criterion, using systematically developed and justified, site-specific mathematical models. A cautious best estimate approach is adopted when selecting parameter values and the models themselves are cautious.

6.2.1 Dose assessments after the period of authorisation

186. The results of the assessments relating to longer term impacts, after the period of authorisation (post-closure), are described in Appendix E, Section E.4. The radiological assessment has considered a range of potential scenarios and these are summarised in Table 9. Intrusion scenarios are addressed in Section 6.3. In cases where a scenario has not been explicitly assessed, because it will not or is very unlikely to occur at the ENRMF, the reasons for this are discussed. The scenarios discussed below are divided into two broad categories – those that are expected to occur and those which have a low likelihood of occurrence. The dose assessment considers exposure of members of the public after the period of authorisation.

Table 9 Summary of radiological assessment scenarios considered after the period of authorisation (excluding intrusion scenarios)

Scenario	Exposed group
After the Period of Authorisation – expected to occur	
Recreational user	Member of public
Groundwater abstraction	Farming family
After the Period of Authorisation – not certain to occur	
Water abstraction at site boundary	Farming family
Bathtubbing	Farming family
Very long term climate change	Not explicitly assessed
Other unlikely events	Not explicitly assessed

187. The detailed results of the assessments for the post-closure period are presented in Appendix E.4. The effects of very long term climate change are not assessed because the site is already permitted as a hazardous waste site and LLW disposal gives rise to no additional considerations in respect of flooding, coastal erosion or sea level rises, this is discussed further in Appendix E, paragraph 692. Future glaciation would have similar or lesser effects than the "residential intrusion scenario" considered in Appendix E.5.6. The list in Table 9 includes a category of "Other unlikely events" which covers seismic events, transport accidents and a criticality event. The reasons why these events have not been assessed in detail are given in Appendix E.4.

6.2.2 Impact on recreational users due to gas releases and external radiation

188. The intended end use of the site includes woodland and grassland with paths and a view point. An assessment is therefore made of the doses to a member of the public who spends time walking over the restored site for about 2 h d^{-1} (hours per day) and is exposed to gases released from the waste and receives external exposure from buried waste packages. The results are calculated at the time of closure and after 60 years (the assumed period of authorisation). The assessment includes the effects of radioactive decay and ingrowth upon the calculated doses. Doses from radon gas are shown under Ra-226.

189. Table 10 presents the dose rate per MBq (µSv y^{-1} MBq^{-1}) calculated from the assessment in Appendix E.4.2 at the time of site closure. The radionuclide specific doses arising from disposing of the maximum inventory (minimum of 89.6 TBq and the radiological capacity) are also presented in Table 10 where the calculated dose is greater than 10^{-13} µSv y^{-1}. The highest dose is from C-14 (14.9 µSv y^{-1}), and the peak dose will always be lower than this due to application of the sum of fractions approach.

190. The assumptions concerning gas release in this period are conservative and this results in gas doses dominating exposures to recreational users of the site.

Table 10 Doses to recreational users of restored site at time of closure

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
H-3	$8.96\ 10^{7}$	$4.86\ 10^{-9}$	0	$4.86\ 10^{-9}$	$4.35\ 10^{-1}$
C-14	$8.96\ 10^{7}$	$1.67\ 10^{-7}$	$1.23\ 10^{-73}$	$1.67\ 10^{-7}$	$1.49\ 10^{1}$
Co-60	$8.96\ 10^{7}$		$5.23\ 10^{-18}$	$5.23\ 10^{-18}$	$4.68\ 10^{-10}$
Nb-94	$8.96\ 10^{7}$		$2.79\ 10^{-20}$	$2.79\ 10^{-20}$	$2.50\ 10^{-12}$
Ag-108m	$8.96\ 10^{7}$		$1.64\ 10^{-21}$	$1.64\ 10^{-21}$	$1.47\ 10^{-13}$
Cs-134	$8.96\ 10^{7}$		$9.88\ 10^{-21}$	$9.88\ 10^{-21}$	$8.85\ 10^{-13}$
Cs-137	$8.96\ 10^{7}$		$1.85\ 10^{-21}$	$1.85\ 10^{-21}$	$1.66\ 10^{-13}$
Eu-152	$8.96\ 10^{7}$		$1.58\ 10^{-19}$	$1.58\ 10^{-19}$	$1.42\ 10^{-11}$
Eu-154	$8.96\ 10^{7}$		$2.28\ 10^{-19}$	$2.28\ 10^{-19}$	$2.04\ 10^{-11}$
Ra-226*	$8.96\ 10^{7}$	$1.49\ 10^{-16}$	$1.58\ 10^{-35}$	$1.49\ 10^{-16}$	$1.33\ 10^{-8}$
Ra-228	$8.96\ 10^{7}$		$4.48\ 10^{-16}$	$4.48\ 10^{-16}$	$4.02\ 10^{-8}$
Th-232	$7.16\ 10^{7}$		$8.39\ 10^{-19}$	$8.39\ 10^{-19}$	$6.01\ 10^{-11}$
Others					$<1.00\ 10^{-13}$

* The gas dose shown for Ra-226 is from the release of Rn-222.

6.2.3 Impact due to groundwater extracted at a well off-site

191. The groundwater risk assessment takes into account gradual deterioration of the waste cell liner (see Appendix E.3.4.1). This assumes a doubling time every 100 years for liner defects that allow a flux of water from the waste cells to the unsaturated zone beneath the waste cells and subsequently to the groundwater.

192. Water abstraction at an existing off-site well was modelled using GoldSim and annual doses were calculated from drinking contaminated water and from the use of water for irrigation (see Appendix E.3.4.4). The activity concentration at the well varies over time, generally rising to a peak and then subsequently reducing. The peak activity concentration was used to derive the annual dose and hence these values are peak annual doses. The calculated peak annual doses from groundwater extracted at the existing well nearest to the ENRMF are lower than the peak annual doses from a potential future new well located at the boundary of the site.

193. The results in Table 11 show the dose at the boundary and the radionuclides resulting in the largest doses. Complete sets of results are presented in Table 78 and Table 79.

Table 11 Peak doses due to groundwater abstraction after the period of authorisation

Radionuclide	Maximum inventory (MBq)	Drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Total ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of Max (y)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
I-129	$4.17\ 10^{4}$	$1.01\ 10^{-4}$	$3.79\ 10^{-4}$	$4.80\ 10^{-4}$	2,100	20
Np-237	$4.52\ 10^{5}$	$9.35\ 10^{-6}$	$3.49\ 10^{-5}$	$4.43\ 10^{-5}$	26,095	20
Cl-36	$1.48\ 10^{6}$	$2.42\ 10^{-6}$	$1.11\ 10^{-5}$	$1.35\ 10^{-5}$	759	20
U-235	$4.92\ 10^{6}$	$6.73\ 10^{-7}$	$3.40\ 10^{-6}$	$4.07\ 10^{-6}$	100,000	20
U-234	$6.41\ 10^{6}$	$4.39\ 10^{-7}$	$2.68\ 10^{-6}$	$3.12\ 10^{-6}$	100,000	20
U-238	$2.53\ 10^{7}$	$1.25\ 10^{-7}$	$6.64\ 10^{-7}$	$7.89\ 10^{-7}$	100,000	20
U-233	$3.13\ 10^{7}$	$1.21\ 10^{-7}$	$5.17\ 10^{-7}$	$6.38\ 10^{-7}$	100,000	20
U-236	$8.96\ 10^{7}$	$2.94\ 10^{-8}$	$1.10\ 10^{-7}$	$1.39\ 10^{-7}$	100,000	12
Tc-99	$8.96\ 10^{7}$	$2.37\ 10^{-8}$	$1.02\ 10^{-7}$	$1.26\ 10^{-7}$	5,205	11
Th-232	$7.16\ 10^{7}$	$4.88\ 10^{-9}$	$1.18\ 10^{-7}$	$1.23\ 10^{-7}$	100,000	9

194. Groundwater abstraction restricts the disposal capacity of the ENRMF for seven radionuclides: I-129, Np-237, Cl-36, U-235, U-234, U-238 and U-233. Hence the dose arising from the maximum inventory is 20 $\mu Sv\ y^{-1}$ for these radionuclides (and the maximum inventory is equal to the radiological capacity), as shown in Table 11; this dose corresponds to the risk guidance level for a scenario with a probability of occurrence of unity. The doses ($\mu Sv\ y^{-1}$) arising from disposal of the maximum inventory (minimum of 89.6 TBq and the radiological capacity) are also given for the other radionuclides in the last two columns. Table 11 also shows that the time at which the peak dose occurs in the future varies from 759 years to more than 100,000 years, depending on the radionuclide. The GoldSim calculations are evaluated to 100,000 years.

195. The variability in time to peak dose means that the sum of fractions approach will be overly cautious. For example the peak dose for I-129 occurs at 2,100 years, but the dose due to Cl-36 at that time will be less than that shown in Table 11 as the peak has passed. The peak dose to an individual, summed over radionuclides at any particular time, could be evaluated for a known inventory once disposals have occurred and this could be used to determine a more accurate estimate of the residual disposal capacity at intermediate stages before the ENRMF closes.

196. Table 12 lists all the radionuclides for which the groundwater pathway is the dominant scenario. Where the dose from the maximum inventory is less than 20 $\mu Sv\ y^{-1}$ the potential disposal inventory (maximum inventory) is constrained by the limit on the tonnage (448,000 t) that can be disposed at a specific activity of 200 $Bq\ g^{-1}$.

Table 12 Radionuclides where the dominant scenario is groundwater abstraction

Radionuclide	Maximum inventory (MBq)	Drinking water pathway (µSv y^{-1} MBq^{-1})	Irrigation pathway (µSv y^{-1} MBq^{-1})	Total (µSv y^{-1} MBq^{-1})	Time of Max (y)	Dose from maximum inventory (µSv y^{-1})
Cl-36	$1.48\ 10^{6}$	$2.42\ 10^{-6}$	$1.11\ 10^{-5}$	$1.35\ 10^{-5}$	759	20
Sn-126	$8.96\ 10^{7}$	$6.32\ 10^{-9}$	$8.47\ 10^{-8}$	$9.10\ 10^{-8}$	100,000	8
I-129	$4.17\ 10^{4}$	$1.01\ 10^{-4}$	$3.79\ 10^{-4}$	$4.80\ 10^{-4}$	2,100	20
U-233	$3.13\ 10^{7}$	$1.21\ 10^{-7}$	$5.17\ 10^{-7}$	$6.38\ 10^{-7}$	100,000	20
U-234	$6.41\ 10^{6}$	$4.39\ 10^{-7}$	$2.68\ 10^{-6}$	$3.12\ 10^{-6}$	100,000	20
U-235	$4.92\ 10^{6}$	$6.73\ 10^{-7}$	$3.40\ 10^{-6}$	$4.07\ 10^{-6}$	100,000	20
U-236	$8.96\ 10^{7}$	$2.94\ 10^{-8}$	$1.10\ 10^{-7}$	$1.39\ 10^{-7}$	100,000	12
U-238	$2.53\ 10^{7}$	$1.25\ 10^{-7}$	$6.64\ 10^{-7}$	$7.89\ 10^{-7}$	100,000	20
Np-237	$4.52\ 10^{5}$	$9.35\ 10^{-6}$	$3.49\ 10^{-5}$	$4.43\ 10^{-5}$	26,095	20
Pu-242	$8.96\ 10^{7}$	$8.31\ 10^{-9}$	$3.23\ 10^{-8}$	$4.06\ 10^{-8}$	100,000	4

6.2.4 Doses from uncertain events after the period of authorisation

Exposure due to groundwater extracted at site boundary

197. The existence of this well is considered as an uncertain event, but the results are used in Section 6.2.3 as the basis for limiting radiological capacity.

Exposure as a result of bathtubbing

198. Calculations to show the impact of bathtubbing have been undertaken (Appendix E, Section E.4.5). The scenario involves degradation of the cap leading to saturation of a waste cell and overtopping of the side liner. As leachate level monitoring will continue following completion of filling, capping and placement of the restoration materials, leachate levels will be controlled as necessary in accordance with the Environmental Permit so that compliance limits are not exceeded. The control of leachate levels at the site will continue until it is considered by the Environment Agency that the landfill is unlikely to present a significant risk to the environment if leachate management ceases. Even following the cessation of active leachate management, regulatory control at the site will be maintained through the Environmental Permit. The Environmental Permit cannot be surrendered until the Environment Agency consider that the site no longer presents a potential risk to groundwater. On this basis the potential for overtopping of leachate at a stage when the leachate could have an unacceptable impact on the environment is unlikely to occur. Accordingly the bathtubbing event is considered very unlikely to occur. Nevertheless the impact of a single event 450 years after closure has been modelled using GoldSim, The time corresponds to 200 years after the onset of cap degradation and is the point in time the groundwater model suggests overtopping will occur.

199. There are no local hydrological features that suggest there will be a build-up of surface water following overtopping, the local fields are well drained and there is one minor surface drainage water channel to the south and east of the site (downslope). The restored site will have drainage channels near the boundary to collect excess surface water and direct this to constructed ponds and then to natural drainage channels to the northwest and southeast of

the site. It is considered likely that overtopping will drain to sub-soil rather than flood and saturate an extensive area or percolate to the site drainage channels which may have degraded after 450 years.

200. The scenario assumes that an area around the site (3 ha) is subject to an inundation event due to bathtubbing; this is a small area relative to the size of the landfill and all activity is assumed to accumulate in the affected area. Seepage will occur at the top of the side liner and this will be at least 1 m below ground level. It is also assumed that 1% of the activity introduced at depth (>1 m) reaches the cultivated surface soils (Shaw, et al., 2004). The remainder is assumed to drain to sub-strata based on the good drainage observed in the surrounding area. No account is taken of potential dilution by rain falling in the surrounding area and draining to the same point. The doses are calculated for a household.

201. The results for this scenario are presented in Table 13 for the ten radionuclides giving the highest doses (µSv y^{-1}). Doses based on disposing of the maximum inventory (minimum of 89.6 TBq and the radiological capacity) are shown in the last column of Table 13. A complete set of results is presented in Table 80.

Table 13 Maximum doses for adults resulting from bathtubbing (overtopping of the side liner)

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose (µSv y^{-1} MBq^{-1})	Dose from maximum inventory (µSv y^{-1})
Tc-99	$8.96\ 10^{7}$	$2.02\ 10^{-7}$	18.06
Nb-94	$8.96\ 10^{7}$	$3.45\ 10^{-9}$	0.31
Ag-108m	$8.96\ 10^{7}$	$2.93\ 10^{-9}$	0.26
Ra-226	$8.96\ 10^{7}$	$2.74\ 10^{-9}$	0.25
Th-232	$7.16\ 10^{7}$	$3.08\ 10^{-9}$	0.22
Cl-36	$1.48\ 10^{6}$	$1.25\ 10^{-7}$	0.18
Sn-126	$8.96\ 10^{7}$	$1.21\ 10^{-9}$	0.11
Th-230	$6.93\ 10^{7}$	$5.74\ 10^{-10}$	0.04
Pa-231	$1.86\ 10^{7}$	$1.55\ 10^{-9}$	0.03
Np-237	$4.52\ 10^{5}$	$1.88\ 10^{-8}$	0.01

202. The highest dose is from Tc-99, calculated to be 18.1 µSv y^{-1} if this was the only radionuclide disposed at the ENRMF and $8.9\ 10^{7}$ MBq were disposed of.

6.3 Human intrusion after the period of authorisation {R7}

203. The NS-GRA provides dose guidance levels to be used for assessments of human intrusion after the period of authorisation (Requirement 7):

"The developer/operator of a near-surface disposal facility should assess the potential consequences of human intrusion into the facility after the period of authorisation on the basis that it is likely to occur. The developer/operator should, however, consider and implement any practical measures that might reduce the chance of its happening. The assessed effective dose to any person during and after the assumed intrusion should not exceed a dose guidance level in the range of around 3 mSv/year to around 20 mSv/year. Values towards the lower end of this range are applicable to assessed exposures

continuing over a period of years (prolonged exposures), while values towards the upper end of the range are applicable to assessed exposures that are only short term (transitory exposures)." (Environment Agencies, 2009), para 6.3.36

204. The NS-GRA defines human intrusion as any human action that accesses the waste or that damages a barrier providing an environmental safety function after the period of authorisation.

205. The NS-GRA (paragraph 6.3.41) requires assessment of future human intrusion into the facility assuming that either the intruder does not have prior knowledge of the disposal facility, or that the intruder has knowledge of the existence of underground workings but does not understand what they contain. It is not necessary to assess intrusions undertaken with full knowledge of the existence, location, nature and contents of the disposal facility; the environment agencies take the view that a society that preserves full knowledge of the disposal facility will be capable itself of exercising proper control over any intrusions into the disposal system. Therefore, the human actions that must be assessed are deliberate acts, for example, to excavate a void or recover materials, but where the intruder is uninformed or oblivious to the radiological hazard. The standard against which human intrusion into a near-surface disposal facility should be assessed is specified in terms of dose, not risk, because the environment agencies believe that the likelihood of human intrusion cannot reliably be assessed in terms of a probability (NS-GRA (Environment Agencies, 2009), para 6.3.38).

206. The NS-GRA dose guidance level of 3 mSv y^{-1} to 20 mSv y^{-1} indicates the standard of environmental safety to be achieved. The guidance levels should not be interpreted as limits and are the same as the levels given in advice issued by the HPA in their publication on the disposal of solid radioactive waste (HPA, 2009).

207. The lower dose criterion of 3 mSv y^{-1} is applied in this ESC for prolonged exposure resulting from human intrusion. Doses in this section are presented as mSv.

6.3.1 Dose assessments following intrusion after the period of authorisation

208. The results of the assessments relating to intrusion, after the period of authorisation (post-closure), are described in Appendix E, Section E.5. The radiological assessment has considered a range of potential scenarios and these are summarised in Table 14. The scenarios discussed below consider both workers and members of the public.

Table 14 Summary of radiological assessment scenarios following intrusion after the period of authorisation

Scenario	Exposed group	Time after closure
Borehole drilling	Worker	60 years
Trial pit excavation	Worker	60 years
Laboratory analyst	Worker	60 years
Gas release and external exposure	Site resident	150 years
Housing	Excavation worker and Resident	150 years
Smallholder	Farming family	200 years
Site re-engineering or removal	not assessed	
Particles and large items	Worker and resident	60 years and 300 years

6.3.2 Dose to workers excavating at the site

209. The exposure of workers who excavate waste at the site has been assessed over two timeframes. It is assumed that small excavations may occur at the site in the short term after closure (60 years) and that larger excavations may occur in the longer term (150 or 200 years). LLW, other waste and cover material are assumed to be excavated. If the LLW is disposed of at a depth greater than 5 m then it would not be extracted or disturbed by small or large excavations and the resulting doses to workers excavating at the site would be zero. The doses to a trial pit excavator (see full results in Table 86) are always lower than for borehole drilling so these are not compared below (see full results in Table 84). It is assumed that a single drilling engineer is involved in 5 boreholes (Hicks & Baldwin, 2011), i.e. the potential dose arising from 5 intrusion events is calculated. The results for the ten radionuclides giving the largest impacts are summarised in Table 15 alongside the potential dose arising from disposing of the maximum inventory (minimum of 89.6 TBq and the radiological capacity).

210. The dose (and hence derived quantities such as the radiological capacity) to the worker in the human intrusion scenarios depends upon the duration of exposure and the activity concentration in the excavated waste. Both of the scenarios presented in Table 15 use exposure times of 80 hours per year to contaminated material and hence it would be expected that the doses would be identical. However, the excavation for housing (150 years) is assumed to occur later than the borehole drilling scenario (60 years) and radioactive decay and ingrowth modifies the doses accordingly.

Table 15 Highest doses to workers excavating at the site for each time period

Radionuclide	Maximum inventory (MBq)	Borehole drilling (60y)		Excavation for housing (150y)	
		Dose per unit disposal (mSv MBq^{-1})	Dose from maximum inventory (mSv)	Dose per unit disposal (mSv MBq^{-1})	Dose from maximum inventory (mSv)
Th-232	$7.16\ 10^{7}$	$2.66\ 10^{-8}$	1.90	$4.39\ 10^{-9}$	$3.15\ 10^{-1}$
Ra-226	$8.96\ 10^{7}$	$1.93\ 10^{-8}$	1.73	0 ($1.51\ 10^{-8}$)*	0 (1.08)*

Radionuclide	Maximum inventory (MBq)	Borehole drilling (60y)		Excavation for housing (150y)	
		Dose per unit disposal (mSv MBq^{-1})	Dose from maximum inventory (mSv)	Dose per unit disposal (mSv MBq^{-1})	Dose from maximum inventory (mSv)
Nb-94	8.96 10^{7}	1.25 10^{-8}	1.12	6.27 10^{-10}	5.62 10^{-2}
Ag-108m	8.96 10^{7}	1.13 10^{-8}	1.01	5.66 10^{-10}	5.07 10^{-2}
Th-229	8.96 10^{7}	1.01 10^{-8}	0.90	4.99 10^{-9}	4.47 10^{-1}
Pa-231	1.86 10^{7}	2.17 10^{-8}	0.40	1.21 10^{-8}	2.25 10^{-1}
Pu-239	8.96 10^{7}	3.74 10^{-9}	0.34	2.29 10^{-9}	2.06 10^{-1}
Pu-240	8.96 10^{7}	3.73 10^{-9}	0.33	2.28 10^{-9}	2.05 10^{-1}
Pu-242	8.96 10^{7}	3.44 10^{-9}	0.31	2.11 10^{-9}	1.89 10^{-1}
Sn-126	8.96 10^{7}	3.29 10^{-9}	0.30	1.66 10^{-10}	1.48 10^{-2}

* Waste containing significant activity concentrations of radium is placed at least 5 m deep so would not be excavated. Hence excavation dose from maximum inventory would be zero. Values in parenthesise assume disposal at any depth limited to <5 Bq g^{-1}.

211. The highest doses occur for Ra-226 and Th-232 with doses of about 2 mSv for disposal of 89.6 TBq and 71.6 TBq respectively at the site (radiological capacity calculations are presented in Section 7.4). These calculated doses are below the dose guidance level for intrusion. Placing wastes containing significant activity concentrations of Ra-226 below 5 m in the site results in a zero dose for housing excavations, but a drill may go below that depth and result in a dose to the operator. The placement depth within the ENRMF for wastes containing significant activity concentrations of Ra-226 is discussed in Section 6.3.6 and Appendix E (see Section E.5.8.2).

212. Table 16 lists all the radionuclides for which the borehole drilling scenario at 60 years gives the highest doses i.e. is the dominant scenario in terms of the radiological capacity. For these radionuclides the maximum inventory is constrained by the limit on the tonnage (448,000 t) that can be disposed at 200 Bq g^{-1} and hence all the doses from disposal of the maximum inventory are less than 3 mSv y^{-1}.

Table 16 Dose to workers from the borehole drilling scenario for radionuclides for which this is the dominant scenario

Radionuclide	Maximum inventory (MBq)	Borehole excavator (60y)	
		Dose from unit disposal (mSv MBq^{-1})	Dose from maximum inventory (mSv)
Fe-55	8.96 10^{7}	5.15 10^{-20}	4.61 10^{-12}
Co-60	8.96 10^{7}	7.84 10^{-12}	7.02 10^{-4}
Sb-125	8.96 10^{7}	9.00 10^{-16}	8.06 10^{-8}
Ba-133	8.96 10^{7}	4.91 10^{-11}	4.40 10^{-3}
Cs-134	8.96 10^{7}	2.19 10^{-17}	1.97 10^{-9}

Radionuclide	Maximum inventory (MBq)	Borehole excavator (60y)	
		Dose from unit disposal ($mSv\ MBq^{-1}$)	Dose from maximum inventory (mSv)
Cs-137	$8.96\ 10^{7}$	$1.11\ 10^{-9}$	$9.95\ 10^{-2}$
Pm-147	$8.96\ 10^{7}$	$6.32\ 10^{-20}$	$5.67\ 10^{-12}$
Eu-152	$8.96\ 10^{7}$	$4.19\ 10^{-10}$	$3.75\ 10^{-2}$
Eu-154	$8.96\ 10^{7}$	$7.86\ 10^{-11}$	$7.05\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$3.79\ 10^{-14}$	$3.40\ 10^{-6}$
Ra-226	$8.96\ 10^{7}$	$1.93\ 10^{-8}$	1.73
Ra-228	$8.96\ 10^{7}$	$1.67\ 10^{-11}$	$1.50\ 10^{-3}$
Ac-227	$8.96\ 10^{7}$	$2.95\ 10^{-9}$	$2.64\ 10^{-1}$
Th-229	$8.96\ 10^{7}$	$1.01\ 10^{-8}$	$9.01\ 10^{-1}$
U-232	$8.96\ 10^{7}$	$7.06\ 10^{-10}$	$6.32\ 10^{-2}$
Pu-238	$8.96\ 10^{7}$	$2.14\ 10^{-9}$	$1.92\ 10^{-1}$
Pu-239	$8.96\ 10^{7}$	$3.74\ 10^{-9}$	$3.36\ 10^{-1}$
Pu-240	$8.96\ 10^{7}$	$3.73\ 10^{-9}$	$3.34\ 10^{-1}$
Pu-241	$8.96\ 10^{7}$	$9.38\ 10^{-11}$	$8.41\ 10^{-3}$
Am-241	$8.96\ 10^{7}$	$2.78\ 10^{-9}$	$2.49\ 10^{-1}$
Cm-243	$8.96\ 10^{7}$	$7.02\ 10^{-10}$	$6.29\ 10^{-2}$
Cm-244	$8.96\ 10^{7}$	$1.88\ 10^{-10}$	$1.69\ 10^{-2}$

213. The dose to a trial pit excavator who uncovers just LLW i.e. a single consignment of 10 t, with a specific activity of 200 $Bq\ g^{-1}$, was also assessed. Assuming a homogeneous consignment the highest doses are for Th-232 and Pa-231 which were between 2 and 2.5 $mSv\ y^{-1}$ (see Appendix E, Section E.5.3.2). Further analysis was undertaken to consider the dose that could occur if a disproportionate amount of activity in a 10t consignment was in a single package and this package was examined for 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 (Th-232 and Pa-231) are between 3 and 4 $mSv\ y^{-1}$ (see Appendix E Section E.7.3).

214. On this basis, the calculation supports a range of activity concentration within a 10 t consignment of up to 1000 $Bq\ g^{-1}$, with a specific activity limit of 200 $Bq\ g^{-1}$ for the consignment.

6.3.3 Dose to Laboratory Analyst on Site 60 Years after closure

215. The LLWR human intrusion assessment (Hicks & Baldwin, 2011) suggests that a reasonable assumption is the analysis of 25 samples in a year. The methodology described (Appendix E, Section E.5.4) is for a single sample and all results assume 25 samples in total are analysed. The largest dose rates per MBq disposed of for this scenario are for

Th-229, Th-232 and Pa-231 (see full results presented in Table 91) but the other intrusion scenarios give higher doses.

216. Doses for all radionuclides are low (<0.4 mSv based on the maximum inventory) and since the doses are below the dose guidance level for intrusion this scenario does not limit radiological capacity.

6.3.4 Site resident exposure

217. The scenario where housing is built on the site but leaves the cap intact is discussed below. The complete results for gas released from the ENRMF and through external irradiation are presented in Table 99. Note that these results include the effects of radioactive decay and ingrowth after 150 years (the assumed time between site closure and the approval of housing development on the site) upon the calculated doses. The four highest doses shown below are dominated by the gas pathway (Table 17). In the case of Ra-226, the dominant pathway is inhalation of radon gas and results are given for wastes containing two different Ra-226 activity concentrations, reflecting the emplacement strategy. If wastes containing up to 200 Bq g^{-1} of Ra-226 (labelled high content) are disposed of at a depth greater than 5 m the resulting doses from radon are insignificant. Waste containing Ra-226 activity concentrations of <5 Bq g^{-1} disposed of at any depth (labelled low content) produces a greater dose from radon gas. The impact of disposing of Ra-226 at depth (below 5 m) is discussed further in Section 6.3.6.

Table 17 Site resident exposure – cap intact

Radionuclide	Maximum inventory (MBq)	Dose (mSv y^{-1} MBq^{-1})			Dose from maximum inventory ($mSv\ y^{-1}$)
		Gas*	External	Total	
H-3	$8.96\ 10^{7}$	$3.30\ 10^{-13}$	0	$3.30\ 10^{-13}$	$2.95\ 10^{-5}$
C-14	$8.96\ 10^{7}$	$4.28\ 10^{-9}$	$3.97\ 10^{-76}$	$4.28\ 10^{-9}$	$3.84\ 10^{-1}$
Nb-94	$8.96\ 10^{7}$	0	$9.09\ 10^{-23}$	$9.09\ 10^{-23}$	$8.14\ 10^{-15}$
Ag-108m	$8.96\ 10^{7}$	0	$4.19\ 10^{-24}$	$4.19\ 10^{-24}$	$3.76\ 10^{-16}$
Sn-126	$8.96\ 10^{7}$	0	$3.58\ 10^{-24}$	$3.58\ 10^{-24}$	$3.20\ 10^{-16}$
Ra-226** (high content)	$8.96\ 10^{7}$	$1.09\ 10^{-16}$	$4.83\ 10^{-38}$	$1.09\ 10^{-16}$	$9.81\ 10^{-9}$
Th-229	$8.96\ 10^{7}$	0	$2.22\ 10^{-24}$	$2.22\ 10^{-24}$	$1.99\ 10^{-16}$
Th-232	$7.16\ 10^{7}$	0	$2.74\ 10^{-21}$	$2.74\ 10^{-21}$	$1.97\ 10^{-13}$
Ra-226** (low content)	$2.24\ 10^{6}$	$1.46\ 10^{-8}$	$3.07\ 10^{-20}$	$1.46\ 10^{-8}$	$3.26\ 10^{-2}$

* Conservative estimate ignoring the effect of the cap

** The gas dose shown for Ra-226 is from the release of Rn-222.

218. The highest dose is from C-14 and all doses are below the dose guidance level for intrusion. The gas model is very conservative since it makes no allowance for the impact on gas migration of either an intact cap membrane or the concrete raft on which the house is built. The physical barriers will reduce gas migration and doses significantly. This scenario has not therefore been used to constrain the radiological capacity.

6.3.5 Dose to Site Occupants at 150 or 200 Years

219. The dose rates to residents on the site following construction of houses 150 years after landfill capping and to a smallholder on the site 200 years after capping are summarised in Table 18 for the ten radionuclides giving rise to the highest doses at the maximum inventory for each scenario. The table also includes waste containing two different Ra-226 activity concentrations to indicate the doses from placement at different depths. The doses shown in Table 18 for Ra-226 (high content), i.e. up to 200 Bq g^{-1} are due to radon coming from a depth of 4 m: it is assumed that the placement of Ra-226 (high content) below the intrusion depth (i.e. 5 m below the surface), with clearance of a further metre across the site, would leave 4 m of cover in place. It is assumed that wastes containing Ra-226 up to 5 Bq g^{-1} could be disposed of without restriction in the landfill; it is assumed that there is dilution of this low content Ra-226 (a factor of 0.096 is used). The sensitivity of the intrusion doses and radon release to the radium placement depth within the ENRMF is discussed below (see Section 6.3.6).

Table 18 Doses to site residents after 150 years or smallholders after 200 years

Radionuclide	Maximum inventory (MBq)	Resident (150 y)		Smallholder (200 y)	
		Dose per MBq (mSv y^{-1} MBq^{-1})	Dose from the maximum inventory (mSv y^{-1})	Dose per MBq (mSv y^{-1} MBq^{-1})	Dose from the maximum inventory (mSv y^{-1})
Cl-36	$1.48\ 10^{6}$	$5.47\ 10^{-9}$	$8.08\ 10^{-3}$	$5.55\ 10^{-8}$	0.08
Nb-94	$8.96\ 10^{7}$	$1.81\ 10^{-8}$	1.62	$2.09\ 10^{-8}$	1.87
Tc-99	$8.96\ 10^{7}$	$7.52\ 10^{-9}$	0.67	$3.31\ 10^{-8}$	2.96
Ag-108m	$8.96\ 10^{7}$	$1.41\ 10^{-8}$	1.26	$1.50\ 10^{-8}$	1.34
Sn-126	$8.96\ 10^{7}$	$5.35\ 10^{-9}$	0.48	$8.33\ 10^{-9}$	0.75
I-129	$4.17\ 10^{4}$	$1.30\ 10^{-8}$	$5.43\ 10^{-4}$	$1.12\ 10^{-7}$	$4.65\ 10^{-3}$
Th-229	$8.96\ 10^{7}$	$4.85\ 10^{-9}$	0.43	$8.17\ 10^{-9}$	0.73
Th-230	$6.93\ 10^{7}$	$8.55\ 10^{-9}$	0.59	$4.33\ 10^{-8}$	3.00
Th-232	$7.16\ 10^{7}$	$3.24\ 10^{-8}$	2.32	$4.19\ 10^{-8}$	3.00
Pa-231	$1.86\ 10^{7}$	$4.25\ 10^{-8}$	0.79	$1.61\ 10^{-7}$	3.00
Ra-226* (high content)	$8.96\ 10^{7}$	$1.62\ 10^{-14}$	$1.46\ 10^{-6}$	$1.49\ 10^{-14}$	$1.34\ 10^{-6}$
Ra-226** (low content)	$2.24\ 10^{6}$	$1.20\ 10^{-6}$	2.68	$5.08\ 10^{-7}$	1.14

* Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface

** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1}) placed within 5 m of the restored surface

220. For the smallholder, the calculations apply critical group consumption rates to the two foodstuffs that give the greatest contribution to the dose, and mean consumption rates to all other foodstuffs. The two foodstuffs giving the highest dose rate varies from radionuclide to radionuclide, for example for U-232 and the higher atomic number actinides they are root vegetables and green vegetables (detailed results are presented in Table 105.). There are also a small number of radionuclides where animal products are included in the two

foodstuffs resulting in the highest dose rates (e.g. Cl-36, Cs-134 and Cs-137). For the resident, the calculations assume that the consumption rate of root vegetables and green vegetables grown in the garden is 50% of the mean consumption rate, a conservative assumption for a household resident where most food is purchased rather than grown on site.

221. The radionuclides for which the residential scenario or the smallholder scenario are the dominant scenarios are shown in Table 19. These scenarios determine the radiological capacity for those radionuclides where the dose from the maximum inventory is calculated to be 3 mSv y^{-1}.

Table 19 Dose from the resident and smallholder scenarios for radionuclides for which this is the dominant scenario

Radionuclide	Maximum inventory (MBq)	Resident (150 y)		Smallholder (200 y)	
		Dose per MBq (mSv y^{-1} MBq^{-1})	Dose from maximum inventory (mSv y^{-1})	Dose per MBq (mSv y^{-1} MBq^{-1})	Dose from maximum inventory (mSv y^{-1})
Ni-63	$8.96\ 10^{7}$	$1.91\ 10^{-12}$	$1.71\ 10^{-4}$	$1.08\ 10^{-11}$	$9.69\ 10^{-4}$
Sr-90	$8.96\ 10^{7}$	$1.13\ 10^{-9}$	0.10	$2.42\ 10^{-9}$	0.22
Nb-94	$8.96\ 10^{7}$	$1.81\ 10^{-8}$	1.62	$2.09\ 10^{-8}$	1.87
Tc-99	$8.96\ 10^{7}$	$7.52\ 10^{-9}$	0.67	$3.31\ 10^{-8}$	2.96
Ag-108m	$8.96\ 10^{7}$	$1.41\ 10^{-8}$	1.26	$1.50\ 10^{-8}$	1.34
Th-230	$6.93\ 10^{7}$	$8.55\ 10^{-9}$	0.59	$4.33\ 10^{-8}$	3.00
Th-232	$7.16\ 10^{7}$	$3.24\ 10^{-8}$	2.32	$4.19\ 10^{-8}$	3.00
Pa-231	$1.86\ 10^{7}$	$4.25\ 10^{-8}$	0.79	$1.61\ 10^{-7}$	3.00
Ra-226* (low content)	$2.24\ 10^{6}$	$1.26\ 10^{-6}$	2.68	$5.08\ 10^{-7}$	1.14

* Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface and the Ra-226 content of wastes disposed of nearer to the surface is limited to 5 Bq g^{-1}

6.3.6 Dose to site occupant from Radium when Building on a Waste/Spoil Mix

222. The site occupant scenario was also evaluated assuming that there was no radium emplacement strategy placing significant radium bearing wastes at a particular depth. Hence, it assumed that a house was built on Ra-226 contaminated waste or spoil excavated from the site. This scenario is described in Appendix E (Section E.5.8) and results are presented in Table 102. The resulting dose from radon gas releases implies that the maximum Ra-226 activity concentration in the excavated wastes that would meet the 3 mSv dose criterion is about 5.6 Bq g^{-1}. This scenario does not consider exposure to the wastes remaining in the site since this is addressed above. Hence, this scenario does not impose a restriction on the Ra-226 activity concentration in the waste below the excavated depth. Since the scenario is only relevant if a dwelling is built on a spoil/waste mixture containing radium bearing waste, waste emplacement strategies within waste cells can be employed to ensure that waste containing > 5 Bq g^{-1} radium is not excavated from the site. If it is cautiously assumed that the maximum depth of any human intrusion event is 5 m, then ensuring that waste containing >5 Bq g^{-1} (significant radium bearing waste) is placed at depths greater than this will prevent mixing of the waste with excavated spoil, and in these

circumstances, this scenario is no longer credible. Hence waste emplacement strategies (i.e. placing significant radium bearing wastes no less than 5 m below the restored surface of the waste cells) are considered for radium bearing wastes.

223. The possibility of radon migration from buried radium bearing wastes through the remaining cell-filling material is also considered. This is the same type of calculation as considered in Appendix E , Section E.3.3, but considering 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 the intact cap. The assessment assumes that all the radon gas only has on average to migrate through 4 m of cover material and ignores the effect of house foundations and impermeable membranes designed to prevent radon ingress. If all radium bearing wastes were placed at depths of greater than 5 m, then this would result in radon migrating through at least 4 m of cell-filling material and as the thickness increases, i.e. the cover depth increases, the dose from radon declines due to radioactive decay during migration. Therefore the assessment represents a very cautious estimate of the dose since significant radium bearing wastes will be placed at various depths from 5 m below the restored surface.

6.3.7 Dose from particles

224. Assessments have been undertaken to calculate the dose that could occur from the disposal of waste containing radioactive particles at the ENRMF. Radioactive particles are small discrete items that could be as small as a grain of sand that could be incorporated in a radioactive waste stream or package. The approach used draws on the work undertaken for the LLWR ESC (Sumerling, 2013) and considers the possibility that future intrusion events could lead to unintentional recovery of, and exposure to, radioactive particles.

225. Following the LLWR ESC approach, exposure to particles will be through one of three pathways:

- inadvertent ingestion;
- inhalation; and,
- external irradiation of skin.

226. Inadvertent ingestion is typically size restricted and it is assumed that particles for ingestion are essentially spherical with a nominal diameter of 1 mm. Inhalation of particles is also size restricted and in this case an upper limit of 10 μm diameter (0.01 mm) is assumed.

227. External exposure of skin is not limited by the size of particle. However, in order to be conservative it is assumed that the particle becomes lodged in direct contact with the skin (for example under a fingernail or toenail) and remains in situ for 8 hours. Consistent with this assumption, a 1 mm diameter is assumed.

228. The LLWR ESC set of particles with different radionuclide characteristics was considered and the doses calculated for exposure to a single particle. The doses are generally dominated by the ingestion scenario and, for some radionuclides, the skin exposure scenario is also important. Ingestion doses are very sensitive to the fraction absorbed across the gastro-intestinal tract. Using measured values of uptake and particle solubility, the doses range from fractions of a mSv to 17 mSv depending on the characteristics of the particle. The results assuming the default ICRP uptake fractions (i.e. conservatively

assuming that the particle dissolves completely in the gastro-intestinal tract) are up to 200 mSv but are considered to be unrealistically conservative.

6.3.8 Dose from heterogeneously large contaminated items

229. 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. 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.

230. Characterisation of wastes is always subject to some uncertainty. Wastes can be homogenised or representatively 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.

231. To consider the potential effects of a range of assumptions regarding the distribution of activity within wastes, the ESC considers heterogeneous large items and demolition rubble. A number of different cases 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). Sensitivity to assumed depth profiles for distribution of activity is explored (see Appendix E, Section E.7.3).

232. Drilling through waste or exposure of waste (through natural processes of erosion or through deliberate human activity) could lead to exposure to heterogeneously contaminated material through external exposure or inhalation of dust or inadvertent ingestion of dust.

233. The assessment considers the case where one or more such boreholes drilled on the site after the end of the period of authorisation may penetrate the contaminated items and waste is retrieved for laboratory analysis. The driller may handle the retrieved core leading 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 the resulting 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.

234. The contamination is assumed to be in the exposed top surface 1 cm of the item.

235. The dose at 60 years is 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 were all well below this.

236. Extrapolating the dose out to 1,000 years (a hypothetical date for ‘natural’ erosion exposing the waste that is used to illustrate impact) gives a dose estimate of 0.03 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 1.5 10^{-6}. Given the grossly conservative nature of the assumption that the contaminated surface 1 cm is uniformly exposed, it is considered that this risk is broadly consistent with the risk guidance criterion of 10^{-6} for the post-closure period.

6.4 Optimisation {R8}

6.4.1 Introduction

237. The NS-GRA requires that radiological risks are as low as reasonably achievable (Requirement 8):

> *The choice of waste acceptance criteria, how the selected site is used and the design, construction, operation, closure and post-closure management of the disposal facility should ensure that radiological risks to members of the public, both during the period of authorisation and afterwards, are as low as reasonably achievable (ALARA), taking into account economic and societal factors. (Environment Agencies, 2009), para 6.3.56*

238. Specific provisions for optimisation against the requirements of the Groundwater Directive have also been introduced (Environment Agency, 2012b). In summary these state that:

> *… the optimisation requirement will potentially entail (a) consideration of alternative design options and (b) establishing an appropriate balance in preventing or limiting, as appropriate, the input of pollutants to groundwater.*

239. The requirement for optimisation in relation to radiological risk may be considered at three levels.

- The design of the ENRMF is consistent with best practice and regulatory requirements for the disposal of hazardous wastes and may therefore be considered to be optimised.
- We have considered a number of specific ways in which the management and the design of the site may be enhanced to achieve an optimised solution for the disposal of radioactive wastes;
- Waste consignors are required to manage wastes in a manner consistent with BAT and must demonstrate that disposal to the ENRMF is an optimal solution and hence consistent with BAT.

240. The first two aspects are discussed below, noting that the third is a matter for consignors.

241. Key aspects of the design of the ENRMF are set out in the 2009 application (Augean, 2009a). Arrangements include:

- A full containment landfill engineering system designed to meet the requirements of the EU Landfill Directive. This requires a basal lining system with, or equivalent to having, a hydraulic conductivity of 1 10^{-9} ms^{-1} or lower and a thickness of no less than 5 m or alternative engineering system which provides a level of environmental protection which meets the groundwater quality criteria set in the EU Groundwater Directives. For the basal liner, the landfill incorporates a 1 m thick layer of reworked, engineered clay with a maximum hydraulic conductivity of 1 10^{-9} ms^{-1} and a 2 mm HDPE synthetic liner. The sidewalls are formed from low permeability engineered clay materials with the HDPE liner placed over these;

- A low permeability cap consisting of a protection layer of 300 mm of soil or clay over the waste, a geomembrane, a geotextile protection layer, a 300 mm granular drainage layer; and at least one metre of soil cover;
- Arrangements for the management of leachate;
- Arrangements for dealing with landfill gases;
- Ancillary systems such as vehicle cleaning equipment;
- A systematic approach to monitoring surface water, groundwater and environmental impacts;
- Restoration of the site to woodland, scrub and species rich neutral grassland with a permissive footpath for public access; and,
- Operational arrangements for site construction, operation, closure, restoration and aftercare.

242. These design attributes and arrangements accord with the Environmental Permitting Regulations. The standard design and approach set out in these regulations, which are the basis of the implemented design and approach at the ENRMF, are the output of an extensive process. The design features and arrangements provide an appropriate strategy to limit the environmental impacts arising from non-radioactive contaminants. In the context of the assumed timescales and approach to landfill risk assessment, these measures will also be effective in limiting the environmental impacts arising from radioactive contaminants. In this sense, the design of the facility may be considered to have been optimised with respect to the release of radioactive contaminants and the arising radiological impacts.

243. As the design of the facility is already recognised as consistent with good practice for landfills and the hazards associated with the proposed disposals of radioactive waste are low (and meet the relevant guidance levels), a detailed and systematic analysis of alternative design and management strategies for the facility has not been undertaken. Rather, the focus has been on consideration of a number of specific alternatives that arise if radioactive wastes are to be disposed. These are discussed in the following subsections.

6.4.2 Alternative strategies for waste emplacement

244. Most large scale human intrusion events (see Section 6.3) only disturb the ground to a limited depth of a few metres and hence if the radioactive waste is placed below that depth then such intrusion events will not disturb it. This is particularly important for radium-bearing wastes, which can give rise to doses from radon if buildings are constructed on waste that has been distributed on the surface as a result of a human intrusion event. Strategies that place the majority of the radioactive waste below the intrusion depth e.g. below 5 m of the restored surface will reduce doses from intrusion.

245. Intrusion doses are dependent on the activity concentration in the material that is excavated and therefore waste emplacement strategies that result in wastes with lower activity concentrations being placed within the top of the site (within the intrusion depth) or co-

disposal of radioactive and non-radioactive wastes within this depth will also minimise doses from intrusion.

246. The doses from the other scenarios depend on the total activity in the landfill site and are therefore not affected significantly by waste emplacement strategies relating to depth of disposal.

247. It is therefore proposed that wastes with significant radium content should be emplaced under at least 5 m of cover. Waste emplacement strategies for other radioactive wastes would be considered if required, bearing in mind the current sequence of cell filling and the importance of intrusion scenarios compared with other exposure scenarios for the radionuclides in the wastes.

6.4.3 Operational approaches

248. A number of approaches have been implemented to manage and optimise potential radiological impacts during operations. Some of the key approaches are discussed in the following paragraphs.

249. The waste packages reduce the probability of doses during operations, reduce leaching post-closure and increase the prospect of the waste being recognised as hazardous during future intrusion.

250. The limit on putrescible materials accepted at the ENRMF ensures that microbial activity is minimised and gaseous release from microbial action or from fire leading to a dose is also minimised.

251. Augean places a constraint on the level of dust on the surface of waste packages to ensure this does not represent a hazard. Wastes placed in the landfill are also covered daily to prevent dust suspension and hence the risk of impacts via the inhalation pathway during the operational period. A check is also undertaken on dose measurements at 1 m above the surface of the covered LLW, to ensure exposure of less than 2 $\mu Sv\ hr^{-1}$. The depth of cover will be increased if necessary to ensure that this limit is not exceeded. These precautions will provide additional confidence that no specific protective measures are needed for workers at the site who are closest to the LLW and will provide additional confidence that anyone off site also is suitably protected.

252. Operational constraints have been put in place to restrict the placement of waste in a landfill cell, placing non-radioactive waste to a specified depth at the base (2 m), sides (2 m) and top (1 m) of a cell. This creates a barrier between the LLW and the side liner of a waste cell which will need to be located when the cell is capped. It also means that all LLW will be 2.6 m below the restored surface of the site. An additional limitation is proposed for wastes with significant radium contamination. Such wastes will be disposed at least 5 m below the restored surface of the site. This places radium below a reasonable intrusion depth and reduces the potential dose due to radon gas release from material extracted from the landfill during intrusion.

253. The profiling of the restored surface will encourage surface runoff, preventing the development of puddles and reducing infiltration. Areas of the site will also be developed as woodland and these areas will have a deeper soil layer over the cap. This will further reduce the chance of intrusion disturbing waste or the prospect of housing development at the site.

6.5 Environmental radioactivity {R9}

254. The NS-GRA asks for an assessment of the impact on non-human species (Requirement 9):

"The developer / operator should carry out an assessment to investigate the radiological effects of a disposal facility on the accessible environment both during the period of authorisation and afterwards with a view to showing that all aspects of the accessible environment are adequately protected." NS-GRA (Environment Agencies, 2009), para 6.3.70

255. There are currently no internationally agreed criteria against which radiological dose assessments for non-human species can be evaluated and, as such, assessors are required to apply best available knowledge to draw conclusions on the potential effects of a facility on the environment (paras 6.3.73 & 6.3.74). Results in this ESC are therefore interpreted taking account of the following:

- the ERICA incremental screening value of 10 µGy h^{-1};
- the FREDERICA effects database; and,
- the derived activity concentration reference levels provided in the ICRP Reference Animals and Plants approach (ICRP, 2008; ICRP, 2008).

256. Consideration is also given to uncertainties inherent in the ERICA assessment approach when applied to sub-surface radioactive waste disposal facilities (see, e.g. the discussion in (Smith, et al., 2010)). We have also considered ongoing developments in the interpretation of screening values, knowledge quality and implied levels of protection at the species or population level (Jackson, et al., 2014).

257. It can be seen (Appendix E.6.3) that for almost all radionuclides assessed the modelled environmental activity concentration is below the limiting value for Terrestrial and Freshwater ecosystems. In many cases the risk quotient indicates that the modelled environmental activity concentration is some orders of magnitude below the limiting activity concentration (see Table 134 and Table 137).

258. One exception to the above is noted, for U-238 the derived risk quotient is exceeded by a factor of 1.04 in the Freshwater ecosystem and the most limiting organism type is identified as Vascular plant. Given the extreme conservatism of the derivation of the freshwater activity concentration it is considered that vascular plants remain adequately protected and certainly the implied dose rate (10.4 µGy h^{-1}), remains below the Environment Agency regulatory action level of 40 µGy h^{-1}.

259. An additional assessment was undertaken for burrowing animals in the waste cells after closure. Given the design of the landfill facility and the design of the cap, it seems very unlikely that burrowing animals will build their warren in the disposed waste.

260. We note that within the regulatory framework the site operator has the obligation to protect a species rather than individual animals. The underlying philosophy of radioactive waste disposal to a landfill is to contain and protect the environment from the waste. This is done by isolating the waste from the many populations of non-human biota around the site. The landfill itself is not part of the environment that is to be protected.

261. The dose rates to burrowing animals such as mice, voles and moles are expected to be zero as their burrows will not be deep enough to reach the waste. If the implied dose rates to rabbits are kept below the Environment Agency regulatory action level of 40 µGy h^{-1} then the radiological capacity would be reduced for 3 radionuclides: Pa-231 by a factor 4, Cm-243 by a factor 6 and Cm-244 by a factor 3.

6.6 Specific activity heterogeneity

262. Under the current permit, solid radioactive wastes can be accepted for disposal at the ENRMF if they do not exceed 200 Bq g^{-1} and this specific activity limit applies to a consignment. This would continue under any new permit. It is expected that individual waste streams will not always have homogeneous physical and chemical characteristics or homogeneous activity concentrations.

263. The ESC includes an assessment of three generic waste forms:

- Homogeneous radioactive wastes;
- Radioactive particles; and,
- Large items that have a contamination profile (e.g. contamination is only near the surface of the item).

264. The homogeneous waste assessment is used to determine the radiological capacity of the ENRMF. Heterogeneity within a consignment was assessed assuming a consignment of 10 t and that up to 50% of the activity in the consignment was contained in one 1 t package. The doses were consistent with the relevant dose criteria. Hence this suggests the application of a waste acceptance criterion of 1000 Bq g^{-1} for small quantities of waste within a consignment.

265. The particle assessment considers intrusion which results in the excavated waste containing a 1 mm particle. The probability of the exposed waste containing a particle is not addressed; although, it would be legitimate to consider probability for exposure to these particles because they are too small to be identified and would not attract attention (HPA, 2005b). The results were obtained for a range of example particles and doses up to 17 mSv were calculated assuming the measured fractional uptake in the gastrointestinal tract. Doses up to 200 mSv were calculated using the default ICRP fractional uptake rates. These levels of dose require that the level of exposure is shown to be optimised and suggest the application of a waste acceptance criterion that limits the activity on a particle to below 1 MBq.

266. The assessment of large items with a contamination profile considers intrusion or erosion that results in large heterogeneously contaminated items becoming exposed. The resulting doses are considered to be consistent with the relevant dose and risk criteria. Hence no waste acceptance criteria regarding heterogeneous contamination within an item are required.

7 Technical Requirements

267. In this section protection against non-radiological hazards at the site is considered. The section then considers the development of the site and the operational aspects of both hazardous waste and LLW operations. Waste acceptance criteria and conditions that could apply to LLW disposals are considered. The last part of this section looks at site monitoring.

7.1 Protection against non-radiological hazards {R10}

268. The NS-GRA includes a requirement that the ESC demonstrates that adequate protection against non-radiological hazards is achieved (Requirement 10):

> *"The developer/operator of a disposal facility for solid radioactive waste should demonstrate that the disposal system provides adequate protection against non-radiological hazards." (NS–GRA (Environment Agencies, 2009) para 6.4.1)*

269. The ENRMF is designed to take hazardous wastes and the HRA (Augean, 2014) for the western landfill area as well as the existing landfill site demonstrates that no unacceptable environmental impacts will arise. The existing landfill at the ENRMF is permitted under the Environmental Permitting Regulations and satisfies the requirements of the Landfill Directive for hazardous waste in terms of the management, engineering and monitoring of the site and an application is being submitted for the variation to the hazardous waste Environmental Permit to include the western landfill area.

270. The wastes accepted at the site are largely hazardous due to harmful, toxic, carcinogenic, irritant or eco-toxic properties. No explosive, flammable, corrosive, oxidising or infectious wastes are accepted at the site. The IMS includes established procedures for safe handling and disposal of the hazardous wastes accepted at the site. These processes are similar to those for the handling of LLW and do not conflict with them.

271. The arrangements for construction design, waste acceptance, groundwater protection, landfill gas management, leachate management, landfill stabilisation, pollution prevention, nuisance prevention and quality assurance, construction quality assurance, maintenance, landfill capping, site restoration, operations, waste handling/placement, security, use of raw materials, secondary wastes, accident arrangements, monitoring, closure, aftercare and surrender are described in existing documentation for the landfill site and will be applied to the western area landfill as well as to the current landfill.

272. These features and arrangements represent a solid foundation for the management of LLW and have been taken into account in the risk assessment for LLW disposal to the extent detailed in this document. The features and arrangements are not described in detail in this document (see Augean (2012a) and references therein). An outline of the key landfill engineering features follows:

- A full containment landfill engineering system designed to meet the requirements of the EU Landfill Directive. This requires a basal lining system with, or equivalent to having, a hydraulic conductivity of 1 10^{-9} ms^{-1} or lower and a thickness of no less than 5 m or alternative engineering system which provides a level of environmental protection which meets the groundwater quality criteria set in the EU Groundwater Directives. For the basal liner, the landfill incorporates a 1 m thick layer of reworked, engineered clay with a maximum hydraulic conductivity of 1 10^{-9} ms^{-1} and a 2 mm

HDPE synthetic liner. The sidewalls are formed from low permeability engineered clay materials with the HDPE liner placed over these;

- A low permeability cap consisting of a protection layer of 300 mm of soil or clay over the waste, a geomembrane, a geotextile protection layer, a 300 mm granular drainage layer; and at least one metre of soil cover;
- Ancillary systems such as vehicle cleaning equipment.
- A surface water, groundwater and environmental monitoring system.
- Restoration of the site to woodland, scrub and species rich neutral grassland with a permissive footpath for public access; and,
- Operational arrangements for site construction, operation, closure, restoration and aftercare.

273. The characteristics of the radioactive wastes introduce no additional non-radiological hazards beyond those already assessed in the HRA (Augean, 2014). Disposed LLW will otherwise be compliant with Augean's Conditions For Acceptance (CFA) specified in site procedure LLW01 (see Section 7.4.3) relating to the non-radioactive properties of the waste (i.e. the proposal is for the disposal of radioactive wastes that would be classified as inert, non-hazardous or hazardous in terms of their content of non-radioactive materials). The impact of non-radioactive properties of the LLW waste are therefore covered by the HRA assessments.

7.2 Site investigation {R11}

274. The NS-GRA includes a requirement that a site investigation has been undertaken (Requirement 11):

> *"The developer/operator of a disposal facility for solid radioactive waste should carry out a programme of site investigation and site characterisation to provide information for the environmental safety case and to support facility design and construction." (Environment Agencies, 2009) para 6.4.6*

275. The site has been the subject of a number of site investigations, the most recent in late 2013, which have characterised the geological and hydrogeological setting of the site. A summary of the results of the site investigation is presented in the HRA (Augean, 2014).

7.3 Use of site and facility design, construction, operation and closure {R12}

276. The NS-GRA includes a requirement concerning the management of the facility from design through to closure (Requirement 12):

> *"The developer/operator of a disposal facility for solid radioactive waste should make sure that the site is used and the facility is designed, constructed, operated and capable of closure so as to avoid unacceptable effects on the performance of the disposal system." (Environment Agencies, 2009) para 6.4.16*

277. The design, construction and operation of the site is in accordance with the Landfill Directive as described in Section 2.3 of this report. The Landfill Directive requires that the site provides long term protection of the environment. The risk assessments reported in the HRA show

that the site will provide an appropriate level of containment for tens of thousands of years. The site uses conventional landfill rather than novel technologies, which provides confidence in the engineered solution.

278. The Environmental Permit for hazardous waste cannot be surrendered until the Environment Agency is satisfied that the site no longer presents a significant potential risk to the environment. Following closure and into the aftercare phase Augean will continue to manage the site in accordance with the Permit. In accordance with the Landfill Directive and the Environmental Permitting Regulations Augean has agreed with the Environment Agency an approach to providing funds for the aftercare of the site in the event that Augean ceases to exist.

7.3.1 LLW operations

279. Most of the LLW that will be accepted at the site will be at a level of activity that can be transported without the need for any specified packaging or containment. Augean have determined that, regardless of whether or not there is a need under the legislation, they will specify that all consignors should send LLW to the ENRMF in drums or double skinned bags or as agreed with the EA. Articles that are too large to be placed in containers will be wrapped. It is a requirement that the activity measured at 1 m from each package face must not exceed 10 μSv hr^{-1} (micro Sieverts per hour).

280. Additional precautions will be implemented after the waste is deposited in the landfill area and has been covered by suitable non-LLW material. Measurements will be made above the surface of the cover material to confirm that the activity measured at 1 m above the surface of the covered LLW would result in an exposure of less than 2 μSv hr^{-1}. The depth of cover will be increased if necessary to ensure that this limit is not exceeded. These precautions will provide additional confidence that no specific protective measures are needed for workers at the site who are closest to the LLW and will provide additional confidence that anyone off site also is suitably protected.

281. Prior to agreement that each specific LLW consignment can be accepted at the site, Augean will require a range of information from the consignor, including detailed characterisation information regarding the physical nature, the chemistry and radioactive content of the waste together with information regarding the quantity, form and proposed packaging of the material. Augean will need to be provided with a copy of the relevant Environment Agency Authorisation or Environmental Permit for the disposal of the waste from the source site. The information will be assessed by Augean Technical Assessors and the site management to determine if the material is suitable for disposal at the site and is consistent with the conditions of the Development Consent Order and Environmental Permit. On approval by the Technical Assessor and site management, the consignor will be permitted to make a booking to deliver the waste to the site. The consignor will be advised of the delivery requirements for the waste, including an external exposure limit of 10 μSv hr^{-1} at a 1 m distance from each package.

282. The LLW will be transported to the site in accordance with relevant transport regulations that apply to the radioactive wastes. The regulations are established to control the risks to vehicle drivers and risks from for example transport accidents that could result in waste spillage. Due to the limited amount of radioactivity in the LLW that can be accepted at the site, most wastes will not need any form of special packaging or shielding during handling or transport. However, as noted above, for ease of handling and in order to minimise the potential for spillage, Augean will oblige waste producers to ensure that waste is transported in enclosed

containers such as drums, bulk bags or other containers. Some large items of waste such as metal sheeting may not be transported in containers but will be wrapped.

283. Prior to the delivery of wastes the timetable and details of the waste will be pre-notified to the site in accordance with the transportation regulations and pre-acceptance checks will be carried out to confirm the suitability of the waste for deposition at the site. Augean will audit the consigning facilities routinely to confirm that the characterisation and packaging procedures are followed. The detailed procedures will be consistent with the requirements of any Environmental Permit issued by the Environment Agency.

284. On arrival at the site and prior to acceptance onto the landfill area, the RPS will confirm that the characterisation information which accompanies the waste load is adequate, conforms to the pre-acceptance information and that the load is acceptable for deposition at the site. Wastes arriving at the landfill will be subject to a physical check on the integrity of the packaging and monitoring to check that the external radiation dose is no more than 10 μSv hr^{-1} at a distance of 1 m from the package. The packages will not be opened or sampled at the site in order to minimise unnecessary exposure.

285. Procedures have been set out to cover the unlikely event that unacceptable wastes arrive at the site. If the wastes can be returned safely to the consignor, they will be refused acceptance at the site and returned to their source. If they may not be safe to return to the sender, quarantine measures will be implemented and the Environment Agency will be notified immediately. The detailed procedures for quarantine are specified in accordance with the radiation protection plan for the site, which is established in accordance with the Environmental Permit and to meet the requirements of the Ionising Radiation Regulations. LLW will not intentionally be accumulated.

286. Once the waste has been accepted, the delivery vehicle will travel along the internal haul roads to an unloading point adjacent to the active landfill area. The waste packages will be lifted from the delivery vehicles using mechanical handling machines such as fork-lift trucks and placed in the landfill. Waste will not be tipped into the landfill. The waste will be disposed of in the operational working cell or cells and will be placed alongside hazardous waste. The disposal of waste will take place only under the supervision of an RPS who will be responsible for the operation of the plant at the disposal face.

287. LLW is not placed within 2 m from the base of the cell and the perimeter seal. No LLW is placed within the top metre of the waste in each cell. Wastes containing significant activity concentrations of radium will be placed at least 5 m below the final restored surface (see Appendix E, Section E.5.8.2).

288. Immediately after placement, the deposited wastes will be covered with a minimum thickness of 300 mm of suitable cover material over all exposed surfaces. The radiation levels at 1 m above the top of the cover material will be measured to check conformance with the specified dose rate of 2 μSv hr^{-1}. If the radiation level exceeds the specified dose rate, additional cover will be placed as necessary until the specified dose rate is achieved.

289. As the predicted doses of radiation to which workers at the site will be exposed are below those specified under the Ionising Radiation Regulations 1999 no workers will be defined as Classified Persons in accordance with the regulations. Specific personal protective equipment additional to the standard equipment used and worn by workers at a hazardous waste landfill site will not be necessary during normal site operations. Passive dosimeters will be worn by staff working in the LLW reception and disposal areas as reassurance to confirm that the exposures received are in accordance with the predictions.

7.4 Waste acceptance criteria {R13}

290. The NS-GRA includes a requirement that the developer/operator of the facility makes sure that the waste accepted for disposal is consistent with the ESC and demonstrates that there are procedures in place to make sure that these criteria are met before waste is emplaced in the facility (Requirement 13).

> *"The developer/operator of a disposal facility for solid radioactive waste should establish waste acceptance criteria consistent with the assumptions made in the environmental safety case and with the requirements for transport and handling, and demonstrate that these can be applied during operations at the facility." (Environment Agencies, 2009) para 6.4.26*

7.4.1 Introduction

291. It is important that only wastes that meet regulatory criteria are accepted for disposal at the ENRMF. Calculations are presented in Appendix E that determine a set of radionuclide-specific limits and Section 7.4.2 discusses how these are used as part of a waste acceptance process. Conditions that are placed on waste consignors and specific controls for waste receipt at the ENRMF are addressed in Section 7.4.3 and 7.4.4.

7.4.2 Determining Radiological Capacity

7.4.2.1 Methodology

292. Radioactive waste that would be disposed at the ENRMF must be consistent with the limits in the permit application. The limit in the existing permit is 200 Bq g^{-1} and the specific activity recorded for compliance with this limit is that for a consignment. In the first two years of ENRMF operation, the average specific activity of disposed consignments was less than 10 Bq g^{-1} and the average activity concentration across the ENRMF landfill site is not expected to be more than a few tens of Bq g^{-1} in the future (assuming that radioactive waste comprises up to 20% of material in the landfill).

293. For most scenarios, it is reasonable to take the view that for each radionuclide the total radiation dose is proportional to the total inventory disposed. When contaminants are transported in groundwater or discharged to a sewer, for example, it is likely that substantial mixing will occur so members of an exposed group are exposed to activity concentrations in environmental media that are a function of an average of those in the landfill. However, for certain cases, it is more reasonable to consider the radiation dose to be proportional to the average activity concentration over some smaller volume of the landfill. This will be true, for example, as a result of growing vegetables on a small plot of contaminated soil where the contamination may derive from only a portion of the disposed waste. This is reasonable because these scenarios involve disruption of the waste and the cap; the exposure mechanism is also likely to result in further mixing of the waste.

294. To account for the possibility that there could be dose contributions from more than one radionuclide at once, a limit is applied that constrains the contribution from each individual radionuclide. A limit, L_{Rn} is defined for each radionuclide corresponding to the total activity within the ENRMF landfill as a whole at which the radiation dose from that radionuclide would be equal to the regulatory criterion. The adopted limit is the lowest value calculated from the

assessment scenarios and is called the radiological capacity. The limit to the disposed activity of that radionuclide, I_{Rn}, should be such that:

$$\sum_{Rn} \frac{I_{Rn}}{L_{Rn}} \leq 1$$

with:

- I_{Rn} is the inventory of radionuclide *Rn* (TBq); and
- L_{Rn} is the limiting radiological capacity for radionuclide *Rn* (TBq).

295. The radionuclide inventory in the site will be assessed using this sum of fractions and no further radioactive waste will be accepted once the sum equals 1. This is a standard approach, as described in an IAEA technical document (IAEA, 2003) and used in other permits (e.g. CD7914 for the Lillyhall landfill site)

296. The dose and risk criteria used to determine the radiological capacity of the ENRMF depends on the scenario being considered. In principle, these can be identified as:

- for site workers, the dose criteria are the site criterion of 1 mSv y^{-1} (see Section 6.1);
- for the public a dose constraint of 300 µSv y^{-1} during the period of authorisation for all exposure pathways other than contamination of groundwater and 20 µSv y^{-1} for exposures based on leachate entering groundwater (see Section 6.1);
- in the post-authorisation period a risk criterion of 10^{-6} y^{-1} for the public is indicated in the NS-GRA and this can be considered equivalent to a dose rate of around 20 µSv y^{-1} (see Section 6.2); and,
- for human intrusion in the post-authorisation period a dose guidance level of 3 mSv y^{-1} is used for prolonged exposure (see Section 6.3).

297. The radiological capacity is the total activity that can be disposed without exceeding the dose criteria specified above. All assessments are based on a disposal of 1 MBq and the results presented as dose per megabecquerel (mSv MBq^{-1} or µSv MBq^{-1}) calculated for each radionuclide considered under each scenario. The appropriate dose criterion divided by the dose per megabecquerel provides the radiological capacity ($L_{Rn,\ Scenario}$) for a given scenario as:

$$L_{Rn,Scenario} = \frac{Dose_{crit}}{Dose_{Rn,Scenario}}$$

with:

- $L_{Rn,\ Scenario}$ is the scenario capacity for radionuclide *Rn* (MBq), also referred to as the scenario radiological capacity;
- $Dose_{crit}$ is the scenario dose criterion (µSv y^{-1} or mSv y^{-1}); and,
- $Dose_{Rn,\ Scenario}$ is the calculated scenario dose for radionuclide *Rn* (µSv MBq^{-1} or mSv MBq^{-1}).

298. The limiting (minimum) scenario capacity for each radionuclide is the radiological capacity, the value L_{Rn} in paragraph 294 that is used in the sum of fractions. The need for a limiting

specific activity below 200 Bq g^{-1} for some radionuclides was also considered for the intrusion scenarios. The calculations indicate that the scenario that considers a resident on a waste/spoil mix also implies a limit on the specific activity of Ra-226 bearing wastes that are disposed of within 5 m of the restored surface of the site. This has been incorporated as a waste emplacement strategy for wastes containing > 5 Bq g^{-1} of Ra-226. No other restrictions on the activity concentration were considered to be necessary.

7.4.2.2 Radiological Capacity

299. The radiological capacity of the ENRMF landfill is presented in four tables showing the limiting scenarios:

- Table 20 Scenario radiological capacity calculated for exposures during the period of authorisation
- Table 21 Scenario radiological capacity calculated for exposures after the period of authorisation
- Table 22 Scenario radiological capacity calculated for exposures from human intrusion - workers
- Table 23 Scenario radiological capacity calculated for exposures from human intrusion – residents and smallholders

300. Each table lists scenarios with a dose per unit disposal (µSv MBq^{-1}) and the scenario radiological capacity ($L_{Rn,\ Scenario}$) calculated as shown above for each radionuclide. For the dose arising from a groundwater pathway, a cut-off at 10^{-10} µSv MBq^{-1} is applied and the capacity is shown as “greater than” indicating the dose per unit disposal is very small. Table 24 lists the radionuclides that have a scenario radiological capacity less than 89.6 TBq. Two values are given for Ra-226 where appropriate: one for wastes containing significant activity concentrations of Ra-226 (>5 Bq g^{-1}) that are buried 5 m below the restored surface, and one for wastes containing small activity concentrations of Ra-226 that could be buried within 5 m of the restored surface.

301. The limiting scenarios are combined in Table 25 which shows the radiological capacity for the ENRMF i.e. the most restrictive scenario radiological capacity and the scenario it corresponds to. These radiological capacity values are proposed for inclusion in the Environment Agency permit variation and would be applied using the sum of fractions approach. This approach will ensure that estimated radiation doses arising from the disposed inventory will never exceed the regulatory criteria whatever the radionuclide mix in the inventory of LLW disposed. The screening value for dose to biota is not intended to represent a limit and no inventory limits are derived based on estimated doses to biota.

302. The limit of 448,000 t LLW disposal at the ENRMF landfill (as specified in the site development order) combined with a specific activity of 200 Bq g^{-1} constrains the maximum disposed inventory to 89.6 TBq.

303. In broad terms, the larger the radiological capacity for a radionuclide in Table 25, the less impact the radionuclide has on constraining inputs to the ENRMF. Considering a single radionuclide, the maximum input to the ENRMF will be controlled either by the calculated radiological capacity or by the limit of 448,000 t of LLW:

- if the radionuclide capacity in Table 25 is greater than 89.6 TBq then the inputs to the ENRMF is constrained by the limit on the tonnage; whereas,

- if the radionuclide capacity Table 25 is less than 89.6 TBq, then the input to the ENRMF is constrained by the radiological capacity.

304. For radionuclides that have a very large radiological capacity, the disposed radionuclide will make only a small contribution to the sum of fractions and will therefore make only a small contribution to the dose. For example the radiological capacity for Eu-154 is 3.8 10^4 TBq, which if 89.6 TBq are disposed (448,000 t at 200 Bq g^{-1}) produces a fraction equal to 0.002 for use in the sum of fractions.

305. The radionuclides listed in Table 24 all have a radiological capacity less than 89.6 TBq. Relatively small disposals of these radionuclides will utilise a larger part of the sum of fractions than an equivalent disposal of Eu-154.

306. Given that the disposal inventory will comprise a range of radionuclides and many of the radionuclides against which potential limits have been identified are present in wastes at only trace concentrations, there is often no realistic likelihood that these limits will be challenged.

307. In addition to the limits set out in Table 25, it is proposed that a category of “Other radionuclides” is included. This category would correspond to radionuclides with half-lives greater than 1 year and that are not otherwise identified in Table 24. This category would be assigned a radiological capacity equal to the lowest capacity in the list in Table 24, i.e. that for I-129: 4.2 10^{-2} TBq (42 GBq).

7.4.2.3 Discussion

308. The sum of fractions approach is an internationally recognised approach (US NRC, 2014) and is considered to be best practice. The sum of fractions methodology described above takes account of the cumulative impact of disposal using the most restrictive scenario for each radionuclide. Steps must be taken to ensure that the accumulated inventory at any time does not result in a sum of fractions exceeding one. Additionally, the total inventory in the site will be controlled by ensuring that the total tonnage of LLW disposed of is consistent with the limits specified in the site development order (448,000 t at a maximum of 200 Bq g^{-1}, equivalent to a total inventory of 89.6 TBq). This is the approach proposed by Augean for the permit variation.

309. An alternative approach would be to attempt to forecast what the disposal inventory will be when the landfill closes and demonstrate that this assumed inventory is consistent with meeting regulatory guidance. For some disposal facilities, such estimates may be possible based on the National Waste Inventory and market projections. However, this approach is not desirable for the ENRMF because the future inventory is very uncertain and subject to future commercial agreements. The existing permit is based on an assumed inventory but the radionuclide mix is not representative of the types of wastes that have been disposed of at the ENRMF.

Table 20 Scenario radiological capacity calculated for exposures during the period of authorisation

Radionuclide	Gas release - public		Recreational (at closure)		Groundwater (Well at boundary)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
H-3	$5.08\ 10^{-9}$	$5.91\ 10^{10}$	$4.86\ 10^{-9}$	$4.12\ 10^{9}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
C-14	$1.74\ 10^{-7}$	$1.72\ 10^{9}$	$1.67\ 10^{-7}$	$1.20\ 10^{8}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Cl-36			$2.66\ 10^{-31}$	$7.51\ 10^{31}$	$7.07\ 10^{-8}$	$2.83\ 10^{8}$
Fe-55			0	nd*	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Co-60			$5.23\ 10^{-18}$	$3.83\ 10^{18}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Ni-63			0	nd*	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Sr-90			$4.01\ 10^{-28}$	$4.98\ 10^{28}$	$1.53\ 10^{-10}$	$1.31\ 10^{11}$
Nb-94			$2.79\ 10^{-20}$	$7.17\ 10^{20}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Tc-99			$5.04\ 10^{-53}$	$3.97\ 10^{53}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Ru-106			$3.78\ 10^{-22}$	$5.29\ 10^{22}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Ag-108m			$1.64\ 10^{-21}$	$1.22\ 10^{22}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Sb-125			$1.21\ 10^{-22}$	$1.66\ 10^{23}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Sn-126			$1.09\ 10^{-21}$	$1.83\ 10^{22}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
I-129			$3.01\ 10^{-155}$	$6.65\ 10^{155}$	$7.83\ 10^{-7}$	$2.55\ 10^{7}$
Ba-133			$1.37\ 10^{-25}$	$1.46\ 10^{26}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Cs-134			$9.88\ 10^{-21}$	$2.02\ 10^{21}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Cs-137			$1.85\ 10^{-21}$	$1.08\ 10^{22}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pm-147			$4.95\ 10^{-52}$	$4.04\ 10^{52}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Eu-152			$1.58\ 10^{-19}$	$1.26\ 10^{20}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Eu-154			$2.28\ 10^{-19}$	$8.77\ 10^{19}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Eu-155			$2.07\ 10^{-44}$	$9.68\ 10^{44}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pb-210			$6.86\ 10^{-26}$	$2.91\ 10^{26}$	$4.50\ 10^{-9}$	$4.44\ 10^{9}$

Radionuclide	Gas release - public		Recreational (at closure)		Groundwater (Well at boundary)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
Ra-226	$2.08\ 10^{-7}$	$1.44\ 10^{9}$	$1.49\ 10^{-16}$	$1.34\ 10^{17}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Ra-228			$4.48\ 10^{-16}$	$4.46\ 10^{16}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Ac-227			$1.48\ 10^{-24}$	$1.35\ 10^{25}$	$2.65\ 10^{-9}$	$7.54\ 10^{9}$
Th-229			$6.88\ 10^{-22}$	$2.91\ 10^{22}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Th-230			$1.25\ 10^{-41}$	$1.60\ 10^{42}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Th-232			$8.39\ 10^{-19}$	$2.38\ 10^{19}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pa-231			$6.04\ 10^{-28}$	$3.31\ 10^{28}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
U-232			$1.01\ 10^{-41}$	$1.97\ 10^{42}$	$2.11\ 10^{-10}$	$9.47\ 10^{10}$
U-233			$2.18\ 10^{-34}$	$9.17\ 10^{34}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
U-234			$1.82\ 10^{-49}$	$1.10\ 10^{50}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
U-235			$5.83\ 10^{-32}$	$3.43\ 10^{32}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
U-236			$4.98\ 10^{-45}$	$4.02\ 10^{45}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
U-238			$1.21\ 10^{-24}$	$1.65\ 10^{25}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Np-237			$2.96\ 10^{-41}$	$6.75\ 10^{41}$	$5.09\ 10^{-9}$	$3.93\ 10^{9}$
Pu-238			$1.32\ 10^{-51}$	$1.52\ 10^{52}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pu-239			$8.55\ 10^{-34}$	$2.34\ 10^{34}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pu-240			$1.18\ 10^{-59}$	$1.69\ 10^{60}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pu-241			$2.84\ 10^{-44}$	$7.04\ 10^{44}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Pu-242			$1.39\ 10^{-72}$	$1.44\ 10^{73}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Am-241			$1.38\ 10^{-67}$	$1.45\ 10^{68}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$
Cm-243			$8.95\ 10^{-32}$	$2.23\ 10^{32}$	$1.11\ 10^{-10}$	$1.80\ 10^{11}$
Cm-244			0	nd*	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$

* Where dose is effectively zero the radiological capacity is infinite, marked here as nd (not determined).

Table 21 Scenario radiological capacity calculated for exposures after the period of authorisation

Radionuclide	Bathtubbing		Groundwater (Well at boundary)		Recreational user	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
H-3	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$5.95\ 10^{-3}$	$3.01\ 10^{11}$
C-14	$1.10\ 10^{-11}$	$1.82\ 10^{12}$	$3.49\ 10^{-9}$	$5.73\ 10^{9}$	$2.97\ 10^{0}$	$6.04\ 10^{8}$
Cl-36	$1.25\ 10^{-7}$	$1.60\ 10^{8}$	$1.35\ 10^{-5}$	$1.48\ 10^{6}$	$3.93\ 10^{-25}$	$7.51\ 10^{31}$
Fe-55	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	0	nd*
Co-60	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.75\ 10^{-13}$	$1.02\ 10^{22}$
Ni-63	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	0	nd*
Sr-90	$1.58\ 10^{-12}$	$1.27\ 10^{13}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$8.48\ 10^{-21}$	$2.11\ 10^{29}$
Nb-94	$3.45\ 10^{-9}$	$5.80\ 10^{9}$	$2.23\ 10^{-9}$	$8.96\ 10^{9}$	$2.50\ 10^{-12}$	$7.18\ 10^{20}$
Tc-99	$2.02\ 10^{-7}$	$9.92\ 10^{7}$	$1.26\ 10^{-7}$	$1.58\ 10^{8}$	$4.52\ 10^{-45}$	$3.97\ 10^{53}$
Ru-106	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$7.49\ 10^{-32}$	$2.39\ 10^{40}$
Ag-108m	$2.93\ 10^{-9}$	$6.83\ 10^{9}$	$9.03\ 10^{-10}$	$2.21\ 10^{10}$	$1.33\ 10^{-13}$	$1.34\ 10^{22}$
Sb-125	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$3.08\ 10^{-21}$	$5.83\ 10^{29}$
Sn-126	$1.21\ 10^{-9}$	$1.66\ 10^{10}$	$9.10\ 10^{-8}$	$2.20\ 10^{8}$	$9.80\ 10^{-14}$	$1.83\ 10^{22}$
I-129	$1.59\ 10^{-7}$	$1.26\ 10^{8}$	$4.80\ 10^{-4}$	$4.17\ 10^{4}$	$1.25\ 10^{-150}$	$6.65\ 10^{155}$
Ba-133	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.35\ 10^{-19}$	$7.63\ 10^{27}$
Cs-134	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.59\ 10^{-21}$	$1.13\ 10^{30}$
Cs-137	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$4.19\ 10^{-14}$	$4.28\ 10^{22}$
Pm-147	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$5.77\ 10^{-51}$	$3.11\ 10^{59}$
Eu-152	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$6.57\ 10^{-13}$	$2.73\ 10^{21}$
Eu-154	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.62\ 10^{-13}$	$1.11\ 10^{22}$
Eu-155	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.98\ 10^{-40}$	$6.02\ 10^{48}$
Pb-210	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$9.45\ 10^{-19}$	$1.90\ 10^{27}$
Ra-226	$2.74\ 10^{-9}$	$7.29\ 10^{9}$	$5.15\ 10^{-9}$	$3.88\ 10^{9}$	$1.30\ 10^{-8}$	$1.38\ 10^{17}$

Radionuclide	Bathtubbing		Groundwater (Well at boundary)		Recreational user	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
Ra-228	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.90\ 10^{-11}$	$6.18\ 10^{19}$
Ac-227	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.96\ 10^{-17}$	$9.15\ 10^{25}$
Th-229	$5.26\ 10^{-11}$	$3.81\ 10^{11}$	$3.68\ 10^{-8}$	$5.44\ 10^{8}$	$6.13\ 10^{-14}$	$8.96\ 10^{7}$
Th-230	$5.74\ 10^{-10}$	$3.48\ 10^{10}$	$6.94\ 10^{-8}$	$2.88\ 10^{8}$	$2.29\ 10^{-29}$	$6.93\ 10^{7}$
Th-232	$3.08\ 10^{-9}$	$6.50\ 10^{9}$	$1.23\ 10^{-7}$	$1.63\ 10^{8}$	$6.01\ 10^{-11}$	$7.16\ 10^{7}$
Pa-231	$1.55\ 10^{-9}$	$1.29\ 10^{10}$	$9.02\ 10^{-8}$	$2.22\ 10^{8}$	$2.33\ 10^{-17}$	$1.86\ 10^{7}$
U-232	$5.16\ 10^{-12}$	$3.87\ 10^{12}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$4.96\ 10^{-34}$	$8.96\ 10^{7}$
U-233	$1.18\ 10^{-10}$	$1.69\ 10^{11}$	$6.38\ 10^{-7}$	$3.13\ 10^{7}$	$1.22\ 10^{-16}$	$3.13\ 10^{7}$
U-234	$8.76\ 10^{-11}$	$2.28\ 10^{11}$	$3.12\ 10^{-6}$	$6.41\ 10^{6}$	$4.33\ 10^{-38}$	$6.41\ 10^{6}$
U-235	$1.42\ 10^{-9}$	$1.41\ 10^{10}$	$4.07\ 10^{-6}$	$4.92\ 10^{6}$	$4.06\ 10^{-24}$	$4.92\ 10^{6}$
U-236	$7.95\ 10^{-11}$	$2.52\ 10^{11}$	$1.39\ 10^{-7}$	$1.44\ 10^{8}$	$2.23\ 10^{-19}$	$8.96\ 10^{7}$
U-238	$3.09\ 10^{-10}$	$6.48\ 10^{10}$	$7.89\ 10^{-7}$	$2.53\ 10^{7}$	$3.07\ 10^{-17}$	$2.53\ 10^{7}$
Np-237	$1.88\ 10^{-8}$	$1.06\ 10^{9}$	$4.43\ 10^{-5}$	$4.52\ 10^{5}$	$2.60\ 10^{-32}$	$4.52\ 10^{5}$
Pu-238	$7.99\ 10^{-13}$	$2.50\ 10^{13}$	$8.28\ 10^{-10}$	$2.42\ 10^{10}$	$7.57\ 10^{-44}$	$8.96\ 10^{7}$
Pu-239	$3.47\ 10^{-11}$	$5.76\ 10^{11}$	$6.62\ 10^{-9}$	$3.02\ 10^{9}$	$7.65\ 10^{-26}$	$8.96\ 10^{7}$
Pu-240	$3.35\ 10^{-11}$	$5.96\ 10^{11}$	$1.51\ 10^{-9}$	$1.32\ 10^{10}$	$7.90\ 10^{-43}$	$8.96\ 10^{7}$
Pu-241	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.92\ 10^{-10}$	$1.04\ 10^{11}$	$1.40\ 10^{-37}$	$8.96\ 10^{7}$
Pu-242	$3.28\ 10^{-11}$	$6.10\ 10^{11}$	$4.06\ 10^{-8}$	$4.93\ 10^{8}$	$1.01\ 10^{-24}$	$8.96\ 10^{7}$
Am-241	$6.89\ 10^{-12}$	$2.90\ 10^{12}$	$8.91\ 10^{-9}$	$2.24\ 10^{9}$	$4.92\ 10^{-38}$	$8.96\ 10^{7}$
Cm-243	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.92\ 10^{-24}$	$8.96\ 10^{7}$
Cm-244	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.62\ 10^{-54}$	$8.96\ 10^{7}$

* Where dose is effectively zero the radiological capacity is infinite, marked here as nd (not determined).

Table 22 Scenario radiological capacity calculated for exposures from human intrusion - workers

Radionuclide	Borehole excavator (60y)		Laboratory analyst (60y)		Trial pit excavator (60y)		Housing site excavator (150y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
H-3	$5.91\ 10^{-13}$	$5.08\ 10^{15}$	$2.49\ 10^{-13}$	$1.21\ 10^{16}$	$1.48\ 10^{-13}$	$2.03\ 10^{16}$	$3.73\ 10^{-15}$	$8.05\ 10^{17}$
C-14	$5.88\ 10^{-10}$	$5.10\ 10^{12}$	$1.85\ 10^{-10}$	$1.62\ 10^{13}$	$1.47\ 10^{-10}$	$2.04\ 10^{13}$	$5.81\ 10^{-10}$	$5.17\ 10^{12}$
Cl-36	$4.29\ 10^{-9}$	$6.99\ 10^{11}$	$4.14\ 10^{-10}$	$7.24\ 10^{12}$	$1.07\ 10^{-9}$	$2.80\ 10^{12}$	$4.28\ 10^{-9}$	$7.01\ 10^{11}$
Fe-55	$5.15\ 10^{-17}$	$5.83\ 10^{19}$	$1.45\ 10^{-17}$	$2.06\ 10^{20}$	$1.29\ 10^{-17}$	$2.33\ 10^{20}$	$6.49\ 10^{-27}$	$4.62\ 10^{29}$
Co-60	$7.84\ 10^{-9}$	$3.83\ 10^{11}$	$3.93\ 10^{-10}$	$7.64\ 10^{12}$	$1.96\ 10^{-9}$	$1.53\ 10^{12}$	$5.67\ 10^{-14}$	$5.29\ 10^{16}$
Ni-63	$7.90\ 10^{-11}$	$3.80\ 10^{13}$	$2.92\ 10^{-11}$	$1.03\ 10^{14}$	$1.98\ 10^{-11}$	$1.52\ 10^{14}$	$4.23\ 10^{-11}$	$7.09\ 10^{13}$
Sr-90	$1.27\ 10^{-8}$	$2.35\ 10^{11}$	$2.05\ 10^{-9}$	$1.47\ 10^{12}$	$3.19\ 10^{-9}$	$9.41\ 10^{11}$	$1.46\ 10^{-9}$	$2.06\ 10^{12}$
Nb-94	$1.25\ 10^{-5}$	$2.40\ 10^{8}$	$6.27\ 10^{-7}$	$4.79\ 10^{9}$	$3.13\ 10^{-6}$	$9.60\ 10^{8}$	$1.24\ 10^{-5}$	$2.41\ 10^{8}$
Tc-99	$1.07\ 10^{-9}$	$2.81\ 10^{12}$	$3.37\ 10^{-10}$	$8.89\ 10^{12}$	$2.66\ 10^{-10}$	$1.13\ 10^{13}$	$1.06\ 10^{-9}$	$2.82\ 10^{12}$
Ru-106	$3.70\ 10^{-24}$	$8.11\ 10^{26}$	$1.89\ 10^{-25}$	$1.58\ 10^{28}$	$9.25\ 10^{-25}$	$3.24\ 10^{27}$	$1.21\ 10^{-50}$	$2.47\ 10^{53}$
Ag-108m	$1.13\ 10^{-5}$	$2.66\ 10^{8}$	$5.66\ 10^{-7}$	$5.30\ 10^{9}$	$2.82\ 10^{-6}$	$1.06\ 10^{9}$	$9.71\ 10^{-6}$	$3.09\ 10^{8}$
Sb-125	$9.00\ 10^{-13}$	$3.33\ 10^{15}$	$4.52\ 10^{-14}$	$6.64\ 10^{16}$	$2.25\ 10^{-13}$	$1.33\ 10^{16}$	$1.36\ 10^{-22}$	$2.21\ 10^{25}$
Sn-126	$3.29\ 10^{-6}$	$9.12\ 10^{8}$	$1.66\ 10^{-7}$	$1.81\ 10^{10}$	$8.22\ 10^{-7}$	$3.65\ 10^{9}$	$3.28\ 10^{-6}$	$9.14\ 10^{8}$
I-129	$7.55\ 10^{-8}$	$3.97\ 10^{10}$	$1.59\ 10^{-8}$	$1.88\ 10^{11}$	$1.89\ 10^{-8}$	$1.59\ 10^{11}$	$7.53\ 10^{-8}$	$3.98\ 10^{10}$
Ba-133	$4.91\ 10^{-8}$	$6.11\ 10^{10}$	$2.47\ 10^{-9}$	$1.22\ 10^{12}$	$1.23\ 10^{-8}$	$2.44\ 10^{11}$	$1.30\ 10^{-10}$	$2.30\ 10^{13}$
Cs-134	$2.19\ 10^{-14}$	$1.37\ 10^{17}$	$1.10\ 10^{-15}$	$2.72\ 10^{18}$	$5.49\ 10^{-15}$	$5.47\ 10^{17}$	$1.66\ 10^{-27}$	$1.81\ 10^{30}$
Cs-137	$1.11\ 10^{-6}$	$2.70\ 10^{9}$	$5.61\ 10^{-8}$	$5.35\ 10^{10}$	$2.78\ 10^{-7}$	$1.08\ 10^{10}$	$1.40\ 10^{-7}$	$2.14\ 10^{10}$
Pm-147	$6.32\ 10^{-17}$	$4.74\ 10^{19}$	$1.71\ 10^{-17}$	$1.75\ 10^{20}$	$1.58\ 10^{-17}$	$1.90\ 10^{20}$	$2.96\ 10^{-27}$	$1.01\ 10^{30}$
Eu-152	$4.19\ 10^{-7}$	$7.16\ 10^{9}$	$2.10\ 10^{-8}$	$1.43\ 10^{11}$	$1.05\ 10^{-7}$	$2.87\ 10^{10}$	$4.17\ 10^{-9}$	$7.19\ 10^{11}$
Eu-154	$7.86\ 10^{-8}$	$3.81\ 10^{10}$	$3.95\ 10^{-9}$	$7.60\ 10^{11}$	$1.97\ 10^{-8}$	$1.53\ 10^{11}$	$5.52\ 10^{-11}$	$5.44\ 10^{13}$
Eu-155	$3.79\ 10^{-11}$	$7.91\ 10^{13}$	$1.92\ 10^{-12}$	$1.56\ 10^{15}$	$9.48\ 10^{-12}$	$3.16\ 10^{14}$	$7.72\ 10^{-17}$	$3.89\ 10^{19}$
Pb-210	$2.00\ 10^{-7}$	$1.50\ 10^{10}$	$6.71\ 10^{-8}$	$4.47\ 10^{10}$	$4.99\ 10^{-8}$	$6.01\ 10^{10}$	$1.20\ 10^{-8}$	$2.50\ 10^{11}$

Radionuclide	Borehole excavator (60y)		Laboratory analyst (60y)		Trial pit excavator (60y)		Housing site excavator (150y)	
	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)
Ra-226*	$1.93\ 10^{-5}$	$1.56\ 10^{8}$	$1.66\ 10^{-6}$	$1.81\ 10^{9}$	$4.82\ 10^{-6}$	$6.22\ 10^{8}$	$1.85\ 10^{-5}$	$1.62\ 10^{8}$
Ra-228	$1.67\ 10^{-8}$	$1.80\ 10^{11}$	$1.65\ 10^{-9}$	$1.82\ 10^{12}$	$4.18\ 10^{-9}$	$7.18\ 10^{11}$	$3.24\ 10^{-13}$	$9.26\ 10^{15}$
Ac-227	$2.95\ 10^{-6}$	$1.02\ 10^{9}$	$1.63\ 10^{-6}$	$1.84\ 10^{9}$	$7.37\ 10^{-7}$	$4.07\ 10^{9}$	$1.68\ 10^{-7}$	$1.79\ 10^{10}$
Th-229	$1.01\ 10^{-5}$	$2.98\ 10^{8}$	$4.99\ 10^{-6}$	$6.01\ 10^{8}$	$2.51\ 10^{-6}$	$1.19\ 10^{9}$	$9.95\ 10^{-6}$	$3.01\ 10^{8}$
Th-230	$3.54\ 10^{-6}$	$8.47\ 10^{8}$	$1.96\ 10^{-6}$	$1.53\ 10^{9}$	$8.85\ 10^{-7}$	$3.39\ 10^{9}$	$4.13\ 10^{-6}$	$7.26\ 10^{8}$
Th-232	$2.66\ 10^{-5}$	$1.13\ 10^{8}$	$4.39\ 10^{-6}$	$6.83\ 10^{8}$	$6.64\ 10^{-6}$	$4.52\ 10^{8}$	$2.65\ 10^{-5}$	$1.13\ 10^{8}$
Pa-231	$2.17\ 10^{-5}$	$1.38\ 10^{8}$	$1.21\ 10^{-5}$	$2.48\ 10^{8}$	$5.44\ 10^{-6}$	$5.52\ 10^{8}$	$2.44\ 10^{-5}$	$1.23\ 10^{8}$
U-232	$7.06\ 10^{-7}$	$4.25\ 10^{9}$	$4.06\ 10^{-7}$	$7.39\ 10^{9}$	$1.76\ 10^{-7}$	$1.70\ 10^{10}$	$2.85\ 10^{-7}$	$1.05\ 10^{10}$
U-233	$3.75\ 10^{-7}$	$7.99\ 10^{9}$	$2.16\ 10^{-7}$	$1.39\ 10^{10}$	$9.38\ 10^{-8}$	$3.20\ 10^{10}$	$4.59\ 10^{-7}$	$6.53\ 10^{9}$
U-234	$3.11\ 10^{-7}$	$9.63\ 10^{9}$	$1.85\ 10^{-7}$	$1.62\ 10^{10}$	$7.78\ 10^{-8}$	$3.85\ 10^{10}$	$3.13\ 10^{-7}$	$9.58\ 10^{9}$
U-235	$1.27\ 10^{-6}$	$2.37\ 10^{9}$	$2.19\ 10^{-7}$	$1.37\ 10^{10}$	$3.17\ 10^{-7}$	$9.47\ 10^{9}$	$1.27\ 10^{-6}$	$2.35\ 10^{9}$
U-236	$2.87\ 10^{-7}$	$1.04\ 10^{10}$	$1.70\ 10^{-7}$	$1.76\ 10^{10}$	$7.18\ 10^{-8}$	$4.18\ 10^{10}$	$2.87\ 10^{-7}$	$1.05\ 10^{10}$
U-238	$4.39\ 10^{-7}$	$6.83\ 10^{9}$	$1.66\ 10^{-7}$	$1.81\ 10^{10}$	$1.10\ 10^{-7}$	$2.73\ 10^{10}$	$4.38\ 10^{-7}$	$6.84\ 10^{9}$
Np-237	$2.99\ 10^{-6}$	$1.00\ 10^{9}$	$1.03\ 10^{-6}$	$2.91\ 10^{9}$	$7.47\ 10^{-7}$	$4.02\ 10^{9}$	$2.98\ 10^{-6}$	$1.01\ 10^{9}$
Pu-238	$2.14\ 10^{-6}$	$1.40\ 10^{9}$	$1.31\ 10^{-6}$	$2.29\ 10^{9}$	$5.35\ 10^{-7}$	$5.61\ 10^{9}$	$1.05\ 10^{-6}$	$2.86\ 10^{9}$
Pu-239	$3.74\ 10^{-6}$	$8.01\ 10^{8}$	$2.29\ 10^{-6}$	$1.31\ 10^{9}$	$9.36\ 10^{-7}$	$3.20\ 10^{9}$	$3.73\ 10^{-6}$	$8.05\ 10^{8}$
Pu-240	$3.73\ 10^{-6}$	$8.05\ 10^{8}$	$2.28\ 10^{-6}$	$1.31\ 10^{9}$	$9.32\ 10^{-7}$	$3.22\ 10^{9}$	$3.68\ 10^{-6}$	$8.14\ 10^{8}$
Pu-241	$9.38\ 10^{-8}$	$3.20\ 10^{10}$	$5.66\ 10^{-8}$	$5.30\ 10^{10}$	$2.35\ 10^{-8}$	$1.28\ 10^{11}$	$8.27\ 10^{-8}$	$3.63\ 10^{10}$
Pu-242	$3.44\ 10^{-6}$	$8.71\ 10^{8}$	$2.11\ 10^{-6}$	$1.42\ 10^{9}$	$8.61\ 10^{-7}$	$3.48\ 10^{9}$	$3.44\ 10^{-6}$	$8.73\ 10^{8}$
Am-241	$2.78\ 10^{-6}$	$1.08\ 10^{9}$	$1.67\ 10^{-6}$	$1.79\ 10^{9}$	$6.94\ 10^{-7}$	$4.32\ 10^{9}$	$2.40\ 10^{-6}$	$1.25\ 10^{9}$
Cm-243	$7.02\ 10^{-7}$	$4.28\ 10^{9}$	$3.28\ 10^{-7}$	$9.15\ 10^{9}$	$1.75\ 10^{-7}$	$1.71\ 10^{10}$	$8.61\ 10^{-8}$	$3.49\ 10^{10}$
Cm-244	$1.88\ 10^{-7}$	$1.59\ 10^{10}$	$1.15\ 10^{-7}$	$2.60\ 10^{10}$	$4.71\ 10^{-8}$	$6.37\ 10^{10}$	$1.59\ 10^{-8}$	$1.89\ 10^{11}$

* House excavation calculation assumes waste containing significant activity concentrations of Ra-226 are 5m below the restored surface

Table 23 Scenario radiological capacity calculated for exposures from human intrusion – residents and smallholders

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
H-3	$3.53\ 10^{-10}$	$8.51\ 10^{12}$	$2.81\ 10^{-11}$	$1.07\ 10^{14}$	$2.95\ 10^{-2}$	$6.06\ 10^{10}$
C-14	$4.35\ 10^{-6}$	$6.90\ 10^{8}$	$3.71\ 10^{-6}$	$8.08\ 10^{8}$	$3.84\ 10^{2}$	$4.67\ 10^{6}$
Cl-36	$5.47\ 10^{-6}$	$5.49\ 10^{8}$	$5.55\ 10^{-5}$	$5.40\ 10^{7}$	$1.29\ 10^{-24}$	$2.30\ 10^{31}$
Fe-55	$1.11\ 10^{-26}$	$2.69\ 10^{29}$	$7.76\ 10^{-31}$	$3.87\ 10^{33}$	0	nd*
Co-60	$8.26\ 10^{-14}$	$3.63\ 10^{16}$	$1.35\ 10^{-16}$	$2.23\ 10^{19}$	$4.16\ 10^{-18}$	$4.31\ 10^{26}$
Ni-63	$1.91\ 10^{-9}$	$1.57\ 10^{12}$	$1.08\ 10^{-8}$	$2.78\ 10^{11}$	0	nd*
Sr-90	$1.13\ 10^{-6}$	$2.64\ 10^{9}$	$2.42\ 10^{-6}$	$1.24\ 10^{9}$	$3.18\ 10^{-21}$	$5.64\ 10^{29}$
Nb-94	$1.81\ 10^{-5}$	$1.66\ 10^{8}$	$2.09\ 10^{-5}$	$1.44\ 10^{8}$	$8.14\ 10^{-12}$	$2.20\ 10^{20}$
Tc-99	$7.52\ 10^{-6}$	$3.99\ 10^{8}$	$3.31\ 10^{-5}$	$9.07\ 10^{7}$	$1.48\ 10^{-44}$	$1.21\ 10^{53}$
Ru-106	$1.80\ 10^{-50}$	$1.66\ 10^{53}$	$6.64\ 10^{-65}$	$4.52\ 10^{67}$	$8.05\ 10^{-58}$	$2.23\ 10^{66}$
Ag-108m	$1.41\ 10^{-5}$	$2.13\ 10^{8}$	$1.50\ 10^{-5}$	$2.00\ 10^{8}$	$3.76\ 10^{-13}$	$4.77\ 10^{21}$
Sb-125	$1.98\ 10^{-22}$	$1.51\ 10^{25}$	$8.08\ 10^{-28}$	$3.71\ 10^{30}$	$1.52\ 10^{-30}$	$1.18\ 10^{39}$
Sn-126	$5.35\ 10^{-6}$	$5.60\ 10^{8}$	$8.33\ 10^{-6}$	$3.60\ 10^{8}$	$3.20\ 10^{-13}$	$5.59\ 10^{21}$
I-129	$1.30\ 10^{-5}$	$2.30\ 10^{8}$	$1.12\ 10^{-4}$	$2.69\ 10^{7}$	$4.10\ 10^{-150}$	$2.03\ 10^{155}$
Ba-133	$1.90\ 10^{-10}$	$1.58\ 10^{13}$	$8.22\ 10^{-12}$	$3.65\ 10^{14}$	$2.04\ 10^{-21}$	$8.76\ 10^{29}$
Cs-134	$2.50\ 10^{-27}$	$1.20\ 10^{30}$	$1.92\ 10^{-34}$	$1.56\ 10^{37}$	$3.94\ 10^{-34}$	$4.55\ 10^{42}$
Cs-137	$2.18\ 10^{-7}$	$1.38\ 10^{10}$	$1.23\ 10^{-7}$	$2.44\ 10^{10}$	$1.73\ 10^{-14}$	$1.03\ 10^{23}$
Pm-147	$7.47\ 10^{-27}$	$4.02\ 10^{29}$	$5.81\ 10^{-32}$	$5.16\ 10^{34}$	$8.85\ 10^{-61}$	$2.02\ 10^{69}$
Eu-152	$6.06\ 10^{-9}$	$4.95\ 10^{11}$	$5.41\ 10^{-10}$	$5.54\ 10^{12}$	$2.15\ 10^{-14}$	$8.35\ 10^{22}$
Eu-154	$8.02\ 10^{-11}$	$3.74\ 10^{13}$	$1.64\ 10^{-12}$	$1.83\ 10^{15}$	$3.72\ 10^{-16}$	$4.82\ 10^{24}$
Eu-155	$1.12\ 10^{-16}$	$2.67\ 10^{19}$	$9.02\ 10^{-20}$	$3.33\ 10^{22}$	$1.99\ 10^{-45}$	$9.02\ 10^{53}$
Pb-210	$2.19\ 10^{-7}$	$1.37\ 10^{10}$	$1.97\ 10^{-7}$	$1.52\ 10^{10}$	$1.86\ 10^{-19}$	$9.63\ 10^{27}$

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
Ra-226**	$1.62\ 10^{-11}$	$1.85\ 10^{14}$	$1.49\ 10^{-11}$	$2.01\ 10^{14}$	$9.81\ 10^{-6}$	$1.83\ 10^{14}$
Ra-226***	$9.87\ 10^{-13}$	$3.04\ 10^{15}$	$7.39\ 10^{-15}$	$4.06\ 10^{17}$	$1.84\ 10^{-15}$	$9.72\ 10^{23}$
Ra-228	$6.63\ 10^{-8}$	$4.53\ 10^{10}$	$2.37\ 10^{-8}$	$1.26\ 10^{11}$	$3.65\ 10^{-18}$	$4.91\ 10^{26}$
Ac-227	$4.85\ 10^{-6}$	$6.18\ 10^{8}$	$8.17\ 10^{-6}$	$3.67\ 10^{8}$	$1.99\ 10^{-13}$	$9.01\ 10^{21}$
Th-229	$8.55\ 10^{-6}$	$3.51\ 10^{8}$	$4.33\ 10^{-5}$	$6.93\ 10^{7}$	$2.25\ 10^{-28}$	$6.17\ 10^{36}$
Th-230	$3.24\ 10^{-5}$	$9.26\ 10^{7}$	$4.19\ 10^{-5}$	$7.16\ 10^{7}$	$1.97\ 10^{-10}$	$7.29\ 10^{18}$
Th-232	$4.25\ 10^{-5}$	$7.06\ 10^{7}$	$1.61\ 10^{-4}$	$1.86\ 10^{7}$	$8.83\ 10^{-17}$	$4.21\ 10^{24}$
Pa-231	$1.76\ 10^{-7}$	$1.71\ 10^{10}$	$3.27\ 10^{-7}$	$9.17\ 10^{9}$	$6.57\ 10^{-34}$	$2.73\ 10^{42}$
U-232	$2.11\ 10^{-7}$	$1.42\ 10^{10}$	$5.57\ 10^{-7}$	$5.39\ 10^{9}$	$9.93\ 10^{-16}$	$6.31\ 10^{23}$
U-233	$1.36\ 10^{-7}$	$2.20\ 10^{10}$	$3.88\ 10^{-7}$	$7.74\ 10^{9}$	$3.54\ 10^{-37}$	$3.62\ 10^{44}$
U-234	$1.66\ 10^{-6}$	$1.81\ 10^{9}$	$2.63\ 10^{-6}$	$1.14\ 10^{9}$	$3.18\ 10^{-23}$	$3.09\ 10^{30}$
U-235	$1.28\ 10^{-7}$	$2.34\ 10^{10}$	$3.67\ 10^{-7}$	$8.17\ 10^{9}$	$1.83\ 10^{-18}$	$9.81\ 10^{26}$
U-236	$3.76\ 10^{-7}$	$7.99\ 10^{9}$	$6.59\ 10^{-7}$	$4.55\ 10^{9}$	$1.01\ 10^{-16}$	$5.04\ 10^{24}$
U-238	$2.92\ 10^{-6}$	$1.03\ 10^{9}$	$5.32\ 10^{-6}$	$5.64\ 10^{8}$	$2.12\ 10^{-31}$	$4.26\ 10^{37}$
Np-237	$2.53\ 10^{-7}$	$1.18\ 10^{10}$	$3.18\ 10^{-7}$	$9.43\ 10^{9}$	$1.31\ 10^{-43}$	$1.36\ 10^{52}$
Pu-238	$8.99\ 10^{-7}$	$3.34\ 10^{9}$	$1.67\ 10^{-6}$	$1.79\ 10^{9}$	$2.50\ 10^{-25}$	$7.18\ 10^{33}$
Pu-239	$8.88\ 10^{-7}$	$3.38\ 10^{9}$	$1.65\ 10^{-6}$	$1.82\ 10^{9}$	$6.44\ 10^{-42}$	$2.78\ 10^{50}$
Pu-240	$2.39\ 10^{-8}$	$1.26\ 10^{11}$	$4.50\ 10^{-8}$	$6.67\ 10^{10}$	$5.95\ 10^{-39}$	$3.01\ 10^{47}$
Pu-241	$8.44\ 10^{-7}$	$3.56\ 10^{9}$	$1.59\ 10^{-6}$	$1.89\ 10^{9}$	$8.26\ 10^{-24}$	$2.17\ 10^{32}$
Pu-242	$6.93\ 10^{-7}$	$4.33\ 10^{9}$	$1.30\ 10^{-6}$	$2.30\ 10^{9}$	$3.75\ 10^{-37}$	$4.78\ 10^{45}$
Am-241	$4.66\ 10^{-8}$	$6.44\ 10^{10}$	$2.13\ 10^{-8}$	$1.41\ 10^{11}$	$7.37\ 10^{-25}$	$2.43\ 10^{33}$
Cm-243	$3.84\ 10^{-9}$	$7.82\ 10^{11}$	$4.95\ 10^{-9}$	$6.06\ 10^{11}$	$9.42\ 10^{-54}$	$1.90\ 10^{62}$
Cm-244	$3.53\ 10^{-10}$	$8.51\ 10^{12}$	$2.81\ 10^{-11}$	$1.07\ 10^{14}$	$2.95\ 10^{-2}$	$6.06\ 10^{10}$

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)

* Where dose is effectively zero the radiological capacity is infinite, marked here as nd (not determined).

** Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface

*** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

Table 24 Radionuclides where radiological capacity is more limiting than the tonnage

Radionuclide	Radiological capacity (MBq)	Scenario
I-129	$4.17\ 10^{4}$	Well at boundary (All pathways)
Np-237	$4.52\ 10^{5}$	Well at boundary (All pathways)
Cl-36	$1.48\ 10^{6}$	Well at boundary (All pathways)
U-235	$4.92\ 10^{6}$	Well at boundary (All pathways)
U-234	$6.41\ 10^{6}$	Well at boundary (All pathways)
Pa-231	$1.86\ 10^{7}$	Small holding 200 years
U-238	$2.53\ 10^{7}$	Well at boundary (All pathways)
U-233	$3.13\ 10^{7}$	Well at boundary (All pathways)
Th-230	$6.93\ 10^{7}$	Small holding 200 years
Th-232	$7.16\ 10^{7}$	Small holding 200 years

Table 25 ENRMF Radiological capacity and constraint

Radionuclide	Radiological Capacity (TBq)	Scenario	Constraint*
H-3	$4.12\ 10^{3}$	Recreational (0 years)	
C-14	$1.20\ 10^{2}$	Recreational (0 years)	
Cl-36	1.48	Well at boundary (All pathways)	Limiting capacity
Fe-55	$5.83\ 10^{13}$	Excavator (Borehole) 60 years	
Co-60	$3.83\ 10^{5}$	Excavator (Borehole) 60 years	
Ni-63	$2.78\ 10^{5}$	Small holding 200 years	
Sr-90	$1.24\ 10^{3}$	Small holding 200 years	
Nb-94	$1.44\ 10^{2}$	Small holding 200 years	
Tc-99	$9.07\ 10^{1}$	Small holding 200 years	
Ru-106	$5.29\ 10^{16}$	Recreational (0 years)	
Ag-108m	$2.00\ 10^{2}$	Small holding 200 years	
Sb-125	$3.33\ 10^{9}$	Excavator (Borehole) 60 years	
Sn-126	$2.20\ 10^{2}$	Well at boundary (All pathways)	
I-129	$4.17\ 10^{-2}$	Well at boundary (All pathways)	Limiting capacity
Ba-133	$6.11\ 10^{4}$	Excavator (Borehole) 60 years	
Cs-134	$1.37\ 10^{11}$	Excavator (Borehole) 60 years	
Cs-137	$2.70\ 10^{3}$	Excavator (Borehole) 60 years	
Pm-147	$4.74\ 10^{13}$	Excavator (Borehole) 60 years	
Eu-152	$7.16\ 10^{3}$	Excavator (Borehole) 60 years	
Eu-154	$3.81\ 10^{4}$	Excavator (Borehole) 60 years	
Eu-155	$7.91\ 10^{7}$	Excavator (Borehole) 60 years	
Pb-210	$4.44\ 10^{3}$	Well at boundary (All pathways) POA	
Ra-226**	$1.56\ 10^{2}$	Excavator (Borehole) 60 years	
Ra-226***	2.51	Residential 150 years	
Ra-228	$1.80\ 10^{5}$	Excavator (Borehole) 60 years	
Ac-227	$1.02\ 10^{3}$	Excavator (Borehole) 60 years	
Th-229	$2.98\ 10^{2}$	Excavator (Borehole) 60 years	
Th-230	$6.93\ 10^{1}$	Small holding 200 years	Limiting capacity
Th-232	$7.16\ 10^{1}$	Small holding 200 years	Limiting capacity
Pa-231	$1.86\ 10^{1}$	Small holding 200 years	Limiting capacity
U-232	$4.25\ 10^{3}$	Excavator (Borehole) 60 years	
U-233	$3.13\ 10^{1}$	Well at boundary (All pathways)	Limiting capacity
U-234	6.41	Well at boundary (All pathways)	Limiting capacity
U-235	4.92	Well at boundary (All pathways)	Limiting capacity
U-236	$1.44\ 10^{2}$	Well at boundary (All pathways)	
U-238	$2.53\ 10^{1}$	Well at boundary (All pathways)	Limiting capacity
Np-237	$4.52\ 10^{-1}$	Well at boundary (All pathways)	Limiting capacity
Pu-238	$1.40\ 10^{3}$	Excavator (Borehole) 60 years	
Pu-239	$8.01\ 10^{2}$	Excavator (Borehole) 60 years	
Pu-240	$8.05\ 10^{2}$	Excavator (Borehole) 60 years	

Radionuclide	Radiological Capacity (TBq)	Scenario	Constraint*
Pu-241	$3.20\ 10^{4}$	Excavator (Borehole) 60 years	
Pu-242	$4.93\ 10^{2}$	Well at boundary (All pathways)	
Am-241	$1.08\ 10^{3}$	Excavator (Borehole) 60 years	
Cm-243	$4.28\ 10^{3}$	Excavator (Borehole) 60 years	
Cm-244	$1.59\ 10^{4}$	Excavator (Borehole) 60 years	

*"Limiting capacity" identifies those radionuclides where the radiological capacity is less than inventory arising from disposing of 448,000 t of LLW at 200 Bq g^{-1}.

** Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface

*** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

7.4.3 Conditions for acceptance of LLW

310. Procedure LLW01 lists the conditions for acceptance (CFA) of LLW at the ENRMF that are part of the contract between the consignor and Augean. The conditions are in two parts: Part A being the "Specification" for the waste and Part B being the "Procedures" associated with the receipt and acceptance of the waste. Part A has four sections dealing with general requirements, radiological waste characteristics, hazardous waste and other conditions. Part B deals with the procedures that are applied. Those aspects that relate to the ESC are summarised below. The CFA is used in the contractual arrangements with consignors and is designed to provide information to Augean that will ensure that disposals at the ENRMF meet permit conditions. The decision process leading to receipt of waste at the ENRMF is detailed in Section 7.3.1.

311. The working procedures that apply to radioactive waste accepted for disposal at the ENRMF include the following:

- A procedure for the pre-acceptance of waste by the central technical team (LLW02).
- A procedure for the receipt of waste, assay, quarantine, waste emplacement, coverage, record keeping and general LLW disposal operations (LLW03).
- A procedure for the quarantine of non-compliant waste packages received at the ENRMF (LLW04).
- A procedure for monitoring employee doses and instructions for measuring X-Ray and Gamma Radiation dose rates during acceptance of LLW waste at the ENRMF (LLW05).
- A procedure for handling asbestos bearing packages.
- Local rules in accordance with the Ionising Radiations Regulations
- A procedure for routine and periodic health surveillance monitoring for contamination and exposure. An emergency plan including response arrangements to identified fault scenarios including:
    i. Dropped load.
    ii. Contamination discovery.
    iii. Non-compliant load.
    iv. Dose above threshold discovery.
    v. Potentially contaminated person or wound.
- Procedures for environmental monitoring incorporated into the Monitoring and Action Plans (MAPs).
- A procedure outlining actions to be taken if consignments are unable to reach the site entrance in order to minimise risks to staff, the site and wider community (LLW06).

7.4.3.1 LLW01 Part A Conditions – Specification for Acceptance

General conditions

312. Consignors handling third party wastes to provide details of the organisation generating the waste and quality assurance to show the CFA have been applied at the point waste was produced.

313. Arrangements should be put in place by the consignor for the immediate return of non-compliant consignments delivered to the ENRMF.

Non-radiological characteristics

314. Non-radiological characteristics must be characterised for the waste to be assessed for acceptance.

315. ENRMF will not accept any of the following types of waste at the facility (definitions are from the Environmental Permitting Regulations):

- any waste in liquid form;
- waste which, in the conditions of landfill, is explosive, corrosive, oxidising, flammable or highly flammable;
- hospital and other clinical wastes which arise from medical or veterinary establishments and which are infectious;
- pressurised gas vessels; or,
- chemical substances arising from research and development or teaching activities, such as laboratory residues, which are not identified or which are new, and whose effects on man or on the environment are not known.

316. In addition, the ENRMF will not accept waste with any of the following characteristics:

- ion exchange materials (any material, whether synthetic or naturally occurring, that has the capability of interchanging ions from one substance to another by means of a reversible chemical or physical process);
- complexing agents (either chelating agents or monodentate organic ligands);
- waste which would otherwise present a danger to the facility operators during handling; or,
- packages where the outer surface of the package is chemically contaminated.

317. All hazardous wastes deposited except asbestos must meet the specified leaching criteria in accordance with the Environmental Permitting Regulations.

318. All hazardous wastes disposed of at the site must meet the organic acceptance criteria; 10% Loss on ignition or 6% Total organics carbon.

Radiological acceptance criteria

319. The specific activity of radionuclides in any LLW consignment to the ENRMF is not greater than 200 Bq g^{-1} (200 MBq t^{-1}).

320. No loose waste will be received at the ENRMF or handled at the facility. The maximum mass of each waste/package/pallet combination to be received at the ENRMF is normally limited to 2 t (arrangements can be made for heavier loads if necessary). The radioactive materials transport container used for transporting the waste to the ENRMF is the package that will be used for handling and final disposal. The container will be disposed directly to the final disposal position by careful offloading and will not be tipped. Packages should contain no void spaces and not be over-packed. Pallets will not be returned. Large surface contaminated objects or large items must be fully wrapped and sealed.

321. The consignor needs to characterise the radionuclides in each package using good practice methods and provide details of quality assurance arrangements. The characterisation must be representative of the contents of the packages and not averaged over more than 10 t. Detection limits must be lower than Basic Safety Standards (BSS) exemption levels (European Commission, 2014). The activity of the radionuclides indicated in Table 2 where these are present at levels above the limit of detection must be reported. "Other radionuclides" need to be identified by name and activity, where reasonably practicable.

322. The total activity for the LLW in the package is the total activity of the radionuclides identified in column 1 of Table 1. Where the radionuclide is shown to have daughters in secular equilibrium (column 3), only the head of the chain should be reported. Where the activity of a daughter that is listed in column 1 (i.e. Pb-210 or Ra-228) exceeds the parent, the excess (i.e. the unsupported activity) of that daughter should also be reported. The risk assessments which underpin the ESC assume that the listed daughters always exist and appropriate dose conversion factors take this into account.

323. Radionuclides of less than one year half-life are not normally included in the "Other radionuclides" category. However, if such nuclides are present in significant quantities (>5 MBq t^{-1} or a high percentage relative to the overall activity content) this must be reported.

324. The specific activity for radionuclides in the consignment, shall be such that the waste is defined as low level or very low level radioactive waste in accordance with current policy, except where wastes of less than a relevant exemption or exclusion order are mixed in with the LLW/VLLW as an inevitable result of the production such that separation is not reasonably practicable.

325. The sum of fractions of the radionuclides in the waste added to the sum of fractions of radionuclides already disposed of in the ENRMF is less than unity,

326. The consignor shall ensure that external non-fixed contamination levels on waste packages is as low as reasonably practicable throughout the process, complies with transport regulations and not more than 4 Bq cm^2 beta/gamma and 0.4 Bq cm^2 alpha averaged over an area of 300 cm^2. The consignment is to be accompanied by monitoring certificates demonstrating compliance with this requirement.

327. External dose rates from packages are to be as low as reasonably practicable, in accordance with the transport regulations and will not exceed 0.01 mSv hr^{-1} at 1 m from the waste package on all sides. Monitoring certificates are required to demonstrate compliance.

328. It is not acceptable to purposely dilute waste or add shielding materials for the sole purpose of achieving compliance with these CFA.

329. Packages should comply with the requirements of the current transport regulations, all the way through to the “as-disposed” condition. Additional shielding should not be used to ensure compliance.

Other conditions

330. Waste characterisation shall be on a package by package basis unless a case can be made that characterisation of a waste stream of several packages can be justified for some or all determinants.

331. Waste to be received at the ENRMF will be provided with a full description including:

- Source and origin of the waste;
- The process producing the waste;
- The composition of the waste and an assessment against relevant CFA values (including activity in consignment, mass of consignment and specific activity of consignment);
- The appearance of the waste and a physical description;
- A description of any non-radiological hazardous properties/classifications ;
- The mass of each package and the waste mass in each package;
- Unique identification labelling of each waste package as required under the transport regulations;
- An estimate of the void space in the package, where relevant;
- Details of any pre-conditioning/treatment of the wastes that has been utilised; and,
- Information relating to the safe transport of the waste as required under the transport regulations and details of the container/package to be used.

7.4.3.2 LLW01 Part B – Acceptance Procedures

332. All wastes must arise in the UK and the consigning site must have an appropriate transfer authorisation issued under EPR 2010. As part of the pre-acceptance process applied by Augean, details of the methodology by which the waste was produced and characterised, the justification for the methodology and BAT reports, the quality assurance arrangements, container specifications including intermediate bulk containers (for waste exempt or excepted under radioactive materials transport regulations) and wrapping of large objects, the waste description and the results are required. Samples used in waste characterisation should be retained for one year after waste is received at the ENRMF and be available to Augean if requested. Pallet design is specified by Augean. Waste can only be shipped by the consignor once approval in writing is obtained from Augean, this will detail date for delivery and transport routing. Waste is to be transported by a carrier approved as competent by the consignor.

333. The pre-acceptance information supplied by the consignor is reviewed by the central technical assessment team (Procedure LLW02) and a decision taken in principle whether to approve or decline the consignment.

334. Wastes arriving at the landfill will be subject to the following on site verification:

- The shipment will be checked while still on the vehicle against the pre-notified characterisation information for consistency and correctness.
- The external dose rate at 1 metre will be checked.
- The packages will be visually checked for integrity.
- The transport documentation will be checked for compliance with the transport regulations.
- The characterisation documentation will be checked to ensure the waste has been pre-accepted and is compliant.
- Receipt records will be generated.
- The waste packages will not be opened or sampled at the landfill in order to minimise unnecessary exposure.

7.4.4 Radioactive waste disposal proposed permit conditions

335. A permit variation is sought to allow receipt and disposal of low level radioactive waste to the ENRMF landfill covering phases 4B to 11.

336. Radioactive waste consignments will be limited to a maximum specific activity of 200 Bq g^{-1}. The wastes will otherwise be compliant with the non-radioactive properties specified in the CFA (i.e. the proposal is for the disposal of radioactive wastes that would be classified as inert, non-hazardous or hazardous in terms of their content of non-radioactive materials). The radioactive waste disposals would not be segregated from other, non-radioactive wastes disposed in the ENRMF.

337. The application for a variation proposes changes to Table 1 of the current permit which lists 43 radionuclides and provides an absolute disposal limit in GBq for each. A replacement table is proposed using the same radionuclides (plus Ra-228) with new values inserted based on the assessments reported in the ESC (see Table 25, columns 1 and 2 will form a revised Table). It is also proposed that a condition of any new permit will require the operator of the disposal site to calculate, for each radionuclide or group of radionuclides listed, the ratio of the radionuclide-specific activity of the radioactive waste disposed of at the ENRMF, to the relevant value in the new table. It will be a condition of the new permit that the sum of these ratios shall be less than 1. There is a revised permit template and Schedule 3 of the new template includes Table 3.1 which is reproduced as Table 26 below. The table includes a column containing a “Relevant value (TBq)” for each radionuclide and this is the radiological capacity referred to throughout this ESC.

338. It is proposed that the limit on the maximum specific activity applies to a consignment of up to 10 t.

Table 26 Suggested Schedule 3 – Disposals of radioactive waste and monitoring

Table 3.1 Disposal by burial on the premises				
Waste type	Disposal route	Sum of fractions limits		
		Radionuclide or group of nuclides	Relevant value (TBq)	Maximum total volume
Solid waste with a maximum total activity concentration of 200 Bq/g	Burial on the premises in cells 4B, 5A, 5B, 6, 7, 8, 9, 10 and 11 of the East Northants Resource Management Facility.	H-3	$4.12\ 10^{3}$	Not specified
		C-14	$1.20\ 10^{2}$	
		Cl-36	1.48	
		Fe-55	$5.83\ 10^{13}$	
		Co-60	$3.83\ 10^{5}$	
		Ni-63	$2.78\ 10^{5}$	
		Sr-90	$1.24\ 10^{3}$	
		Nb-94	$1.44\ 10^{2}$	
		Tc-99	$9.07\ 10^{1}$	
		Ru-106	$5.29\ 10^{16}$	
		Ag-108m	$2.00\ 10^{2}$	
		Sb-125	$3.33\ 10^{9}$	
		Sn-126	$2.20\ 10^{2}$	
		I-129	$4.17\ 10^{-2}$	
		Ba-133	$6.11\ 10^{4}$	
		Cs-134	$1.37\ 10^{11}$	
		Cs-137	$2.70\ 10^{3}$	
		Pm-147	$4.74\ 10^{13}$	
		Eu-152	$7.16\ 10^{3}$	
		Eu-154	$3.81\ 10^{4}$	
		Eu-155	$7.91\ 10^{7}$	
		Pb-210***	$4.44\ 10^{3}$	
		Ra-226*	$1.56\ 10^{2}$	
		Ra-226**	2.51	
		Ra-228***	$1.80\ 10^{5}$	
		Ac-227	$1.02\ 10^{3}$	
		Th-229	$2.98\ 10^{2}$	
		Th-230	$6.93\ 10^{1}$	
		Th-232	$7.16\ 10^{1}$	
		Pa-231	$1.86\ 10^{1}$	
		U-232	$4.25\ 10^{3}$	
		U-233	$3.13\ 10^{1}$	
		U-234	6.41	
		U-235	4.92	
		U-236	$1.44\ 10^{2}$	
		U-238	$2.53\ 10^{1}$	
		Np-237	$4.52\ 10^{-1}$	

Table 3.1 Disposal by burial on the premises				
Waste type	Disposal route	Sum of fractions limits		
		Radionuclide or group of nuclides	Relevant value (TBq)	Maximum total volume
		Pu-238	$1.40\ 10^{3}$	
		Pu-239	$8.01\ 10^{2}$	
		Pu-240	$8.05\ 10^{2}$	
		Pu-241	$3.20\ 10^{4}$	
		Pu-242	$4.93\ 10^{2}$	
		Am-241	$1.08\ 10^{3}$	
		Cm-243	$4.28\ 10^{3}$	
		Cm-244	$1.59\ 10^{4}$	
		Any other radionuclide	$4.17\ 10^{4}$	

* Assuming that wastes containing significant activity concentrations of Ra-226 are 5 m below the restored surface

** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

*** Only applies to activity that is not supported by the parent

339. The minimum depth of non-radioactive waste or material covering LLW and the constraining time periods for disposal or cover to be in place remain the same as in the current permit at 0.3 m (metre) and 8 hours, respectively. Constraints are suggested on the placement of waste in a landfill cell, placing non-radioactive waste to a specified depth at the base (2 m), distance from sides (2 m) and top (1 m) of a cell. An additional limitation is proposed for wastes containing a significant activity concentration of Ra-226 (>5 Bq g^{-1}) with a requirement to bury these wastes at least 5 m below the restored surface of the site.

7.5 Monitoring {R14}

340. The NS-GRA outlines the requirement for the operator to undertake a monitoring programme to support the environmental safety case (Requirement 14):

> *"In support of the environmental safety case, the developer/operator of a disposal facility for solid radioactive waste should carry out a programme to monitor for changes caused by construction, operation and closure of the facility.*
>
> *The developer/operator should establish a reasoned and proportionate approach to a programme for monitoring the site and facility. This monitoring will provide data during the period of authorisation to ensure that the facility is operating within the parameters set out in the environmental safety case. However, the monitoring must not itself compromise the environmental safety of the facility.*
>
> *(Environment Agencies, 2009), para 6.4.31 and 6.4.32."*

341. There are two main reasons for a monitoring programme at the site:

- Demonstration of compliance with stated regulatory requirements; and,
- Reassurance of stakeholders that the ENRMF is safe and being managed appropriately.

342. Augean currently operates a monitoring programme that meets the regulatory requirements specified in the Permit. The variation to the Permit does not lead to any required change and hence Augean propose to continue with the same monitoring programme and reporting arrangements.

7.5.1 Existing monitoring programme

343. The site has operated as a landfill site since 2002 and Augean have presented available data relating to the site to the Planning Inspector and also undertaken a detailed Environmental Assessment (Augean, 2012a) of the Western Extension. This substantial body of information provides details on geological, physical and chemical parameters which are relevant to environmental safety and which might change as a result of construction and waste emplacement (for example groundwater properties such as pressures, flows and chemical composition).

344. Prior to disposal of LLW in December 2011, measurements of pre-existing radioactivity were also undertaken. Baseline data were collected and included samples of groundwater from existing boreholes at the edge of the site and leachate taken from capped waste cells. The results of the baseline survey are presented in Appendix B.

345. The Environment Agency Permit number CD8503 includes conditions relating to monitoring the environment around the disposal site (Table 27). The Permit also requires a report to the Environment Agency reviewing the monitoring results and providing a comparison with the assessments submitted in support of the permit application. Such a report has been prepared recently (Galson Sciences Ltd, 2014) looking at the 2012 and 2013 monitoring results and comparing these with the previous radiological assessments for disposal of LLW at the ENRMF (Augean, 2009a).

346. The monitoring programme implemented by Augean encompasses these requirements (Table 27) and also provides information on specific radionuclides. Independent sample analyses have also been undertaken by the Environment Agency with a data report published recently (LGC Ltd, 2014) on 20 samples taken in February 2014.

347. The routine monitoring data are published by Augean on a regular basis (http://www.augeanplc.com/enrmf) and include measurements relating to groundwater, air quality, dust and asbestos. The website includes a section on measurements of radioactivity at the ENRMF and monitoring data are updated twice yearly for groundwater, dust, surface soils and site perimeter dose rates (see Appendix L).

Table 27 Monitoring specification in Permit CD8503

Sample type	Location	Frequency	Analysis
Groundwater	Boreholes K02a, K03, K04, K05, K06, K07, K08, K09, K14	Bi-annual	Gamma spec, total alpha and beta, tritium
Surface water	Surface water collected points KCSWNWPD, KCSWMP1, KCSWLAG and wheel wash run off.	Bi-annual	Gamma spec, total alpha and beta, tritium
Landfill gas	Environmental raw gas input point KCFLAINL	Bi-annual	Radon
		Bi-annual	Tritium
Dust	Downwind air sampler point KCDD01, Upwind air sampler KCDD02	Quarterly	Gamma spec, total alpha and beta.
Surface Soils	Surface soil locations as boundary to site KCSOIL01, KCSOIL02, KCSOIL03, KCSOIL04	Annual	Gamma spec, total alpha and beta.
Leachate	All leachate sumps and monitoring points in each phase used for LLW disposal	Annual bulked per cell	Gamma spec, total alpha and beta, tritium
	Leachate transferred off-site for disposal	Quarterly bulked per transfer	Gamma spec, total alpha and beta, tritium
Site perimeter dose rate	5 perimeter locations adjacent to monitoring points K03, KCSOIL01, KCSOIL02, KCSOIL03, K13	Quarterly	Gamma dose rate at 1m above ground level using continuous measurement thermoluminescent dosimeter (measurement of TLD by an approved dosimetry service) or other technique as otherwise agreed by the Agency.

348. The monitoring results for H-3 in groundwater taken from boreholes at the ENRMF site boundary, including both background data and monitoring data, indicate that no significant levels of H-3 have been detected before or since LLW has been accepted at the site. H-3 is a good indicator as it migrates relatively quickly and would be the first radionuclide to be found if barriers to migration were not behaving as expected.

349. The radionuclides routinely monitored by Augean are shown in Table 28 with the detection limits.

Table 28 Monitoring programme radionuclides and detection limits

Radionuclide	Decay Chain	Typical detection limit
Tritium		4 Bq l^{-1}
K-40		0.003 Bq g^{-1}
Co-60		0.001 Bq g^{-1}
Cs-137		0.001 Bq g^{-1}
Ac-228	Th-232	0.001 Bq g^{-1}
Ra-224	Th-232	0.003 Bq g^{-1}
Pb-212	Th-232	0.001 Bq g^{-1}
Th-234	U-238	0.003 Bq g^{-1}
Ra-226	U-238	0.003 Bq g^{-1}
Pb-214	U-238	0.001 Bq g^{-1}
Pb-210	U-238	0.002 Bq g^{-1}
U-235	U-235	0.001 Bq g^{-1}
Am-241		0.001 Bq g^{-1}

350. Samples taken by the Environment Agency were analysed for the following (detection limits in parenthesise);

- Gross alpha (about 0.001 Bq g^{-1}) and gross beta (not specified)
- Tritium (4 Bq l^{-1})
- Radionuclides through gamma spectrometry:
  - K-40 , Co-60, Nb-95, Ru-106, Ag-110m, Sb-125, Eu-152, Eu-154, Eu-155, Np-237 and Am-241 (0.001 Bq g^{-1})
  - Cs-134, Cs-137 (1 10^{-4} Bq g^{-1})
  - Ac-228, Ra-224, Pb-212, Bi-212, Tl-208, Th-234, Pa-234m, Pb-214, Bi-214, Pb-210, U-235, Th-231, Pa-231, Th-227, Ra-223 (0.01 Bq g^{-1})
  - Ra-226 (2 10^{-5} Bq g^{-1})
- Alpha spectrometry: Th-228, Th-230, Th-232, Pu-238, Pu-239/240, U-234, U-235 and U-238 (about 1 10^{-5} Bq g^{-1})

7.5.2 Reassurance

351. The monitoring results are made available for public scrutiny and are published through the company website (http://www.augeanplc.com/enrmf). This includes a commentary to provide a context for the monitoring results and help with their interpretation.

352. The independent analysis of samples from the site by the Environment Agency provides a check on the validity of the monitoring work undertaken by Augean.

353. Additional monitoring will also be undertaken prior to work starting on the new waste cells to ensure the development has no impact on system performance. This will be repeated once work on each cell is completed.

8 Summary of the Environmental Safety Case

354. This document is a new ESC for the disposal of LLW at the ENRMF and updates a document (Augean, 2009a) supporting an application that was the basis for Environment Agency Permit number CD8503. A permit variation is now sought to allow receipt and disposal of radioactive waste to the landfill extension (phases 6 to 11) in addition to the currently permitted cells (4B, 5A and 5B).

355. A revised submission has been made to the European Commission under Article 37 of the Euratom treaty based on this ESC.

356. The overall safety strategy for the disposal of LLW at the ENRMF involves both active (operational) management and the construction of passive barriers ensuring that disposed wastes will give rise to low impacts, within the dose and risk guidance levels laid down in the regulatory guidance, the NS-GRA (Environment Agencies, 2009). The ESC has considered all of the requirements in the NS-GRA and put forward calculations and arguments to demonstrate compliance. The sections of this document follow the structure of the NS-GRA (section titles indicate how document sections relate to the NS-GRA requirements). This final section draws together the main arguments that demonstrate the environmental safety of the ENRMF now and in the future.

357. The ESC takes into account changes to the design of the site as detailed in the site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013). Specifically, the ESC considers an increase in the number of waste cells and addresses the radiological impacts for the proposed variation to Permit CD8503. Other applications are being submitted to the Environment Agency in parallel concerning the disposal of hazardous waste to the landfill and hazardous waste treatment.

358. The ENRMF landfill site has been designed as a hazardous waste landfill and has been operating since 2002. The site has predominantly accepted hazardous waste since 2004 and first received LLW in December 2011. The design, construction and management of the ENRMF are undertaken in accordance with the requirements of Council Directive 1999/31/EC of 26 April 1999 on the landfill of waste (the Landfill Directive). Its performance in terms of general environmental impact was assessed during the Public Inquiry (The Planning Inspectorate, 2013) and the evidence included a HRA which has now been updated (Augean, 2014).

359. The ESC reiterates the strategic need for disposal of LLW at the ENRMF in terms of national policy and location. There have been no fundamental changes to the strategic need or legislation relating to the ENRMF since the planning inspectors report (The Planning Inspectorate, 2013).

360. The proposal to vary the LLW permit at the site will not change the annual tonnage or the physical capacity of the site, the current specific activity limit for LLW or the physical features that contributed to the hazardous waste landfill planning decision. The Development Consent Order reduces the tonnage of LLW that could be accepted for disposal to 448,000 t.

8.1 Protection against radiological hazards

361. The inventory requiring disposal is uncertain at this stage. Our approach is therefore to define the inventory that can be safely accepted and to put in place controls to ensure that this inventory is not exceeded. The ESC considers scenarios involving exposure to waste during normal operations, scenarios involving the expected site evolution and a full range of scenarios involving unexpected exposure resulting from the disposal of LLW. This range of scenarios ensures that for all reasonably foreseeable circumstances doses or risks remain below the relevant dose and risk guidance levels. The level of complexity that we have used in our assessments is considered to be proportionate and consistent with the level of detail in other safety cases including the previous ESC for the ENRMF.

362. The new ESC takes a similar approach to the previous application document (Augean, 2009a) using many of the same models that supported the radiological assessments underpinning the proposed disposal limits for LLW. The parameters used in the models have been updated as necessary to reflect any intervening changes in recommendations or the revised landfill design.

363. The assessment methodology that we have used draws heavily on methodologies developed under the sponsorship of the Environment Agency. We have used approaches developed by the Health Protection Agency (now PHE), the environment agencies (SNIFFER) and a screening methodology developed by the Environment Agency for operational releases. Where necessary we have also adopted approaches used in the LLWR ESC that have already been subject to detailed review by the Environment Agency.

364. The SNIFFER methodology and data have been used for a number of scenarios (SNIFFER, 2006) as previously. Model parameters have been adjusted to account for site specific inputs and have been adapted to take into account National Dose Assessment Working Group (NDAWG) recommendations concerning critical groups (NDAWG, 2013). The scenarios that use the SNIFFER approaches are shown in Table 29.

365. The assessment of worker exposures was been carried out by the HPA (Appendix H) and UKAEA (Appendix I and Appendix J) in support of the previous application. These assessments are based on the specific activity limit of 200 Bq g^{-1} which is unchanged and have not therefore been revisited. The assessment of dropped loads and the aircraft crash adopts the UKAEA methodology as used in the previous assessment. Additional models were adapted from the LLWR's ESC to consider the impact of radioactive particles and contaminated large items.

366. The ERICA assessment tool has been used to look at the impact of disposal at the ENRMF on non-human biota (ERICA, 2008). This assessment has been undertaken using a Tier 1 approach with the assessment tool developed as part of the ERICA project (Environmental Risk from Ionising Contaminants: Assessment and Management) and has used the version released in November 2014. The ERICA toolkit allows for consideration of three ecosystems: terrestrial, freshwater and marine. Only the first two have been considered for the ENRMF. Within these ecosystems, the ERICA Tool considers a range of wildlife groups. The assessment undertaken for non-human biota shows that the controls on the waste inventory, which are aimed at protecting the public, do not represent a risk to local biota.

367. The groundwater pathways have now been assessed using a model implemented specifically for the ENRMF site and environs. The model was developed using the GoldSim software, which was used because it provides a flexible modelling framework and the effects of decay and ingrowth can easily be accounted for. Where appropriate, input data have been used that are consistent with the HRA (Augean, 2014) and the previous assessment of radiological impacts (Augean, 2009a). Data have been used from other sources where appropriate.

Table 29 Summary of modelling approaches

Scenario	Exposed group	Modelling approach
Period of Authorisation – expected to occur		
Direct exposure	Worker	HPA/UKAEA assessments
Leachate processing off-site	Treatment worker	Initial radiological assessment methodology (Environment Agency)
	Farming family	
	Angler	
Release to atmosphere	Member of public	SNIFFER
Release to groundwater	Member of public	GoldSim
Period of Authorisation – not certain to occur		
Leachate spillage	Farming family	SNIFFER
Dropped load	Worker	UKAEA methodology
Aircraft impact	Member of public	
Wound exposure	Worker	HPA assessment
After the Period of Authorisation – expected to occur		
Recreational user	Member of public	SNIFFER
Groundwater abstraction	Farming family	GoldSim
Wildlife exposure	Critical species	ERICA assessment tool
After the Period of Authorisation – not certain to occur		
Water abstraction at boundary	Farming family	GoldSim
Bathtubbing	Farming family	
Gas release and external	Site resident	SNIFFER
Borehole drilling	Worker	
Trial pit excavation	Worker	
Laboratory analyst	Worker	LLWR ESC
Excavation for housing	Worker/Resident	SNIFFER
Excavation for smallholder	Farming family	
Intrusion (items and particles)	Public and worker	LLWR ESC

368. The previous ESC considered a landfill capacity of about 1 10^6 t at a maximum specific activity of 200 Bq g^{-1}. The total activity that could be disposed was calculated to be 313 TBq and limits for individual radionuclides were specified. The current permit which was issued in response to an application based on the previous ESC allows for disposal of 17 TBq at a maximum specific activity of 200 Bq g^{-1} and sets limits for individual radionuclides (see Table 2). The recent site Development Consent Order places restrictions on the total LLW tonnage (448,000 t) that can be received at the site, specifies an annual limit for direct landfill disposal of 150,000 t y^{-1}

(LLW and hazardous waste combined) and a maximum specific activity of 200 Bq g^{-1}; these restrictions limit the total activity that can be disposed to 89.6 TBq.

369. The radiological assessments described in the ESC have been used to derive a limit for each radionuclide that will ensure the dose constraints and risk guidance levels are not exceeded in any of the assessed scenarios. The use of a sum of fractions approach based on these limits ensures that the disposed inventory will not result in impacts in excess of regulatory requirements. The following criteria have been used based on the NS-GRA (Environment Agency, 2012a) and Environment Agency Guidance (Environment Agency, 2012b). During the Period of Authorisation:

- Dose constraint for the public from a single source 0.3 mSv yr^{-1}; and,
- Site dose criterion for workers – 1 mSv yr^{-1}

After the end of management:

- 0.02 mSv yr^{-1} for events that are certain to occur; and,
- 3 mSv yr^{-1} for human intrusion.

From supplementary guidance implementing the requirements of the groundwater directive:

- 0.02 mSv yr^{-1} for groundwater pathways during the period of authorisation.

370. The radiological assessments of dose to the public from disposals of LLW to the ENRMF look at the behaviour of radionuclides in the landfill, consider ways that material can enter the local environment and have looked at the timescale over which this may occur. Particular attention has been given to groundwater and leachate. Assessments also take into account the future of the site once it has been closed examining different site uses and potential intrusion scenarios. The assessment approaches are cautious in nature and overestimate the doses that may occur, this leads to a radiological capacity that is also cautious. The radiological capacity that is proposed for use with the sum of fractions is given in Table 25 and shown as the proposed relevant values for Schedule 3 of a revised permit (Table 26). In many cases the limit of 89.6 TBq, based on the 448,000 t limit for LLW disposal at the ENRMF, is lower than the radiological capacity used in the sum of fractions. Since this disposal limit of 448,000t will also be applied to the LLW in the site, this means that the prospective dose from these radionuclides is effectively capped at a much lower dose than the dose criteria that have been applied.

371. The current inventory (June 2015), radiological capacity and fractions are presented below (Table 30). The sum of fractions for the inventory to June 2015 is 0.0024, representing use of a very small proportion of the available radiological capacity. Applying the same mix of radionuclides and their average specific activity with the remaining tonnage that can be accepted, the sum of fractions is 0.071 over the lifetime of the ENRMF.

372. The impact of uncertainty in estimated doses and risks has been considered and demonstrates that the ESC is robust in meeting all relevant dose and risk guidance levels.

Table 30 Inventory, radiological capacity and fraction of radiological capacity calculation

Radionuclide	June 2015 Disposal inventory (MBq)	Radiological capacity (MBq)	Fraction of radiological capacity used by disposal inventory
H-3	$2.38\ 10^{4}$	$4.12\ 10^{9}$	$5.77\ 10^{-6}$
C-14	$2.24\ 10^{3}$	$1.20\ 10^{8}$	$1.87\ 10^{-5}$
Cl-36	$3.10\ 10^{1}$	$1.48\ 10^{6}$	$2.10\ 10^{-5}$
Fe-55	$4.58\ 10^{2}$	$5.83\ 10^{19}$	$7.86\ 10^{-18}$
Co-60	$1.49\ 10^{3}$	$3.83\ 10^{11}$	$3.90\ 10^{-9}$
Ni-63	$1.46\ 10^{3}$	$2.78\ 10^{11}$	$5.28\ 10^{-9}$
Sr-90	$2.73\ 10^{3}$	$1.24\ 10^{9}$	$2.20\ 10^{-6}$
Nb-94	$2.76\ 10^{-1}$	$1.44\ 10^{8}$	$1.92\ 10^{-9}$
Tc-99	$1.71\ 10^{1}$	$9.07\ 10^{7}$	$1.88\ 10^{-7}$
Ru-106	$1.80\ 10^{-2}$	$5.29\ 10^{22}$	$3.41\ 10^{-25}$
Ag-108m	$3.75\ 10^{-1}$	$2.00\ 10^{8}$	$1.87\ 10^{-9}$
Sb-125	1.15	$3.33\ 10^{15}$	$3.43\ 10^{-16}$
Sn-126	0	$2.20\ 10^{8}$	0
I-129	1.80	$4.17\ 10^{4}$	$4.31\ 10^{-5}$
Ba-133	$2.11\ 10^{1}$	$6.11\ 10^{10}$	$3.46\ 10^{-10}$
Cs-134	2.23	$1.37\ 10^{17}$	$1.63\ 10^{-17}$
Cs-137	$1.59\ 10^{4}$	$2.70\ 10^{9}$	$5.89\ 10^{-6}$
Pm-147	5.08	$4.74\ 10^{19}$	$1.07\ 10^{-19}$
Eu-152	$8.77\ 10^{2}$	$7.16\ 10^{9}$	$1.22\ 10^{-7}$
Eu-154	$5.92\ 10^{1}$	$3.81\ 10^{10}$	$1.55\ 10^{-9}$
Eu-155	9.04	$7.91\ 10^{13}$	$1.14\ 10^{-13}$
Pb-210	$1.47\ 10^{4}$	$4.44\ 10^{9}$	$3.31\ 10^{-6}$
Ra-226	$1.86\ 10^{4}$	$1.56\ 10^{8}$	$1.20\ 10^{-4}$
Ra-228	0	$1.80\ 10^{11}$	0
Ac-227	4.43	$1.02\ 10^{9}$	$4.35\ 10^{-9}$
Th-229	0	$2.98\ 10^{8}$	0
Th-230	$1.42\ 10^{2}$	$6.93\ 10^{7}$	$2.05\ 10^{-6}$
Th-232	$3.98\ 10^{3}$	$7.16\ 10^{7}$	$5.55\ 10^{-5}$
Pa-231	4.05	$1.86\ 10^{7}$	$2.18\ 10^{-7}$
U-232	0	$4.25\ 10^{9}$	0
U-233	$2.70\ 10^{-2}$	$3.13\ 10^{7}$	$8.61\ 10^{-10}$
U-234	$1.60\ 10^{2}$	$6.41\ 10^{6}$	$2.50\ 10^{-5}$
U-235	6.58	$4.92\ 10^{6}$	$1.34\ 10^{-6}$
U-236	$4.64\ 10^{-1}$	$1.44\ 10^{8}$	$3.23\ 10^{-9}$
U-238	$3.18\ 10^{2}$	$2.53\ 10^{7}$	$1.26\ 10^{-5}$
Np-237	0	$4.52\ 10^{5}$	0
Pu-238	$6.92\ 10^{1}$	$1.40\ 10^{9}$	$4.94\ 10^{-8}$
Pu-239	$3.59\ 10^{2}$	$8.01\ 10^{8}$	$4.49\ 10^{-7}$
Pu-240	$5.17\ 10^{2}$	$8.05\ 10^{8}$	$6.42\ 10^{-7}$
Pu-241	$2.53\ 10^{3}$	$3.20\ 10^{10}$	$7.93\ 10^{-8}$

Radionuclide	June 2015 Disposal inventory (MBq)	Radiological capacity (MBq)	Fraction of radiological capacity used by disposal inventory
Pu-242	$4.35\ 10^{-1}$	$4.93\ 10^{8}$	$8.83\ 10^{-10}$
Am-241	$5.81\ 10^{2}$	$1.08\ 10^{9}$	$5.37\ 10^{-7}$
Cm-243	1.51	$4.28\ 10^{9}$	$3.52\ 10^{-10}$
Cm-244	$5.56\ 10^{1}$	$1.59\ 10^{10}$	$3.49\ 10^{-9}$
Any other radionuclide	$8.49\ 10^{1}$	$4.17\ 10^{4}$	$2.04\ 10^{-3}$
		Sum of fractions	$2.35\ 10^{-3}$

373. Environmental monitoring during the period of authorisation will check the integrity of barriers and safety plans. A site monitoring plan is in place to check the levels of radioactivity in groundwater, surface water, landfill gas, dust, surface soils and leachate. These media might become contaminated as the result of the migration of radionuclides from the site to the surrounding area. Samples are taken on a regular basis and an interpretative report is prepared for the Environment Agency, who also undertake an independent sampling programme. All these samples provide additional assurance that the site is performing as expected and can be used as the basis for dose assessments to confirm that impacts are low. Site perimeter dose rate measurements are also undertaken.

374. Monitoring will continue to the end of the period of authorisation (the period of management control). If any undue adverse impacts were to arise appropriate action will be agreed with the Environment Agency.

375. The Augean management culture and safety procedures ensure that wastes are transported and handled safely reducing the potential for dose impact to the workforce and the risk of accidents leading to unplanned impacts on the environment. The site management controls will ensure that the inventory is not exceeded. There are working procedures in place controlling LLW activities at the ENRMF (Section 5.2.5). The procedures cover prior agreement between the consignor and Augean for disposal, detail appropriate receipt procedures and keeping records of disposals, procedures for waste emplacement, monitoring worker exposure, environmental monitoring and emergency plans to deal with events such as dropped loads. These are all part of Augean's Integrated Management System.

8.2 Optimisation

376. The requirement for optimisation in relation to radiological risk may be considered at three levels.

- The design of the ENRMF is consistent with best practice and regulatory requirements for the disposal of hazardous wastes and may therefore be considered to be optimised.
- We have considered a number of specific ways in which the management and the design of the site may be enhanced to achieve an optimised solution for the disposal of radioactive wastes;

- Waste consignors are required to manage wastes in a manner consistent with BAT and must demonstrate that disposal to the ENRMF is an optimal solution and hence consistent with BAT. We note that this aspect is a matter for consignors.

377. The design features and arrangements provide an appropriate strategy to limit the environmental impacts arising from non-radioactive contaminants. The design satisfies the requirements set out in the Landfill Directive. In the context of the assumed timescales and approach to landfill risk assessment, these measures will also be effective in limiting the environmental impacts arising from radioactive contaminants. In this sense, the design of the facility may already be considered to have been optimised. As the design of the facility is already recognised as consistent with good practice for landfills and the hazards associated with the proposed disposals of radioactive waste are low, a detailed and systematic analysis of alternative design and management strategies for the facility has not been undertaken.

378. A number of specific considerations have led to enhancements to the operational or emplacement approach to ensure that performance for radioactive waste is optimised. These include:

- The use of waste packages, which reduce the probability of doses during operations, will also reduce leaching post-closure and increase the prospect of the waste being recognised as hazardous during future intrusion.
- The implementation of a limit on putrescible materials accepted at the ENRMF ensures that microbial activity is minimised and gaseous release from microbial action or the potential for fire is minimised.
- Augean places a constraint on the level of dust on the surface of waste packages to ensure this does not represent a hazard. Wastes placed in the landfill are also covered daily to prevent dust suspension and hence the risk of impacts via the inhalation pathway during the operational period. A check is also undertaken on dose measurements at 1 m above the surface of the covered LLW, to ensure exposure of less than 2 $\mu Sv\ hr^{-1}$. The depth of cover will be increased if necessary to ensure that this limit is not exceeded. These precautions will provide additional confidence that no specific protective measures are needed for workers at the site who are closest to the LLW and will provide additional confidence that anyone off site is also suitably protected.
- Operational constraints have been put in place to restrict the placement of waste in a landfill cell, placing non-radioactive waste to a specified depth at the base (2 m), distance from sides (2 m) and top (1 m) of a cell. This creates a barrier between the LLW and the side liner of a waste cell which will need to be located when the cell is capped. An additional limitation is proposed for wastes with significant radium contamination. Such wastes will be disposed at least 5 m below the restored surface of the site. This places radium below a reasonable intrusion depth and reduces the potential dose due to radon gas release from the landfill.

379. The profiling of the restored surface will encourage surface runoff, preventing the development of puddles and reducing infiltration. Areas of the site will also be developed as woodland and these areas will have a deeper soil layer over the cap. This will further reduce the chance of intrusion disturbing waste or the prospect of housing development at the site.

8.3 Protection against non-radiological hazards

380. The ENRMF is designed to take hazardous wastes and the HRA (Augean, 2014) for the site demonstrates that no unacceptable environmental impacts will arise. The existing landfill at the ENRMF is permitted under the Environmental Permitting Regulations and satisfies the requirements of the Landfill Directive for hazardous waste in terms of the management, engineering and monitoring of the site.

381. The wastes accepted at the site are largely hazardous due to harmful, toxic, carcinogenic, irritant or eco-toxic properties. No explosive, flammable, corrosive, oxidising or infectious wastes are accepted at the site. The IMS includes established procedures for safe handling and disposal of the hazardous wastes accepted at the site. These processes are similar to those for the handling of LLW and do not conflict with them.

382. The arrangements for construction design, waste acceptance, groundwater protection, landfill gas management, leachate management, landfill stabilisation, pollution prevention, nuisance prevention and quality assurance, construction quality assurance, maintenance, landfill capping, site restoration, operations, waste handling/placement, security, use of raw materials, secondary wastes, accident arrangements, monitoring, closure, aftercare and surrender are described in existing documentation for the landfill site.

8.4 Reliance on human action

383. The disposal facility is designed to minimise reliance on human action to maintain the safety case during the period of operation. During the post-closure Period of Authorisation (i.e. the period after which no further disposals are received and the disposal cells are capped, but during which the site Permit issued under EPR 2010 remains in force), leachate management will continue alongside monitoring to demonstrate that the overall system is continuing to limit entry of radionuclides to the accessible environment, consistent with the arguments in this ESC.

384. Following revocation of the site Permit (i.e. at the end of the period of authorisation), there is no continuing reliance on monitoring or any other active management or intervention measure to ensure the continuing safety of the overall system.

8.5 Openness and inclusivity

385. Following the decision of the Secretary of State to grant the Development Consent Order in July 2013 Augean has engaged with the Environment Agency in correspondence and at meetings to discuss the radiological proposals for the extension and to agree the approach taken by Augean for the ESC. Specifically meetings were held on the 11th November 2013 and the 10th June 2014 at which Augean set out the principles of their approach and the programme for the application.

386. The report by Jonathan Green on the ENRMF (The Planning Inspectorate, 2013) considered that the consultations Augean have undertaken covered all aspects of the proposed development including the disposal of LLW. The inspector concluded that the local community has had extensive engagement with Augean on this issue over several years, including public meetings, open days at the site, provision of written

information, the opportunity to make written submissions and engagement with the public inquiry. The inspector was satisfied that the consultation requirements of the national policy for LLW management had been met.

387. Augean has continued to engage with the local community through the KCLG and the TLG. This has involved annual open days, a twice yearly newsletter and maintenance of a register of stakeholders. The KCLG has been kept up to date with the programme for the application to vary the radiological Environmental Permit and is aware that the application is scheduled for the third quarter of 2015.

388. On submission of the application for the permit variation Augean will inform the local community representatives of the submission. Augean will also prepare a non-technical summary of the application proposals for circulation in the community. A site open day will be organised in October 2015 at which the community can discuss the application with Augean and the company's expert advisors. It is understood that the Environment Agency will take part in this event.

8.6 Conclusion

389. Overall, we consider that the measures set out in this ESC provide assurance that the proposed disposal of LLW will be managed appropriately and will give rise to radiological impacts well within relevant regulatory criteria.

390. The ESC will be subject to periodic review. It is suggested that this is undertaken every 5 years. However, should any new information arise that affects the assumptions supporting the ESC, or monitoring results indicate that the assessments could be challenged, a review would be initiated.

391. Continued disposal of LLW at the ENRMF would secure a cost-effective, regional LLW disposal solution for nuclear sites located in the south of the United Kingdom, which exceeds the required environmental standards. In accordance with national objectives for LLW management, it would help to ensure that disposal capacity at the LLWR is only used for wastes requiring a more highly engineered disposal solution.

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Appendix A. Glossary

In the context of this Glossary, the term ‘waste’ refers, in general, to radioactive waste unless otherwise specified.

absorbed dose. See *dose, absorbed.*

activation. The process of inducing *radioactivity.* Most commonly used to refer to the induction of *radioactivity* in moderators, coolants, and structural and shielding materials, caused by irradiation with neutrons.

activation product. A *radionuclide* produced by *activation*. Often used in distinction from *fission products*. For example, in decommissioning waste comprising structural materials from a *nuclear facility*, activation products might typically be found primarily within the matrix of the material, whereas *fission products* are more likely to be present in the form of *contamination* on surfaces.

activity. The quantity *A* for an amount of *radionuclide* in a given energy state at a given time. The SI unit of activity is the reciprocal second (s^{-1}), termed the Becquerel (Bq). Formerly expressed in curie (Ci), which is still sometimes used.

activity concentration. Of a material, the *activity* per unit mass or volume of the material in which the *radionuclides* are essentially uniformly distributed.

activity, specific. Of a *Waste Consignment* means the *Activity in the consignment* divided by the weight of the consignment. In the context of conditioned wastes, the weight of the consignment is the weight of the waste and immobilising material or grout. In accounting for *Activity* against these limits, the *Activity* of *Decay Products* shall be accounted for as listed in Column 1 of Table 1.

ALARP & ALARA. As low as reasonably practicable. As low as reasonably achievable. ALARP & ALARA describe approaches to optimisation. The optimisation principle states “in relation to any particular source within a practice, the magnitude of individual doses, the number of people exposed, and the likelihood of incurring exposures where these are not certain to be received should all be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account…” ALARA is incorporated in UK law via RSA 1993 (BSS) Direction 2000. ALARA & ALARP focus on impacts to people.

aquifer. A water bearing formation below the surface of the earth that can furnish an appreciable supply of water for a well or spring.

area, controlled. A defined area in which specific protection measures and safety provisions are or could be required for controlling *normal exposures* or preventing the spread of *contamination* during normal working conditions, and preventing or limiting the extent of *potential exposures*.

assessment. The process, and the result, of analysing systematically the hazards associated with *sources* and *practices*, and associated protection and safety measures, aimed at quantifying performance measures for comparison with criteria.

assessment, environmental (impact). An evaluation of radiological and nonradiological impacts of a proposed activity, where the performance measure is overall environmental

COMMERCIAL

Environmental Safety Case:

Disposal of Low Activity Low Level Radioactive Waste at East Northants Resource Management Facility

Final: ENE-154/001

Appendices A to G

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assessment. The process, and the result, of analysing systematically the hazards associated with *sources* and *practices*, and associated protection and safety measures, aimed at quantifying performance measures for comparison with criteria.

assessment, environmental (impact). An evaluation of radiological and nonradiological impacts of a proposed activity, where the performance measure is overall environmental impact, including radiological and other global measures of impact on safety and environment.

assessment, performance. An *assessment* of the performance of a system or subsystem and its implications for protection and safety at a planned or an authorized *facility*. This differs from *safety assessment* in that it can be applied to parts of a *facility*, and does not necessarily require assessment of radiological impacts.

assessment, risk. An *assessment* of the radiological *risks* associated with normal *operation* and potential accidents involving a *source* or *practice*. This will normally include *consequence assessment* and associated probabilities.

assessment, safety. An analysis to evaluate the performance of an overall system and its impact, where the performance measure is radiological impact or some other global measure of impact on safety. See also *assessment, performance.*

audit. A documented activity performed to determine by investigation, examination and evaluation of objective evidence the adequacy of, and adherence to, established procedures, instructions, specifications, codes, standards, administrative or operational programmes and other applicable documents, and the effectiveness of implementation.

authorization. The granting by a *regulatory body* or other governmental body of written permission for an *operator* to perform specified activities. Authorization could include, for example, a permit, licensing, certification and registration. See also *licence*.

background (radiation). The *dose,* dose rate or an observed measure related to the *dose* or dose rate, attributable to all *source*s other than the one(s) specified.

barrier. A physical obstruction that prevents or delays the movement of *radionuclide*s or other material between components in a system, for example a waste *repository*. In general, a barrier can be an engineered barrier which is constructed or a natural (or geological) barrier.

barrier, intrusion. The components of a *repository* designed to prevent inadvertent access to the *waste* by humans, animals and plants.

barriers, multiple. Two or more natural or engineered *barriers* used to isolate *radioactive waste* in, and prevent *radionuclide* migration from, a *repository*. See also *barrier.*

borehole. A cylindrical excavation, made by a drilling device. Boreholes are drilled during *site* investigation and testing and are also used for *waste* emplacement in repositories and *monitoring*.

Bq/g A Becquerel (abbreviated as Bq) is the International System (SI) unit for the activity of radioactive material. One Bq of radioactive material is that amount of material in which one atom is transformed or undergoes one disintegration every second. A Gram (abbreviated as g) is a unit of mass. A Becquerel per Gram (abbreviated Bq/g) is therefore a measure of the concentration of radioactivity in a material.

characterization, site. Detailed surface and subsurface investigations and activities at candidate *disposal* sites to obtain information to determine the suitability of the *site* for a *repository* and to evaluate the long term performance of a *repository* at the *site.*

characterization, waste. Determination of the physical, chemical and radiological properties of the *waste* to establish the need for further adjustment, *treatment*, *conditioning*, or its suitability for further handling, *processing*, *storage* or *disposal.*

clay. *Mineral*s that are essentially hydrated aluminium silicates or occasionally hydrated magnesium silicates, with sodium, calcium, potassium and magnesium cations. Also denotes a natural material with plastic properties which is essentially a composition of fine to very fine clay particles. Clays differ greatly mineralogically and chemically and consequently in their physical properties. Because of their large surface areas, most of them have good *sorption* characteristics.

clearance. Removal of *radioactive materials* or radioactive objects within authorized *practice*s from any further *regulatory control* by the *regulatory body.*

closure. Administrative and technical actions directed at a *repository* at the end of its operating lifetime — for example covering the disposed *waste* (for a *near surface repository*) or backfilling and/or sealing (for a *geological repository* and the passages leading to it) — and termination and completion of activities in any associated structures.

conductivity, hydraulic, *K*. Ratio of groundwater flow rate n to driving force d*h*/d*l* (the change of hydraulic head with distance) for viscous flow of a fluid in a porous medium. This is the so-called constant of proportionality *K* in Darcy's Law and depends on both the porous medium and the fluid properties. See also *permeability.*

consignment, a set of one or more waste packages not exceeding 10 tonnes.

container, waste. The vessel into which the *waste form* is placed for handling, transport, *storage* and/or eventual *disposal*; also the outer *barrier* protecting the *waste* from external intrusions. The waste container is a component of the *waste package*. See also *barrier*; *waste package*.

containment. Methods or physical structures designed to prevent the release of *radioactive substances*.

contamination. (1) *Radioactive substances* on surfaces, or within solids, liquids or gases (including the human body), where their presence is unintended or undesirable, (2) the presence of such substances in such places or (3) the process giving rise to their presence in such places.

control, institutional. Control of a *waste site* by an authority or institution designated under the laws of a country. This control may be active (*monitoring*, *surveillance* and remedial work) or passive (land use control) and may be a factor in the *design* of a *nuclear facility* (e.g. a *near surface repository*).

control, regulatory. Any form of control applied to facilities or activities by a *regulatory body* for reasons related to protection or safety.

criteria. Conditions on which a decision or judgement can be based. They may be qualitative or quantitative and should result from established principles and standards. See also *requirement*; *specifications*.

critical group. A group of members of the public which is reasonably homogeneous with respect to its *exposure* for a given radiation *source* and given *exposure pathway* and is typical of individuals receiving the highest *effective dose* or *equivalent dose* (as applicable) by the given *exposure pathway* from the given *source*. The same as a *representative person*.

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the *regulatory control*s from a *facility.* This does not apply to a *repository* or to certain *nuclear facilities* used for mining and *milling* of *radioactive materials,* for which *closure* is used.

decontamination. The complete or partial removal of *contamination* by a deliberate physical, chemical or biological process.

diffusion. The movement of atoms or molecules from a region of higher concentration of the diffusing species to regions of lower concentration, due to a concentration gradient.

discharge. A planned and controlled release of (usually gaseous or liquid) *radioactive material* to the environment.

disintegration per second. See also Bq/g. A disintegration is any nuclear transformation

disposal. Emplacement of *waste* in an appropriate *facility* without the intention of retrieval. Some countries use the term *disposal* to include *discharge*s of effluents to the environment.

distribution coefficient, K_d. The ratio of the amount of substance sorbed on a unit mass of dry solid to the concentration of the substance in a solution in contact with the solid, assuming equilibrium conditions. The SI units are: $m^3 kg^{-1}$.

dose. A measure of the energy deposited by radiation in a target. *Absorbed dose*, committed equivalent dose, committed effective dose, *effective dose, equivalent dose* or organ dose, depending on the context. All these quantities have the dimensions of energy divided by mass.

dose, absorbed, *D*. The fundamental dosimetric quantity *D.* The unit is J kg^{-1}, termed the gray (Gy).

dose constraint. A prospective and source related restriction on the individual dose from a source, which provides a basic level of protection for the most highly exposed individuals from a source and serves as an upper bound on the dose in optimization of protection for that source. The UK government has set a maximum dose constraint value of 0.3 mSv y^{-1} when determining applications for discharge authorization from a single new source.

dose, effective, *E*. A summation of the tissue *equivalent dose*s, each multiplied by the appropriate tissue weighting factor: The unit of effective dose is J kg^{-1}, with the special name Sievert (Sv). The committed effective dose is the effective dose that will be received by the person over their lifetime as a result of radionuclides taken into the body e.g. by ingestion or inhalation.

dose, equivalent, H_T. The radiation-weighted dose in a tissue or organ. This takes account of the different amounts of damage caused by different types of radiation eg alpha particles, gamma radiation. The unit of equivalent dose is J/kg, termed Sievert (Sv).

dose limit. See *limit, dose.* The value of the effective dose or the equivalent dose to individuals from planned exposure situations that shall not be exceeded. For the purposes of discharge authorizations, the UK has (since 1986) applied a dose limit of 1 mSv y^{-1} to members of the public from all man-made sources of radioactivity (other than from medical applications).

effluent. Gaseous or liquid *radioactive materials* which are discharged to the environment. See also *discharge, authorized*.

emanation. Generation of radioactive gas by the decay of a radioactive solid.

environmental impact statement. A set of documents recording the results of an evaluation of the physical, ecological, cultural and socioeconomic effects of a planned *facility* (e.g. a *repository*) or of a new technology.

exemption. The determination by a *regulatory body* that a *source* or *practice* need not be subject to some or all aspects of *regulatory control* on the basis that the *exposure* (including *potential exposure*) due to the *source* or *practice* is too small to warrant the application of those aspects. See also *level, clearance*.

exposure. The act or condition of being subject to irradiation. Exposure can either be external exposure due to *sources* outside the body or internal exposure due to *sources* inside the body.

exposure, normal. *Exposure* which is expected to occur under the normal operating conditions of a *facility* or activity, including possible minor mishaps that can be kept under control, i.e. during normal operation and anticipated operational occurrences.

exposure, potential. *Exposure* that is not expected to occur with certainty but that may result from an accident at a *source* or owing to an event or sequence of events of a probabilistic nature, including equipment failures and operating errors.

exposure pathway. A route by which radiation or *radionuclide*s can reach humans and cause *exposure*. An exposure pathway may be very simple, for example external *exposure* from airborne *radionuclide*s, or involve a more complex chain, for example internal *exposure* from drinking milk from cows that ate grass contaminated with deposited *radionuclides*.

fissile material. Uranium-233, uranium-235, plutonium-239, plutonium-241, or any combination of these *radionuclide*s. Excepted from this definition is: (a) *natural uranium* or *depleted uranium* which is unirradiated, (b) *natural uranium* or *depleted uranium* which has been irradiated in thermal reactors only.

fission product. A *radionuclide* produced by nuclear fission.

flow, unsaturated. The flow of water in unsaturated soil by capillary action and gravity.

fracture. A general term for any breaks in *rock* whether or not it causes displacement.

gradient, hydraulic. The change in total hydraulic head per unit distance of flow in a given direction.

groundwater. Water that is held in *rock*s and soil beneath the surface of the earth.

half-life, $T_{1/2}$. The time taken for the quantity of a specified material (e.g. a *radionuclide*) in a specified place to decrease by half as a result of any specified process or processes that follow similar exponential patterns to radioactive decay.

half-life, effective, *Teff*. The time taken for the *activity* of a *radionuclide* in a specified place to halve as a result of all relevant processes.

half-life, radioactive. For a *radionuclide*, the time required for the *activity* to decrease, by a radioactive decay process, by half.

Harwell. The UKAEA Harwell site in Oxfordshire is an ex-RAF WWII airbase that has been used since 1946 for nuclear research, mainly in support of civilian power generation. The site is now well advanced with decommissioning. The aim is to return the site to a delicensed status by 2025.

HV-VLLW. High volume very low level waste. A sub-category of LLW as defined in “Policy for the Long Term Management of Solid Low Level Radioactive Waste in the United Kingdom” (DEFRA, 2007).

HPA. The Health Protection Agency (HPA) was an independent body, now Public Health England (PHE) that protects the health and well-being of the population. The HPA includes the ex-National Radiological Protection Board (NRPB).

HSE. Britain's Health and Safety Commission (HSC) and the Health and Safety Executive (HSE) are responsible for the regulation of almost all the risks to health and safety arising from work activity in Britain.

inadvertent human intrusion. Accidental intrusion into a disposal facility without prior knowledge of the presence of the facility or accidental intrusion, without prior knowledge, into an area adjacent to the facility in such a way that it degrades the environmental safety performance of the facility.

immobilization. Conversion of *waste* into a *waste form* by *solidification*, *embedding* or *encapsulation*. The aim is to reduce the potential for *migration* or *dispersion* of *radionuclide*s during handling, transport, *storage* and/or *disposal*. See also *conditioning.*

inert waste. Material which does not undergo any significant physical, chemical or biological transformations; does not dissolve, burn or otherwise physically or chemically react, biodegrade or adversely affect other matter with which it comes into contact in a way likely to give rise to environmental pollution or harm to human health; and whose total leachability and pollutant content and the ecotoxicity of its leachate are insignificant and in particular do not endanger the quality of any surface water or groundwater. This is defined by UK waste legislation for non-radioactive wastes.

infiltration. The downward entry of water through the ground surface into soil or *rock*.

intervention. Any action intended to reduce or avert *exposure* or the likelihood of *exposure* to *source*s which are not part of a controlled *practice* or which are out of control as a consequence of an accident.

leach rate. The rate of dissolution or erosion of material or the release by *diffusion* from a solid, this is hence a measure of how rapidly radionuclides may be released from that material. The term usually refers to the durability of a solid *waste form* but also describes the removal of sorbed material from the surface of a solid or porous bed.

leach test. A test conducted to determine the *leach rate* of a *waste form*. The test results may be used for judging and comparing different types of *waste form*s, or may serve as input data for a *long term safety assessment* of a *repository*. Many different test parameters have to be taken into account, for example water composition and temperature.

leachate. A solution that has been in contact with *waste form* and, as a result, may contain *radionuclide*s.

level, clearance. A value, established by a *regulatory body* and expressed in terms of activity concentration and/or total *activity*, at or below which a *source* of radiation may be released from *regulatory control*. See also *clearance.*

level, exemption. A value, established by a *regulatory body* and expressed in terms of activity concentration and/or total *activity*, at or below which a *source* of radiation may be granted *exemption* from *regulatory control* without further consideration.

licence. A legal document issued by the *regulatory body* granting *authorization* to perform specified activities related to a *facility* or activity. The holder of a current licence is termed a licensee**.** A licence is a product of the *authorization* process, although the term licensing process is sometimes used.

limit, dose. The value of the *effective dose* or the *equivalent dose* to individuals from controlled *practice*s that shall not be exceeded.

liner. (1) A layer of material placed between a *waste form* and a container to resist *corrosion* or any other degradation of a *waste package*. (2) A layer of *clay*, plastic, asphalt or other low permeability material placed around or beneath a landfill site, *repository* or *tailings impoundment* to minimise leakage and/or erosion. (3) A structural component (made, for example, of concrete or steel) on the surface of a tunnel or *shaft* in a *repository*.

LLW. See *waste, low and intermediate level.* Low Level Radioactive Waste. With certain specific exceptions, LLW is defined as waste which has an activity concentration greater than the *out of scope levels* and up to 4,000 Bq g^{-1} for alpha emitters and 12,000 Bq g^{-1} for beta-gamma emitters. Where Bq g^{-1} is Becquerel per gram, a measure of activity within the SI system equivalent to 1 disintegration per second. Where an alpha emitter is a form of radioactive decay involving emission of alpha particles (a helium nucleus). Where beta decay is a type of radioactive decay involving the emission of electrons or positrons.

Low Level Waste Repository (LLWR). The LLWR is located 6 km southeast of Sellafield near the village of Drigg, and has operated safely for over 40 years disposing of Low Level Radioactive Wastes (LLW) from the nuclear and general industries, universities and hospitals.

long term. In *radioactive waste disposal*, refers to periods of time that exceed the time during which active *institutional control* can be expected to last.

long term stewardship. Conducting, supervising, or managing something entrusted to one's care. In the context of nuclear waste sites the phrase encompasses the activities undertaken after closure of the site to maintain and monitor the wastes in the long term.

LSG. Local Stakeholder Group. A group of stakeholders that meet regularly in relation to a nuclear licensed site.

Isotope. Different forms of atoms of the same element that have different numbers of neutrons in their nuclei. An element may have a number of isotopes. For example, the three isotopes of hydrogen are protium, deuterium, and tritium. All three have one proton in their nuclei, but deuterium also has one neutron, and tritium has two neutrons. Different isotopes can have different radioactive properties and present different risks.

migration. The movement of contaminants in the environment as a result of natural processes.

minimization, waste. The process of reducing the amount and *activity* of *radioactive waste* to a level as low as reasonably achievable, at all stages from the *design* of a *facility* or activity to *decommissioning*, by reducing *waste* generation and by means such as recycling and reuse, and *treatment*, with due consideration for secondary as well as primary *waste.* See also *pretreatment*; *treatment*; *volume reduction*.

model. A representation of a system and the ways in which phenomena occur within that system, used to simulate or assess the behaviour of the system for a defined purpose.

model, computational. A calculation tool that implements a *mathematical model*.

model, conceptual. A set of qualitative assumptions used to describe a system.

model, mathematical. A set of mathematical equations designed to represent a *conceptual model*.

model, pathways. A mathematical representation used to simulate the transport of *radionuclide*s from a *source* to a receptor.

model, transport. A mathematical representation of mechanisms controlling the movement of finely dispersed or dissolved substances in fluids.

monitoring. Continuous or periodic measurement of radiological and other parameters or determination of the status of a system.

naturally occurring radioactive material (NORM). Material containing no significant amounts of *radionuclide*s other than *naturally occurring radionuclide*s. The exact definition of 'significant amounts' would be a regulatory decision. Materials in which the *activity* concentrations of the *naturally occurring radionuclides* have been changed by human made processes are included. These are sometimes referred to as technically enhanced NORM or TENORM.

naturally occurring radionuclides. *Radionuclide*s that occur naturally in significant quantities on earth. The term is usually used to refer to the primordial *radionuclide*s potassium-40, uranium- 235, uranium-238 and thorium-232 (the decay product of primordial uranium-236), their radioactive decay products, and tritium and carbon-14 generated by natural *activation* processes.

NDA. Nuclear Decommissioning Authority. A public body that oversees nuclear decommissioning in the UK on designated sites such as Harwell.

nuclear facility. A facility and its associated land, buildings and equipment in which radioactive materials are produced, processed, used, handled, stored or disposed of on such a scale that consideration of safety is required.

nuclear material. Plutonium except that with isotopic concentration exceeding 80% in plutonium- 238; uranium-233; *uranium* enriched in the isotope 235 or 233; uranium containing the mixture of isotopes occurring in nature other than in the form of ore or ore residue; any material containing one or more of the foregoing.

nuclear site licence. A licence issued under the Nuclear Installations Act.

off-site. Outside the physical boundary of a *site*.

ONR. Office for Nuclear Regulation. Under UK law (the Health and Safety at Work etc. Act 1974) employers are responsible for ensuring the safety of their workers and the public, and this is just as true for a nuclear site as for any other. This responsibility is reinforced for nuclear installations by the Nuclear Installations Act 1965 (NIA), as amended. Under the relevant statutory provisions of the NIA, a site cannot have nuclear plant on it unless the user has been granted a site licence by the Health and Safety Executive (HSE). This licensing function is administered by HSE's Office for Nuclear Regulation (ONR).

on-site. Within the physical boundary of a *site*.

operation. All the activities performed to achieve the purpose for which a *facility* was constructed.

operational period. The period during which a *nuclear facility* (e.g. a *repository*) is being used for its intended purpose until it is decommissioned or is submitted for permanent *closure*.

optimization. The process of determining what level of protection and safety makes *exposure*s, and the probability and magnitude of *potential exposure*s, ‘as low as reasonably achievable, economic and social factors being taken into account’ (ALARA).

out of scope level (OoSL). The activity concentration of a radionuclide that is out of the scope of the radioactive substances regulations. Material and waste containing levels of radioactivity below the OoSL are not considered to be *radioactive material* or radioactive waste. Often the same as *clearance levels*.

overpack. A secondary (or additional) outer container for one or more *waste package*s, used for handling, transport, *storage* or *disposal*.

package, waste. The product of *conditioning* that includes the *waste form* and any container(s) and internal *barrier*s (e.g. absorbing materials and *liner*s), prepared in accordance with the *requirement*s for handling, transport, *storage* and/or *disposal*.

permeability, *k*. The ability of a porous medium to transmit fluid.

Permit. A document issued by the Environment Agency to allow the accumulation, disposal or discharge of waste.

plume. The spatial distribution of a release of airborne or waterborne material as it disperses in the environment.

PHE. Public Health England (PHE) is an independent body, formerly The Health Protection Agency (HPA), that protects the health and well-being of the population. The HPA includes the ex-National Radiological Protection Board (NRPB).

porosity. The ratio of the aggregate volume of interstices in *rock*, soil or other porous media to its total volume.

post-closure period. The period of time following the *closure* of a *repository* and *decommissioning* of related surface facilities. Some type of *surveillance* or control will probably be maintained in this period, particularly for *near surface repositories*. See also *closure*; *preclosure period*.

practice. Any human activity that introduces additional *sources* of *exposure* or *exposure pathway*s or extends *exposure* to additional people or modifies the network of *exposure pathway*s from existing *sources,* so as to increase the *exposure* or the likelihood of *exposure* of people or the number of people exposed.

preclosure period. The period of time spanning the construction and *operation* of a *repository* up to and including the *closure* and *decommissioning* of related surface *facilities*. See also *closure*; *post-closure period*.

predisposal. Any *radioactive waste management* steps carried out prior to *disposal*, such as *pretreatment*, *treatment*, *conditioning*, *storage* and transport activities. *Decommissioning* is considered to be a part of predisposal management of *radioactive waste*.

pretreatment. Any or all of the operations prior to *waste treatment*, such as collection, *segregation*, chemical adjustment and *decontamination*.

quality assurance (QA). Planned and systematic actions necessary to provide adequate confidence that an item, process or service will satisfy given *requirement*s for quality, for example those specified in the *licence*.

quality control (QC). The part of *quality assurance* intended to verify that systems and components correspond to predetermined *requirement*s.

radioactive material. Material designated in national law or by a *regulatory body* as being subject to *regulatory control* because of its *radioactivity*.

radioactivity. The phenomenon whereby atoms undergo spontaneous random disintegration, usually accompanied by the emission of radiation.

radionuclide. A nucleus (of an atom) that possesses properties of spontaneous disintegration (*radioactivity*). Nuclei are distinguished by their mass and atomic number.

records. A set of documents, such as instrument charts, certificates, log books, computer printouts and magnetic tapes for each *nuclear facility,* organized in such a way that it provides past and present representations of facility *operation*s and activities including all phases from *design* through *closure* and *decommissioning* (if the facility has been decommissioned). Records are an essential part of *quality assurance*.

regulatory body. An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing *authorization*s*,* and thereby for regulating the *siting, design,* construction, *commissioning*, *operation*, *closure*, *decommissioning* and, if required, subsequent *institutional control* of the *nuclear facilities* (e.g. *near surface repositories*) or specific aspects thereof.

remedial action. Action taken when a specified action level is exceeded, to reduce a radiation *dose* that might otherwise be received, in an intervention situation involving chronic *exposure*. Examples are: (a) actions which include *decontamination*, *waste* removal and environmental restoration of a *site* during *decommissioning* and/or *closure* efforts; (b) actions taken beyond stabilization of *tailings impoundments* to allow for other uses of the area or to restore the area to near pristine conditions.

repository. A *nuclear facility* where *waste* is emplaced for *disposal*.

repository, near surface. A *facility* for *disposal* of *radioactive waste* located at or within a few tens of metres from the earth's surface.

representative person. See *critical group*.

retardation. A reduction in the rate of *radionuclide* movement through the soil due to the interaction (e.g. by *sorption*) with an immobile *matrix*.

retardation coefficient, *R*d. A measure of capability of porous media to impede the movement of a particular *radionuclide* being carried by fluid.

retrievability. The ability to remove *waste* from where it has been emplaced.

risk. A multiattribute quantity expressing hazard, danger or chance of harmful or injurious consequences associated with actual or *potential exposures.* It relates to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences. (2) The combination of the frequency, or probability, of occurrence and the consequence of a specified hazardous event. The concept of risk always has two elements: the frequency or probability with which a hazardous event occurs and the consequences of the hazardous event. Risk = Probability x Consequence.

safety case. An integrated collection of arguments and evidence to demonstrate the safety of a *facility.* This will normally include a *safety assessment*, but could also typically include information (including supporting evidence and reasoning) on the robustness and reliability of the *safety assessment* and the assumptions made therein.

safety culture. The assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

safety report. A document required from the *operating organization* by the *regulatory body* containing information concerning a *nuclear facility* (e.g. a *repository*), the site characteristics, *design*, operational procedures, etc., together with a *safety analysis* and details of any provisions needed to restrict *risk* to personnel and the public.

scenario. A postulated or assumed set of conditions and/or events. They are most commonly used in *analysis* or *assessment* to represent possible future conditions and/or events to be modelled, such as possible accidents at a *nuclear facility*, or the possible future evolution of a *repository* and its surroundings.

screening. A type of *analysis* aimed at eliminating from further consideration factors that are less significant for the purpose of the *analysis*, in order to concentrate on the more significant factors. Screening is usually conducted at an early stage in order to narrow the range of factors needing detailed consideration in an *analysis* or *assessment*.

segregation. An activity where *waste* or materials (radioactive and exempt) are separated or are kept separate according to radiological, chemical and/or physical properties which will facilitate *waste* handling and/or *processing*. For example, it may be possible to segregate radioactive waste from exempt waste and thus reduce the *waste* volume.

Semi infinite plane. A semi-infinite plane is bounded in one direction, i.e. it is a surface, and unbounded in another (stretches infinitely in all directions).

shielding. A material interposed between a *source* of radiation and persons, or equipment or other objects, in order to absorb radiation and thereby reduce radiation exposure.

site. The area containing, or under investigation for its suitability for, a *nuclear facility* (e.g. a *repository*). It is defined by a boundary and is under effective control of the *operating organization*.

solidification. *Immobilization* of gaseous, liquid or liquid-like materials by conversion into a solid *waste form*, usually with the intent of producing a physically stable material that is easier to handle and less dispersible. *Calcination*, drying, *cementation*, *bituminization* and *vitrification* are some of the typical ways of solidifying liquid *waste*. See also *conditioning*; *immobilization*.

solubility. The amount of a substance that will dissolve in a given amount of another substance.

sorption. The interaction of an atom, molecule or particle with the surface of a solid. A general term including absorption (sorption taking place largely within the pores of a solid) and adsorption (surface sorption with a non-porous solid). The processes involved may also be divided into chemisorption (chemical bonding with the substrate) and physisorption (physical attraction, for example by weak electrostatic forces).

source. (1) Anything that may cause radiation *exposure,* such as by emitting ionizing radiation or by releasing *radioactive substances* or materials. (2) More specifically, *radioactive material* used as a source of radiation.

source, natural. A naturally occurring *source* of radiation, such as the sun and stars (*source*s of cosmic radiation) and *rock*s and soil (terrestrial *source*s of radiation).

source term. A mathematical expression used to denote information about the actual or potential release of radiation or *radioactive material* from a given *source*, which may include further *specification*s, for example the composition, the initial amount, the rate and the mode of release of the material.

storage. (1). The holding of *spent fuel* or of *radioactive waste* in a *facility* that provides for its *containment*, with the intention of retrieval. (2). Storage is by definition an interim measure, and the term interim storage would therefore be appropriate only to refer to short term temporary storage when contrasting this with the longer term fate of the *waste*. Storage as defined above should not be described as interim storage.

surface water. Water which fails to penetrate into the soil and flows along the surface of the ground, eventually entering a lake, a river or the sea.

survey, radiological. An evaluation of the radiological conditions and potential hazards associated with the production, use, transfer, release, *disposal*, or presence of *radioactive material* or other *source*s of radiation.

transport, radionuclide. The movement (*migration*) of *radionuclide*s in the environment, for example radionuclide transport by *groundwater*. This could include processes such as *advection*, *diffusion*, *sorption* and *uptake*. This usage does not include intentional transport of radioactive materials by humans (transport of radioactive *waste*s in *cask*s, etc). See also *migration*.

treatment. Operations intended to benefit safety and/or economy by changing the characteristics of the *waste*. Three basic treatment objectives are: *volume reduction,* removal of *radionuclide*s from the *waste* and change of composition. Treatment may result in an appropriate *waste form*.

UKAEA The United Kingdom Atomic Energy Authority (UKAEA) was incorporated as a statutory corporation in 1954 and pioneered the development of nuclear energy in the UK. Today UKAEA are responsible for managing the decommissioning of the nuclear reactors and other radioactive facilities used for the UK's nuclear research and development programme in a safe and environmentally sensitive manner. UKAEA is a non-departmental public body, funded mainly by its lead department the Department of Trade and Industry under contract to the NDA.

uptake. A general term for the processes by which *radionuclide*s enter one part of a biological system from another. Used in a range of situations, particularly in describing the overall effect when there are a number of contributing processes, for example *root uptake*, the transfer of *radionuclide*s from soil to plants through the plant roots.

very low level waste (VLLW). See *waste, very low level*.

volume reduction. A *treatment* method that decreases the physical volume of a *waste*. Volume reduction is employed because it is economical and facilitates subsequent handling, *storage*, transport and *disposal* of the *waste*. Typical volume reduction methods are mechanical *compaction*, *incineration* and *evaporation*. Volume reduction of a given *waste* results in a corresponding increase in *radionuclide* concentration. The total volume of *waste* may also be reduced through *decontamination* (with subsequent *exemption*) or through the avoidance of *waste* generation. See also *minimization, waste*.

waste. Material in gaseous, liquid or solid form for which no further use is foreseen.

waste, alpha bearing. *Radioactive waste* containing one or more alpha emitting *radionuclide*s. Alpha bearing waste can be short lived or long lived.

waste, exempt. *Waste* released from *regulatory control* in accordance with *exemption* principles. See also *clearance level*s; *exemption*.

waste, mixed. *Radioactive waste* that also contains non-radioactive toxic or hazardous substances.

waste, radioactive. For legal and regulatory purposes, *waste* that contains or is contaminated with *radionuclide*s at concentrations or *activities* greater than *clearance levels* or *out of scope levels* as established by the *regulatory body*. It should be recognized that this definition is purely for regulatory purposes and that material with *activity* concentrations equal to or less than *clearance level*s is radioactive from a physical viewpoint — although the associated radiological hazards are considered negligible.

waste, secondary. A form and quality of *waste* that results as a by-product from *processing* of *waste*.

waste, very low level (VLLW). *Radioactive waste* considered suitable by the *regulatory body* for authorized *disposal*, subject to specified conditions, with ordinary *waste* in facilities not specifically designed for radioactive waste disposal.

waste acceptance criteria. Quantitative or qualitative criteria for *radioactive waste* to be accepted by the *operator* of a *repository* for *disposal*, or by the *operator* of a storage facility for *storage*. Waste acceptance criteria might include, for example, restrictions on the *activity* concentration or the total *activity* of particular *radionuclide*s (or types of *radionuclide*) in the *waste* or *requirement*s concerning the *waste form* or *waste package*.

waste form. *Waste* in its physical and chemical form after *treatment* and/or *conditioning* (resulting in a solid product) prior to packaging. The waste form is a component of the *waste package*.

waste generator. The *operating organization* of a *facility* or activity that generates *waste*. See also *operator*.

waste inventory. Quantity, *radionuclide*s, *activity* and *waste form* characteristics of *waste*s for which an operator is responsible.

waste management, radioactive. All activities, administrative and operational, that are involved in the handling, *pretreatment*, *treatment*, *conditioning*, transport, *storage* and *disposal* of *radioactive waste*.

water table. The upper surface of a zone of *groundwater* saturation.

zone, saturated. A subsurface zone in which all the interstices are filled with water. This zone is separated from the *unsaturated zone*, i.e. the zone of aeration, by the *water table*. See also *zone, unsaturated*.

zone, unsaturated. A subsurface zone in which at least some interstices contain air or water vapour, rather than liquid water. Also referred to as the ‘zone of aeration’. See also *zone, saturated*.

Appendix B. Baseline samples of Leachate and Groundwater

GAU - Radioanalytical
(Geosciences Advisory Unit)

UKAS TESTING
2758
Accredited to ISO/IEC 17025:2005

Report

Determination of ^{238}U, ^{235}U, ^{234}U, ^{232}Th, ^{230}Th, ^{228}Th, ^{226}Ra, ^{3}H and gross alpha and gross beta in 8 water samples.
(Samples: KO2A etc...)
UKAEA Harwell

Customer	Jon Blackmore UKAEA B175 Harwell International Business Centre Didcot Oxfordshire OX11 0RA
Customer reference number	Quote620
GAU job number	GAU1278 (Final)
Date samples received	18th August 2008
Report date	1st October 2008
Report produced by	Dr P. Gaca (Radiochemist, GAU-Radioanalytical)
Signed	[redacted]
Report checked by	Dr P. E. Warwick (Deputy Director, G[redacted]dioanalytical)
Signed	[redacted]

Methodology

Samples were received at the National Oceanography Centre, Southampton on **18th August 2008** in good condition.

Gamma spectrometry (Method GAU/RC/2032: Accredited to ISO/IEC 17025:2005)
100ml of the sample was evaporated down to less than 20ml and transferred to a scintillation vial. The sample was then counted on a well-type HPGe detector previously calibrated with a mixed nuclide standard of identical geometry. The resulting spectrum was analysed using Fitzpeaks spectral analysis software. All anthropogenic radionuclides were identified and quantified. In addition ^{60}Co, and ^{137}Cs were specifically searched for and limits of detection reported where no activity was detected.

Gross alpha / beta in waters (Method GAU/RC/2034)
200 ml of the sample was acidified with H_2SO_4 and evaporated to dryness and the residue ignited at 350 °C. The ignited residue was ground and mounted onto a 47 mm filter paper. The source was then counted on a gas flow proportional counter previously calibrated against ^{241}Am (alpha) and ^{137}Cs (beta).

^{3}H in aqueous samples (Method GAU/RC/2004)
50ml of the sample was removed for ^{3}H analysis. The sub-sample was purified by distillation. The ^{3}H content of the distillate was then measured using a Quantulus ultra-low level liquid scintillation counter.

^{226}Ra in aqueous samples (Method GAU/RC/2038)
An aliquot of the aqueous sample is mixed with a water-immiscible scintillation cocktail in a glass vial. The vial is sealed and immediately counted on a Perkin Elmer Quantulus liquid scintillation counter with alpha-beta discrimination activated to determine the total ^{222}Rn activity. The sample is then stored for two weeks and recounted to determine the activity of supported ^{222}Rn/^{226}Ra.

Th isotopes by alpha spectrometry (Method GAU/RC/2027)
An aliquot of the sample is spiked with ^{229}Th and acidified. An iron hydroxide precipitation followed by anion exchange chromatography is used to isolate Th from the solution. The activities of ^{230}Th and ^{232}Th are then determined by alpha spectrometry.

U by alpha spec & ICPMS (Method GAU/RC/2026)
An aliquot of the sample is spiked with ^{232}U and acidified. A combination of anion exchange and extraction chromatography is used to isolate U from the solution. ^{238}U and ^{234}U are determined by alpha spectrometry, and the ^{235}U content is determined relative to ^{238}U by ICP-MS.

Limits of detection / quantification

For gamma data, limits of quantification, L_Q, is calculated as defined by Currie (1968) and Gilmore & Hemingway (2000)

$$L_{Q(gamma)} = 0.5 \times \sigma^2 \times \left(1 + \sqrt{1 + \left(4 \times \left(1 + \frac{n}{2m}\right) \times \frac{C}{\sigma^2}\right)}\right) \times \frac{1}{t} \times \frac{100}{E} \times \frac{100}{Y} \times \frac{1}{M_g}.$$

where σ is set at 2.00, C is the background counts, n is the number of channels covering the peak, m is the number of background channels taken either side of the photopeak, t is the count time in seconds, E is the counting efficiency, Y is the gamma emission probability and M_g is the mass of sample analysed in grams

Limits of detection for H-3 analyses are quoted as L_D as defined by Currie, 1968.

$$L_D(Bq/g) = \frac{2.71 + 4.65\sqrt{C}}{t} \times \frac{100}{E} \times \frac{100}{R} \times \frac{1}{M_g}$$

where C is the background count, t is the count time in seconds, E is the measurement efficiency, R is the chemical recovery and m is the sample mass in grams.

References

Currie L.A. (1968). Limits of qualitative detection and quantitative determination. *Anaytical Chemistry, 40 (3), 586-593.*
Gilmore G. and Hemingway J. (2000). Practical gamma-ray spectrometry. John Wiley, Chichester, UK

Summary of samples and results

All uncertainties quoted are propagated method uncertainties unless otherwise stated.

* Indicates results obtained using an accredited method.

GAU ID	Customer ID	Sample type
GAU1278/1	KO2a	Water
GAU1278/2	KO3	Water
GAU1278/3	KO5	Water
GAU1278/4	KO6	Water
GAU1278/5	KO7	Water
GAU1278/6	KO8	Water
GAU1278/7	KCLW2A2	Water
GAU1278/8	KCLW3A1	Water

Results

Gross alpha/beta

GAU ID	Gross alpha [Bq/L]	+/-	Gross beta [Bq/L]	+/-
GAU1278/1	<0.1	-	0.46	0.15
GAU1278/2	<0.1	-	<0.2	-
GAU1278/3	<0.2	-	<0.3	-
GAU1278/4	<0.2	-	<0.3	-
GAU1278/5	<0.2	-	<0.3	-
GAU1278/6	<0.1	-	<0.2	-
GAU1278/7	<2	-	90	3
GAU1278/8	<1	-	20	1

Coverage factor k=2 S.D.
Uncertainties quoted are propagated method uncertainties

^{3}H

GAU ID	^{3}H [Bq/L]	+/-
GAU1278/1	<5	-
GAU1278/2	<5	-
GAU1278/3	<5	-
GAU1278/4	<5	-
GAU1278/5	<5	-
GAU1278/6	<5	-
GAU1278/7	59	7
GAU1278/8	10	4

Coverage factor k=2 S.D.
Uncertainties quoted are propagated method uncertainties

^{226}Ra

GAU ID	^{226}Ra [Bq/L]	+/-
GAU1278/1	0.30	0.07
GAU1278/2	0.29	0.07
GAU1278/3	0.29	0.07
GAU1278/4	0.30	0.07
GAU1278/5	0.35	0.07
GAU1278/6	0.33	0.07
GAU1278/7	0.34	0.07
GAU1278/8	0.58	0.08

Coverage factor k=2 S.D.
Uncertainties quoted are propagated method uncertainties

^{238}U, ^{235}U, ^{234}U

GAU ID	^{238}U [Bq/L]	+/-	^{235}U [Bq/L]	+/-	^{234}U [Bq/L]	+/-
GAU1278/1	0.039	0.012	<0.005	-	0.066	0.013
GAU1278/2	0.042	0.009	<0.005	-	0.039	0.009
GAU1278/3	0.018	0.005	<0.005	-	0.018	0.005
GAU1278/4	0.016	0.005	<0.005	-	0.028	0.006
GAU1278/5	0.014	0.005	<0.005	-	0.021	0.006
GAU1278/6	0.013	0.005	<0.005	-	0.010	0.006
GAU1278/7	<0.01	-	<0.005	-	<0.01	-
GAU1278/8	<0.06	-	<0.005	-	<0.05	-

Coverage factor k=2 S.D.
Uncertainties quoted are propagated method uncertainties
^{235}U activity concentration calculated using $^{238}U/^{235}U$ ratio obtained with ICP-MS measurement.

^{232}Th, ^{230}Th, ^{228}Th

GAU ID	^{232}Th [Bq/L]	+/-	^{230}Th [Bq/L]	+/-	^{228}Th [Bq/L]	+/-
GAU1278/1	<0.003	-	<0.003	-	<0.005	-
GAU1278/2	<0.002	-	0.0024	0.0014	<0.004	-
GAU1278/3	<0.004	-	0.0030	0.0018	<0.004	-
GAU1278/4	<0.003	-	<0.002	-	<0.006	-
GAU1278/5	<0.003	-	0.0027	0.0018	<0.008	-
GAU1278/6	<0.003	-	0.0026	0.0016	0.0059	0.0022
GAU1278/7	<0.008	-	<0.005	-	<0.007	-
GAU1278/8	<0.002	-	0.0043	0.0020	0.013	0.003

Coverage factor k=2 S.D.
Uncertainties quoted are propagated method uncertainties

Gamma Spectrometry*

Artificial Radionuclides

GAU ID	^{241}Am	+/-	^{60}Co	+/-	^{137}Cs	+/-	^{154}Eu	+/-	^{54}Mn	+/-	^{65}Zn	+/-
GAU1278/1	<1	-	<3	-	<2	-	<30	-	<1	-	<3	-
GAU1278/2	<1	-	<2	-	<1	-	<20	-	<1	-	<3	-
GAU1278/3	<1	-	<3	-	<2	-	<30	-	<2	-	<4	-
GAU1278/4	<1	-	<3	-	<1	-	<20	-	<1	-	<4	-
GAU1278/5	<0.9	-	<2	-	<1	-	<20	-	<1	-	<3	-
GAU1278/6	<0.9	-	<2	-	<1	-	<20	-	<1	-	<3	-
GAU1278/7	<1	-	<2	-	<1	-	<20	-	<1	-	<3	-
GAU1278/8	<0.7	-	<2	-	<0.9	-	<20	-	<0.9	-	<3	-

*Indicates results obtained using an accredited method.
Results are quoted in Bq/L.
Coverage factor k=2 S.D.
Reference date: 18/08/08

Gamma spectrometry*

Natural Radionuclides

GAU ID	^{228}Ac	+/-	^{40}K	+/-	^{210}Pb	+/-	^{212}Pb	+/-	^{214}Pb	+/-	^{226}Ra	+/-	^{208}Tl	+/-	^{234}Th	+/-	^{235}U	+/-
GAU1278/1	<10	-	<30	-	<10	-	1.4	0.6	<3	-	<20	-	<10	-	<10	-	<4	-
GAU1278/2	<10	-	<30	-	<10	-	<1	-	<4	-	<20	-	<10	-	<10	-	<4	-
GAU1278/3	<10	-	<40	-	<10	-	<1	-	<4	-	<20	-	<10	-	<10	-	<4	-
GAU1278/4	<10	-	<30	-	<10	-	<2	-	<4	-	<20	-	<10	-	<20	-	<4	-
GAU1278/5	<7	-	18	9	<10	-	<1	-	<3	-	<10	-	<9	-	<10	-	<4	-
GAU1278/6	<7	-	<30	-	<10	-	1.3	0.6	<3	-	<10	-	<9	-	<10	-	<4	-
GAU1278/7	<20	-	93	15	<10	-	4.2	0.8	<4	-	<20	-	<10	-	<20	-	<4	-
GAU1278/8	<6	-	18	8	<10	-	2.3	0.7	<2	-	<10	-	<10	-	<10	-	<3	-

*Indicates results obtained using an accredited method.
Results are quoted in Bq/L.
Coverage factor k=2 S.D.
Reference date: 18/08/08

Appendix C. Stakeholder Engagement Programme

C.1. Planned Communications Meetings

392. The communications events which are planned before the end of 2015 are listed in Table 31.

Table 31 Forthcoming communications events

Forthcoming	Communications	Activities
Spring 2015 Flyers, news releases and advertisements	Public	Publicise forthcoming ENRMF Open Day and Environment Agency consultation on ENRMF Environmental Permit application.
Spring 2015 Enewsletter	Public	Enewsletter to explain Environment Permit application process with links to further information will be sent to local Parish Councils
October 2015 ENRMF Open Day	Public	Site Open Day. Tours of the site, soil treatment plant and laboratory available as well as an opportunity to speak with The Environment Agency and Augean staff.

C.2. Record of previous meetings and activities

393. The meetings and activities that have occurred since 2006 are listed below in date order.

Table 32 Consultation record for extension of the ENRMF landfill

Date & event	Parties informed	Communication
26/12/06 KCLG meeting	KCLG	Under Augean update at Paragraph 11, Page 2: "*Northamptonshire is looking at forward planning. Current planning consent is to 2011 but Augean want to look forward to 20 years and they will be putting something forward to the Issues and Options stage of the Minerals and Development framework. This would include Augean keeping open their option for land to the west as a possible extension*"
12/04/07 Minerals and Waste Development Framework (MWDF) submission	NCC	Submission to the MWDF process promoting the allocation of the western extension of King's Cliffe Landfill. The submission included an appraisal following the County Site Assessment Methodology, a landscape report and a hydrogeology report.
6/06/07 KCLG meeting	KCLG	At this meeting Augean made a presentation regarding proposals for a soil treatment plant and

Date & event	Parties informed	Communication
		for the extension of the site. Under King's Cliff Future Paragraph 4.9, Page 2: "*Waste Development Framework. An extension to the site has been put forward for consideration as part of this process"* Paragraphs 4.15 and 4.16, Page 3: *"Fiona Cowan – What is Augean's thinking on the strategy of the life of the site, will it last longer?* (Note the question was posed in the context of the proposed soil treatment facility which was later approved and installed) *"Gene Wilson – Most probably it will last longer. However, it is difficult to predict because the definitions of hazardous waste are changing bringing more waste streams into hazardous. It will be a couple of years before the final assessment of this can be made but it is likely that there will be need for an extension."* Under County Council Update Paragraph 6.1, Page 4: *"Phil Watson explained that the Minerals and Waste Development Framework Issues and Options Consultations have completed. Augean have put forward an extension to their site for consideration."*
20/06/07 MWDF submission	NCC	Submission of further information comprising cultural heritage assessment, soils and agricultural classification and Phase 1 habitat survey in support of the inclusion of King's Cliffe extension in the locations document.
10/09/07 NCC cabinet meeting	NCC	The proposed allocation of King's Cliffe as a preferred option in the MWDF was considered at the Cabinet meeting: *"However following a representation made to the issues and options consultation, and because of the significance of the site, it is proposed that the landfill site at King's Cliffe that deals with hazardous waste should have a reserve area added to it. However planning permission should not be forthcoming for this extension until at least 2016, and furthermore the extension should be for hazardous waste from a defined catchment area of less than a 50 mile radius".* No objection or comment regarding the allocation of King's Cliffe is included in the minutes of the meeting
29/09/07 KCLG meeting	KCLG	Under County Council Update Paragraph 4.1, Page 2: *"Phil Watson confirmed that the Minerals and Waste Development Framework has reached the Preferred Options stage and that this will be sent out for public consultation shortly. An extension to the hazardous waste site is in the document for consideration. This is the land at the*

Date & event	Parties informed	Communication
		rear of the existing site as previously explained by Augean at the last meeting."
__/10/07 Preferred options document to the MWDF	NCC	The document includes Preferred option WA16: King's Cliffe Western Extension (Pages 207 to 209) An initial and a detailed evaluation of the site is presented in the Technical appendices to the Preferred Options Documents (Pages 239 to 243 and Pages 533 to 542)
4/12/07 Augean comments on the MWDF Core Strategy and Preferred Options	NCC	Comments were submitted in respect of the Core Strategy and clarification of certain details of the allocation of the King's Cliffe extension. (Page 5)
--/12/07 Responses to the Preferred options	N/A	Responses were received from Collyweston, East Northamptonshire Council (ENC), Environment Agency (EA), Peterborough City Council, English Heritage and a number of local residents some of whom are members of Waste Watchers.
6/02/08 KCLG meeting	KCLG	Under County Council Update Paragraph 3.5, Page 2: *"Minerals and Waste Development Framework update. The consultation on the Preferred Options document is now complete. The site extension proposed by Augean did have some comments but not a huge amount of response. The Parish Council said it would be useful to know how many were received. This was subsequently found to be 26 respondents of which:* *19 objected* *5 supported* *1 supported on condition that a 50 mile catchment is implemented* *1 raised concern but did not object"* Under Local Community Update Paragraph 3.8, Page 2: *"Heather Smith pointed out that some people were concerned that the site could be in existence for longer than originally planned"*
14/02/08 KCPC meeting	KCPC	A Power Point presentation was made to the Parish Council regarding the site operations and development. Slides 20 to 22 provide information regarding the proposals for extension of the site. On slide 22 it is stated: *"If progressed application not until at least 2010* *Extensive studies needed and full Environmental Impact Assessment (EIA)* *Full public consultation including exhibition"*
28/01/09 KCLG meeting	KCLG	Under County Council Update Paragraph 6.1, Page 3: *"Phil Watson pointed out that an extension to the King's Cliffe landfill was not specifically*

Date & event	Parties informed	Communication
		proposed in the plan (or any extension to existing or new landfill sites) as it was considered there is enough provision at least until 2016. However paragraph 3.23 of the document was read out which enables future extensions to be considered later in the plan period if applicants 'robustly justify need and ensure that only residual wastes are disposed of'."
11/3/09 Representation on the MWDF Locations Proposed Submission Development Plan Document (DPD)	NCC	In the Proposed Submission DPD of January 2009 the County determined that no specific provision would be made for non-inert sites including hazardous wastes Augean made representation that the approach of not allocating specific sites for hazardous waste is unsound. The current planning permission for this site expires in 2013 and there will be a need for additional capacity during the Plan Period
20/05/09 Pre-exhibition meeting for low level radioactive waste (LLW)	KCLG, King's Cliffe Parish Council (KCPC), NCC, ENC	From our notes of the meeting the following question was raised: Q: Will you extend the site to take more? A: As you know we will most likely need more time to complete the site and we can't guarantee there won't be an extension.
21/05/09 LLW Exhibition and surgery	Public	During the exhibition and surgery a number of persons asked about the long term future of the site. The following information was given: It is possible that the site will not be complete by the current planning date of 2013 Due to the recession and the volatile nature of the market it is difficult to predict the life of the site It is likely we shall be in a position to make a decision in mid 2010 Augean has made submissions to the MWDF regarding the extension of the site but has not made a decision whether to pursue the proposal Any extension of time or new void space will be the subject of full public consultation and EIA
--/05/09 onwards	Public, NCC, parish councils	Since the public exhibition on the 21st May 2009 Augean has made 19 public presentations regarding the proposals. At many of the presentations questions were asked regarding the long term future of the site. The answers were consistent with the answers given at the public exhibition.
17/03/10 Email and report to EA	EA	Following instruction in January ESI consultants submitted to the EA a report regarding the acceptability of the extension of void in respect of Groundwater Policy
20/04/10 EA	EA	The EA confirm that there is no objection on groundwater policy grounds to the proposal for further void space subject to risk assessment and

Date & event	Parties informed	Communication
Letter		providing an appropriate in situ geological barrier.
14/06/10 Meeting with planning officers	NCC	Augean informed NCC of its intention to make the application for extension of time and for new void space. The following information was given: The application will be within the landownership of Augean and the extant planning boundary The estimated additional life of the current void is 3 years The estimated additional void will be 10 years The application will include the soil treatment facility The application will be made in mid 2011 Consideration is being given to inclusion of LLW The application will be made under the major infrastructure regime Full EIA will be conducted Full public consultation will be conducted in early 2011
7/07/10 KCLG meeting	KCLG	Augean informed the attendees, which included representatives from Apethorpe, Laxton and Woodnewton, its intention to make an application for planning permission for extension of time and void space. Similar information was given to that of the meeting with the planning officers on 14th February 2010.
28/07/10	Infrastructure Planning Commission (IPC)	Letter sent by Augean to IPC to notify the IPC of intention to submit an application.
09/08/10	IPC	IPC response to Letter of 28/07/11 explaining that the relevant section of the Planning Act 2008 is not yet in force.
--/09/10 Community Newsletter	Public, NCC, ENDC, Parish Councils	A section on the time and void extension was included in the September 2010 Community Newsletter and distributed widely in the local area and made available online. Paragraph 1: "*At the last Local Liaison Group meeting in July, we made known our intentions regarding the future of the site. For a number of years we have explained to local community organisations that ENRMF has not filled as quickly as anticipated and that it may be necessary to extend its life beyond the planned completion date in 2013. It is now clear that it is unlikely that we can complete this site by then and consequently we plan to apply for an extension of time.*" Paragraph 4 and 5: "*For management reasons and to give certainty to the local community about the future of the site, we have decided to seek additional void space by preparing a single*

Date & event	Parties informed	Communication
		application to cover the extension in time and an extension in landfill area. This will give the site an additional ten year life. *"Environmental studies and scheme design work are currently underway but the details will not be complete until the end of the year. Thereafter, in early 2011, the company will consult widely with local communities before submitting an application in mid-2011."*
07/01/11 Meeting Phil Watson	NCC	Informally discussed with Phil Watson the nature and scope of community consultation and how this could be improved.
11/01/11 Letter Clare Langan		Following comments made at the Inquiry, asked advice on public consultation and how this could be improved.
28/01/11 Letter Heather Smith		Asking advice on public consultation and how this could be improved.
31/01/11 Clare Langan Consultation response		Clare Langan responds positively and welcomes any improvements in consultation, in particular improved two way communication.
02/02/11 KCLG Meeting	KCLG	Members of the KCLG were given presentations on the proposed application and to ask advice on the nature and scope of the consultation proposed to be undertaken as set out in the draft Statement of Community Consultation (SOCC). See minutes Paragraphs 4.36 to 4.45
16/02/11	NCC	EIA Scoping document sent of NCC and circulated to range of consultees.
18/02/11 Draft SOCC circulated	KCLG, NCC, ENDC, Parish Councils, Kings Cliffe Waste Watchers (KCWW)	A draft Statement of Community Consultation (SoCC), outlining the scope and methods of the consultation that Augean proposed, was circulated among members of the King's Cliffe Local Liaison Group for their advice and comments. In addition, the draft SoCC was also sent to the clerks of Apethorpe Parish Meeting and Woodnewton Parish Councils, to pass on to their nominated representative on the Liaison Group and to a representative of KCWW for their comments by a deadline of 23 March 2011
25/02/11 Stamford Mercury	Public	At the end of an article on the referendum it is said that "The company is also due to be submitting two

Date & event	Parties informed	Communication
		planning applications later this year to extend its site and be given more time to use the dump. The permission runs out in 2013."
25/02/11	NCC ENDC Parish Councils Statutory Consultees	Requests for Scoping opinion were sent to 78 consultees on 25 February 2011 and comments requested by a deadline of 18 March 2011
02/03/11 Heather Smith draft SOCC response		Email from Heather Smith on draft SOCC suggesting advertising the consultation events in parish magazines and the council newspaper.
10/03/11 Mark Chant Draft SOCC response	NCC	Requested that note should be taken of the sensitivity of the local community following the application for LLW.
06/04/11 KCLG Meeting	KCLG	Update on the scoping and clarification of what scoping exercise encompasses. Agreement for NCC to write to all consultees to clarify scoping purpose. Additional time for responses agreed to 13 May 2011. Details of the consultation to take place following advice on the draft SoCC
08/04/11 Woodnewton Parish Magazine		Woodnewton Parish Magazine given details of the consultation events to publish.
15/04/11 Stamford Mercury	Public	The Stamford Mercury publishes a statement from Augean regarding the results of the referendum which also restates Augean's commitment to public consultation and the forthcoming consultation on the time and void extension.
27/04/11	NCC Statutory Consultees	NCC write to all EIA Scoping opinion consultees to clarify purpose of the scoping consultation. Further comments requested by deadline of 13 May 2011
05/05/11	Statutory Consultees	Preliminary Environmental Information Report, Non- Technical Summary , a copy of the SoCC and Public Information Leaflet circulated to 129 statutory consultees with a request for comment by 8 July 2011
06/05/11 SOCC Published	Public	SOCC advertised in the Stamford Mercury.
06 & 07/05/11 Public Information Leaflet and Going	Public, All Elected Reps	Public Information Leaflet were sent by addressed mail to around 2,600 households and business premises in the villages of Apethorpe, Barrowden,

Date & event	Parties informed	Communication
Public		Blatherwycke, Bulwick, Collyweston, Dunnington, Easton on the Hill, Fineshade, Fotheringhay, Harringworth, King's Cliffe, Laxton, Nassington, Tixover, Wakerley, Woodnewton and Yarwell. A copy of the SoCC, multiple copies of the Public Information Leaflet and a disc of the Preliminary Environmental Information and Non-technical Summary was mailed to elected representatives at every level, including MPs, Northamptonshire County, East Northamptonshire District and Peterborough City Councillors, all 47 of the Parish Councils and Parish Meetings, including members of Parish Watch, within a 10 kilometre radius of the site, the Town Councils of Stamford and Oundle, the King's Cliffe Local Liaison Group members, Kings Cliffe Waste Watchers and other interested groups and individuals.
11/05/11 News Release / Online Content	Public	News release regarding consultation days submitted to the news media and Nene Valley News. A shorter version issued to parish councils requesting inclusion in Parish Magazines and to put on their website. A dedicated area for the application and its consultation was placed on the Augean website
13/05/11 Stamford Mercury	Public	The Stamford Mercury publishes an article on and giving details of the Consultation Days
14/05/11 Nene Valley News	Public	Nene Valley News, an East Northants District Council publication, prints an article on and giving details of the Consultation Days
19 – 21/05/11 Consultation Days	Public	Community Consultation Days were held in the King's Cliffe Memorial Hall on Thursday 19 May 2011 from 13:00 to 19:00, Collyweston Village Hall on Friday 20 May from 13:00 to 19:00 and in Woodnewton Village Hall on Saturday 21 May from 10:00 to 15:00. Representatives from Augean and their professional team were present to explain or discuss the proposals with visitors. 110 people attended the three Consultation Days.
19/05/11 Interviews	Public	Interviews were given at the King's Cliffe Consultation Day to Martin Borley of BBC Radio Northants and Mike Sargeant of BBC Radio 4.
20/05/11 Peterborough Evening Telegraph	Public	Article published regarding the King's Cliffe Consultation Day published in the Peterborough Evening Telegraph
20/05/11 Northants	Public	Article published regarding the Consultation Days and details of the events published in the Northants Evening Telegraph. Quotes taken from

Date & event	Parties informed	Communication
Evening Telegraph		Waste Watchers and Local Democracy in Action, but Augean not asked for comment.
26/05/11 Northants Evening Telegraph	Public	Letter from Augean published in Northants Evening Telegraph correcting some of the statements made in the article published on 20/05/11. Details also given of the forthcoming Open Day.
--/06/11 Open Day Advertising	Public	Flyers giving details of the event were hand-delivered to the villages of Apethorpe, Barrowden, Blatherwyck, Bulwick, Collyweston, Dunnington, Easton on the Hill, Fineshade, Fotheringhay, Harringworth, King's Cliffe, Laxton, Nassington, Tixover, Wakerley, Woodnewton and Yarwell. Copies of these flyers and posters were also sent to the Parish Councils, Parish Meetings and Town Councils within 10 kilometers of the site, with a request that the posters be displayed at suitable location. An advertisement was also placed in the Stamford Mercury. A banner advertising the Open Day was displayed on the site entrance a week before the event. A news release was also issued to the news media.
07/06/11 Open Day News Release		News Release regarding the site Open Day is issued to the local news media.
08/06/11 Site Visit		Two member of the public unable to attend the Open Day arranged to have a site visit beforehand.
09/06/11 Northants Evening Telegraph	Public	Details of the Open Day are published in an article in the Northants Evening Telegraph.
10/06/11 Stamford Mercury	Public	Details of the Open Day are published in an article in the Stamford Mercury.
11/06/11 Site Open Day	Public	ENRMF was opened to the public on Saturday 11 June 2011 between 11:00 and 15:00. In addition to a tour of the site, soil treatment facility and the laboratory, the information boards and materials used on the Consultation Days were displayed in the site offices, manned by Augean staff and their professional team. Representatives from the Environment Agency, Health Protection Agency and Research Sites Restoration Limited were also available to answer questions. 87 members of the public and news media attended.
11/06/11 Interviews		Interviews are given to journalists from BBC Radio Cambridgeshire and BBC Look East who attended the Open Day.

Date & event	Parties informed	Communication
11/06/11 BBC Website	Public	Article on the Open Day published on the website of BBC Cambridgeshire.
13/06/11	Public	Follow-up article published regarding the Open Day by a journalist who attended the event.
20/07/11 KCLG meeting	KCLG	Update KCLG on the progress of the application and the consultation events and progress of engineering works. As a result of comments on consultation days, Augean had undertaken a survey of the routing of Heavy Goods Vehicles (HGVs) to the site. Results were made known to KCLG, who also gave advice on possible routing of vehicles transporting LLW.
23/07/11 Workshop	Public	A Workshop was rescheduled from June to take place on Saturday 23 July 2011 from 9:30 to 16:45 at the Haycock Hotel, Wansford. The new date and details of the Workshop was circulated to elected representatives, all Parish Councils, Parish Meetings and Town Councils and those who had registered their details on the list of stakeholders. A total of 12 members of the local community attended, with the event chaired by an independent facilitator. The Workshop was divided into four separate subjects, namely; ***Development Options and Constraints***, ***Environmental Impact Assessments***, ***Radiation*** and ***Monitoring***. A section on radiation was given by a representative of the Health Protection Agency.
05/08/11 Community Feedback	Public	The first in a series of Topic Sheets sent by email to a list of local stakeholders, developed during the consultation process. Feedback Topic Sheet 1: Water Protection
08/07/11	Statutory Consultees	Deadline for responses from Statutory Consultees. 36 replies received
12/08/11 Community Feedback	Public	Feedback Topic Sheet 2: Safe Transport of Waste
15/08/11 Engineering Open Days	Public	See How a Landfill Site is Engineered. Email sent to stakeholders informing them of days in September when the public could visit the site to see the engineering of a cell.
19/08/11 Community Feedback	Public	Feedback Topic Sheet 3: Site Monitoring

Date & event	Parties informed	Communication
26/08/11 Community Feedback	Public	Feedback Topic Sheet 4: Land Ownership
01/09/11 Engineering Open Days Reminder	Public	Email to stakeholders reminding them of the open days.
02/09/11 Community Feedback	Public	Feedback Topic Sheet 5: Waste Hierarchy and the Proximity Principle
05 - 16/09/11 Engineering	Public	From Monday 5 September until Friday 16 September, the public could see the engineering in progress between 11:00 and 16:00 and also on Saturday 10 September from 9:00 to 13:00 by appointment. A News Release was also sent to members of the local news print and broadcast media.
05/09/11 Site Visit	Public	Visit to the site of Wansford residents.
07/09/11 Public Feedback	Public	Feedback from Wansford visitor: "a great deal of confidence"
12/09/11 Public Feedback	Public	Email of thanks from a King's Cliffe resident for engineering presentation and tour.
13/09/11 Public Feedback	Public	Email of thanks from David Burgess (KCLG) for engineering presentation and tour
13/09/11 IPC Change	KCLG	Notify KCLG of the change of application from NCC to the IPC
16/09/11 Community Feedback	Public	Notify stakeholder list by email of the change of application from NCC to IPC.
23/09/11 Community Feedback	Public	Feedback Topic Sheet 6: Site Life & Inputs
28/09/11 KCLG	KCLG	Letter to KCLG about transport of LLW on A43 as a result of discussions regarding this at the KCLG meeting of 20/07/11.

Date & event	Parties informed	Communication
03/10/11 Company Newsletter	Public	A company newsletter was distributed in the area containing a section on the consultation held in May, the Open Day, Workshop and giving information about the change to the IPC and the reasons.
05/10/11	NCC	Informal meeting to discuss change to IPC
05/10/11 KCLG Meeting	KCLG	King's Cliffe Liaison Meeting. Discussions about change to IPC
12/10/11 Community Feedback	Public	Email to stakeholder list of the laying of the HDPE liner in the new cell and giving details of a when this was available to view over a two week period.
13/10/11	IPC	Augean re-notify IPC in accordance with Section 46 of their intention to submit an application and the intention to submit an Environmental Statement with the application.
17 – 21/10/11 24 – 28/10/11 Site Visits	Public	The engineering at the site was available to view for two weeks between October 17-21 and 24 -28 by appointment Monday to Friday 10.00 am until 4.00 pm.
25/10/11	IPC	IPC inform Augean of the consultation bodies notified by IPC.
26/10/11 Community Feedback	Public	Feedback Topic Sheet 7: Ensuring Only Suitable Wastes Are Accepted.
30/11/11 Community Feedback	Public	Email to stakeholder list, primarily regarding the granting of appeal hearing in January 2012, but also letting stakeholders know that Augean would update them on the IPC programme when this was clearer.
7/12/11 Site Visit	Oundle Probus Club	12 members of the Oundle Probus Club were given a tour of the site.
11/12/11 Presentation	Stamford Rotary Club	A presentation was given to the Stamford Rotary Club together with a question and answer session at the end.
13/12/11	NCC	Informal discussion about Section 73 Application with NCC
16/12/11 Advertisement	Public	Section 48 advert published in the Stamford Mercury and The Times.
19/12/11 Further	Public	All stakeholders were notified by post or email of a further period of consultation before an application to the IPC, inviting further comments by 29

Date & event	Parties informed	Communication
Consultation Notification		January 2012. They were also sent a copy of the Section 48 notification together with an update table showing a schedule of the material aspects of the proposed development which have changed or may change compared with the description in the Preliminary Environmental Information Report (PEIR) dated April 2011.
20/12/11 Advertisement	Public	Section 48 advert published in the London Gazette.
23/12/11 Advertisement	Public	Section 48 advert published in the Stamford Mercury For the second time.
30/12/11 Draft DCO	NCC ENDC	Augean send draft Development Consent Order (DCO) to NCC and ENDC for consultation
04/01/11 NCC Meeting	NCC	Meeting to discuss Section 73 applications and submission of application to IPC
11/01/11 Presentation	Stamford Rotary Club	Presentation to members of Stamford Rotary Club.
20/01/12 Planning update	Public	Update on planning; expected submission date to IPC and notification of Section 73 application.
19/01/12 IPC presentation	IPC NCC ENDC Rural Community Council (RCC) Peterborough City council (PCC)	Inception meeting with presentation by IPC. Site visit by council officers and ward councillors and chairman of PCC planning committee to ENRMF
27/01/12 NCC response	NCC	NCC initial response to draft DCO
29/01/12 End of Further Period of Consultation	Public	Deadline for responses to the further period of pre-application consultation.
03/02/12 Radio interview	News media	Interview given to Samantha Appleby of BBC Radio Cambridgeshire.
08/02/12 NCC meeting	NCC	Meeting with NCC to discuss draft DCO
20/02/12 Letter	NCC	Letter from Augean to NCC regarding draft DCO
29/02/12 Liaison Group	KCLG	Liaison Group meeting. Discussion about the IPC application process. Confirm post-submission arrangements; advertising and local community notification via Public Information Leaflet.
16/02/12	NCC, ENDC	Sent progress update confirming submission of

Date & event	Parties informed	Communication
Planning update	RCC PCC	application to Planning Inspectorate
09/03/12 College visit	Education	Educational visit by year 13 students from Lodge Park Technical College, Corby studying Environmental Health
20/03/12 Enewsletter	Public	Enewsletter sent to all on Register of Stakeholders confirming submission of application to IPC
23/03/12 Fact sheet	Public	Fact sheet about proposed applications and operational matters prepared for Persimmon Homes for distribution to potential buyers at Sovereign Grange. Offer to brief Persimmon site sales staff
13/04/12 Letter	NCC ENDC RCC PCC	Sent informal notification of acceptance of the application for examination by the Planning Inspectorate
20/04/12 Public notice published	Public	Publication of statutory Section 56 Notice in The Times, London Gazette and Stamford Mercury
20/04/12 Letter	Public	Formal notification letters sent to Section 42, Former Section 42 and Section 47 consultees and all on the Register of Stakeholders about the acceptance of the application for examination by Planning Inspectorate.
20/04/12 Application documents	NCC ENDC RCC PCC	Liaison about public access to hard copies of application at council offices
23/04/12 Public Information Leaflet	Public	Public Information leaflet about acceptance of the application, summary of the proposed application and details of how to make representations to Planning Inspectorate and advertisement of forthcoming drop-in sessions and open day. Delivered to 2,800 households and businesses in Apethorpe, Barrowden, Blatherwycke, Bulwick, Collyweston, Duddington, Easton on the Hill, Fineshade, Fotheringhay, Harringworth, King's Cliffe, Laxton , Nassington, Tixover , Wakerley, Woodnewton and Yarwell. Multiple copies sent to all 47 town , parish councils and parish meetings in the 10k consultation area as well as to MPs County and District Councillors and members of KCLG
27/04/12 Public Notice	Public	Second publication of Section 56 Notice in Stamford Mercury
27/04/12 MP visit	Elected Representative	Visit to ENRMF by Mrs Louise Mensch MP accompanied by Andrew Howard , Chairman King's Cliffe Parish Council
27/04/12 Poster	Planning Inspectorate for England and	Circulate Planning Inspectorate Outreach Event poster to Register of Stakeholders

Date & event	Parties informed	Communication
	Wales (PINS)	
01/05/12 Meeting	NCC ENDC RCC PCC	Offer to meet to discuss and clarify aspects of application that may be of concern to local authorities
02/05/12 Liaison Group	KCLG	Liaison Group meeting discussion points included operational update and recent complaint about mud on road, Augean group and planning updates. Presentation by Planning Inspectorate.
03/05/12 Drop in session	Public	Drop-in session at ENRMF
08/05/12 Drop in session	Public	Drop-in session at ENRMF
09/05/12 Enewsletter	Public	Enewsletter reminder about drop-in sessions
15/05/12 Drop in session	Public	Drop-in session at ENRMF
18/05/12 Newspaper article	Public	Article in Stamford Mercury about drop-in sessions
21/05/12 Enewsletter	Public	Further Enewsletter reminder about last remaining drop-in sessions and extended access to hard copies of application documents
25/05/12 Drop in session	Public	Drop-in session at ENRMF
26/05/12 Drop in session	Public	Drop-in session at ENRMF
31/05/12 Letter	ENDC	Letter to clarify issues raised verbally by ENDC Which was forwarded to district councillors for information
18/06/12 Open Day flyers	Public	Distribution of flyers to 2800 households and businesses in Apethorpe, Barrowden, Blatherwycke, Bulwick, Collyweston, Duddington, Easton on the Hill, Fineshade, Fotheringhay, Harringworth, King's Cliffe, Laxton , Nassington, Tixover , Wakerley, Woodnewton and Yarwell. Multiple copies sent to all 47 town, parish councils and parish meetings in the 10k consultation area as well as to MPs County and District Councillors and members of KCLG.
27/06/12 Enewsletter	Public	Enewsletter sent to Register of Stakeholders with latest monitoring results and planning update
30/06/12 Enewsletter	Public	Enewsletter reminder about open day sent to all on Register of Stakeholders
30/06/12 News release	Media	News release sent to local print and broadcast media
07/0712 Open Day	Public	Annual site open day. Augean professional team available as well as representatives from EA,HPA and RSRL

Date & event	Parties informed	Communication
11/07/12 Meeting	NCC	Meeting held to discuss application and NSIP process and forthcoming Section 73 determination.
20/07/12 Site visit	NCC	Development Control Committee visit ENRMF site.
23/07/12 Supreme Court ruling	Public	Enewsletter sent to all on Register of Stakeholders about Supreme Court ruling on scope of EIA at ENRMF
24/07/12 NCC Development Control Committee	NCC	Local Impact Report debated and adopted by Development Control Committee. Dr Wilson in attendance to answer questions from Councillors about the application.
24/07/12 Radio interview	Media	Interview given to Martin Borley BBC Radio Northants by Dr Wilson
29/08/12 Community Newsletter	Public	Augean Community Newsletter distributed to all households and businesses in King's Cliffe, Duddington, Fineshade, Easton on the Hill, Collyweston, Tixover, Apethorpe, Fotheringhay, Woodnewton, Thornhaugh and Wansford. Multiple copies sent to 47 town and parish councils and parish meetings in the 10k consultation area. Copies sent to elected representatives and members of KCLG and TLG
18/09/12 NCC Development Control Committee	NCC	NCC Development Control Committee determination of Section 73 application meeting. Dr Wilson in attendance to answer questions from County Councillors
18/09/12 News release	Media	News release sent to local print and broadcast media. Interview given by Paul Blackler to BBC Radio Northants
18/09/12 Enewsletter	Public	Enewsletter sent to Register of Stakeholders about approval of Section 73 application
27and 28/09/12 Advertisements	Public	Advertisement of Issue Specific Hearings published in Stamford Mercury and Northants Evening Telegraph and in notices sent to local Parish Councils for display
03/10/12 Liaison Group	KCLG	Liaison Group meeting. Operational and planning update about ENRMF, Cooks Hole and Thornhaugh . Update on Augean group acquisitions
10/10/12 Enewsletter	Public	Enewsletter sent to Register of Stakeholders with planning update and agenda for issue specific hearings
17/and 18/10/12 NSIP public hearings	Public	Issue Specific Hearings on Control of Emissions, Impact on Health and Transport held in King's Cliffe
6/11/12 PCC Planning and Environmental	PCC	PCC Planning and Environmental Protection Committee determination of application to extend lifetime of Thornhaugh landfill site is approved. Dr

Date & event	Parties informed	Communication
Protection Committee		Wilson in attendance to answer questions from City Councillors
7/11/12 News release	Public	News release sent to Peterborough Evening Telegraph, Northants Evening Telegraph and Stamford Mercury about approval of Thornhaugh application
7/11/12 Enewsletter	Public	Enewsletter sent to Register of Stakeholders on approval of Thornhaugh application
7/11/12 Website	Public	Website updated to reflect approval of Thornhaugh application
8 and 9/11/12 Advertisement	Public	Advertisement of further Issue Specific Hearings and Open Floor Hearings published in Stamford Mercury and Northants Evening Telegraph and in notices sent to local Parish Councils for display
21/11/12 Enewsletter	Public	Enewsletter sent to Register of Stakeholders with agenda for Issue Specific Hearings and Open Floor hearings
6 and 7/12/12 NSIP public hearings	Public	Issue Specific Hearings on Draft Development Consent Order, Section 106 undertakings and Local Impact Report matters. Open Floor Hearings on issues of public concern
24/01/13 Liaison Group meeting	TLG	Meeting of Thornhaugh Liaison Group. Planning updates, monitoring and operational issues were discussed
29/01/13 Letter	KCLG TLG	Letter to explain proposed consignments of Air Pollution Control Residues to ENRMF mentioned in planning application by Balfour Beatty Urbaser in response to article in Peterborough Evening Telegraph
30/01/13 Enewsletter	Public	Enewsletter sent to Register of Stakeholders with about close of ENRMF Project examination by Planning Inspector
6/02/13 Liaison Group meeting	KCLG	King's Cliffe Liaison Group meeting Operational matters, joint monitoring with the Environment Agency and the Community Fund were discussed
14/02/13 Monitoring	Public	ENRMF updated monitoring data published on website
14/03/13 MDWF Submission	NCC	Submission of representations to the NCC Minerals and Waste Development Framework Partial Review at the invitation of NCC
18/03/13 Community feedback	Public	Topic Sheet on Transport of radioactive and hazardous waste
8/04/13 Community Feedback	Public	Enewsletter Topic Sheet on Packaging of low level radioactive waste and asbestos

Date & event	Parties informed	Communication
17/05/13 MP visit	Elected representative	Visit to ENRMF by Andy Sawford, MP for Corby and East Northants
06/06/13 Newsletter	Public	Augean Community Newsletter distributed to all households and businesses in King's Cliffe, Duddington, Fineshade, Easton on the Hill, Collyweston, Tixover, Apethorpe, Fotheringhay, Woodnewton, Thornhaugh and Wansford. Multiple copies sent to 47 town and parish councils and parish meetings in the 10k consultation area. Copies sent to elected representatives and members of KCLG and TLG
12/06/13 Liaison Group meeting	KCLG	King's Cliffe Liaison Group meeting
14/06/13 Community feedback	Public	Topic sheet about Community Funds
17/06/13 Letter	KCLG TLG	Letter to explain ENRMF acceptance procedures following Sellafield prosecution and fine for disposing of non-compliant radioactive waste into landfill at Lillyhall.
28/06/13 Engineering Days	Public	Publicise forthcoming Engineering Days inviting community to see the construction of new landfill cell at ENRMF. News Release sent to local newspapers
11/07/13 Enewsletter	Public KCLG TLG	Email to stakeholders to inform them of Secretary of States' decision to grant a Development Consent Order
11/07/13 News Release	News Media	News release about grant of Development Consent Order at ENRMF sent to local , regional, and national new media – print and broadcast
11/07/13 Radio interview	News Media	Dr Gene Wilson interviewed by BBC Radio Northants, Heart FM, ITV Anglia and BBC Look East about grant of Development Consent Order.
19/07/13 Newspaper article	News Media	Article about grant of Development Consent Order published in the Stamford Mercury
19/07/13 Enewsletter	Public	E Newsletter to stakeholders as a reminder about the availability of Engineering Days
25/07/13 Newspaper article	News media	Article about Development Consent Order and Community Fund published in Northants Evening Telegraph
2/08/13 Newspaper article	News media	Articles published in Stamford Mercury and Northants Evening Telegraph about Engineering Days
15/08/13 Enewsletter	Public	Enewsletter to stakeholders -last call for Engineering Days. Engineering Days attended by about 10 members of the public.
30/08/13 Open Day	Public	ENRMF Annual Open Day publicised by flyers distributed to all homes and businesses in Apethorpe, Barrowden, Bulwick, Collyweston ,

Date & event	Parties informed	Communication
		Duddington, Easton on the Hill, Fineshade, Fotheringhay, Harringworth, King's Cliffe, Laxton, Nassington, Tixover, Wakerley, Woodnewton and Yarwell. Multiple copies sent to 47 Parish Councils within 10K of ENRMF. E Newsletter sent to stakeholders. The flyer included a letter about the grant of the Development Consent Order. News Release sent to local papers
4/09/13 Community feedback	Public	Topic Sheet Soil Treatment Plant sent to stakeholders
5/09/13 Minerals and Waste Local Plan	NCC	Start of consultation on NCC proposed submission of partial review of Mineral and Waste Local Plan
14/09/13 Open Day	Public	ENRMF Open Day – attended by about 10 members of the public – principally representing Parish Councils that had not been previously actively involved with ENRMF. Tours of the site including the laboratory and soil treatment plant were available as well as relevant information boards and monitoring data. In addition to Augean staff representatives from the Environment Agency were available to answer questions.
16/09/13 Complaint	Public	Response to complaint made about lorry turning out of ENRMF and crossing central line. Undertaking by Simon Moyle to review turning space at site entrance.
16/10/13 University visit	Educational Visit	Visit by Department of Chemical and Environmental Engineering, University of Nottingham staff and students.
22/11/13 Liaison Group	Liaison Group	Thornhaugh Liaison Committee Operational issues and Mick George application to alter hours of working were discussed
26/11/13 University visit	Educational Visit	Visit by University of Northampton staff and students
28/11/13 Meeting	Liaison Group	Follow up on –site meeting at Cooks Hole/Thornhaugh Landfill site to resolve issues arising from Mick George application to increase working hours. Local residents agree to withdraw objections.
5/12/13 Minerals and Waste Local Plan	NCC	End of NCC consultation on partial review of Minerals and Waste Local Plan. As a result of representations made Augean invited to take part in forthcoming public examination.
6/12/13 Monitoring	Public	ENRMF updated monitoring data available on website
06/12/13 Liaison Group	Liaison Group	King's Cliffe Liaison Group meeting. The completed construction of cell 5B, Environment Agency groundwater monitoring and the Community Fund were discussed.
9/04/14 Minerals and	NCC	Dr Gene Wilson and Ms Claire Brook for Augean attend NCC partial review of Minerals and Waste

Date & event	Parties informed	Communication
Waste Local Plan		Local Plan hearing held by appointed Planning Inspector to examine soundness of plan.
25/06/14 King's Cliffe Liaison Group	Liaison Group	King's Cliffe Liaison Group meeting. Matters discussed included forthcoming application for Environmental Permit at ENRMF, minor planning application to improve site entrance, Environment Agency major audit of radioactive waste acceptance, management, and operational procedures and Community Funds
15/07/14 Presentation	Public	Dr Gene Wilson and Simon Moyle Presentation on ENRMF to Corby Rotary Club
3/09/14 Thornhaugh Liaison Group	Liaison Group	Meeting to discuss forthcoming planning application for Thornhaugh Landfill Site and application for Environmental Permit as well as to update on operational issues and site monitoring.
3/09/14 Community Consultation	Public	Community Consultation Event about forthcoming application at Thornhaugh Landfill site to be held in Village Hall Wansford
6/10/ 2014	Environment Agency	Permit application for Thornhaugh Landfill Site submitted to the Environment Agency by Augean

Appendix D. Policy statement and integrated management system

HEALTH, SAFETY, QUALITY AND ENVIRONMENT POLICY

Augean PLC
Specialist services focused on managing hazardous wastes

This policy provides a framework of goals as part of our commitment to reduce our effects on the environment and to ensure the health, safety and welfare of our personnel, stakeholders, contractors, visitors and the public as well as maintaining client satisfaction through service excellence, across the Group. The policy is driven from top level in the Group through Directors and Managers to every employee.

We strive to achieve our health, safety, quality and environmental commitments by complying with applicable legal requirements and through following these key principals:

Health and Safety

- Recognising that our employees are our greatest asset and their health and safety is a top priority for the Group
- Ensuring the health and safety risks arising from our activities are well controlled and injuries and ill health are prevented
- Sustaining a safe and healthy working environment by providing and maintaining appropriate plant and equipment; providing safe systems of work and ensuring safe storage, use, handling and transport of substances
- Providing all required instruction, information, training, supervision and other relevant health and safety information to employees, visitors and contractors to ensure health and safety risks arising from our activities are controlled and injuries and ill health are prevented
- Making available, as necessary, safety and protective equipment at no cost to employees
- Complying with all applicable legal and other requirements.
- Preventing injury and ill health to employees and others who may be affected by our activities.
- Engaging and consulting with employees on day-to-day health and safety conditions and providing advice and supervision on occupational health.
- Maintaining effective emergency response procedures for potential incidents including, but not limited to, fire, major spillages or uncontrolled emissions.

Quality

- Applying a consistent management focus on quality including monitoring performance
- Motivating our employees to take ownership of their work and communicating the importance of customer satisfaction
- Understanding our customers' goals, embracing them and delivering to their expectations
- Providing ongoing training to advance the skills of our employees
- Identifying and solving problems to avoid compromising the quality of our services.

Environmental

- Setting clear objectives and regularly monitoring progress against them
- Recognising that the minimum acceptable level of environmental performance is that stipulated in environmental legislation
- Seeking to avoid and reduce the pollution of air, water and land that may result as a consequence of our activities
- Promotion of sustainable transport alternatives to, from and between Augean sites
- Ensuring that activities and building developments are sensitive to visual amenity and the local community, and the impact on ecology and wildlife habitats is benign, if not beneficial
- Providing suitable environmental training for appropriate personnel and promoting the general environmental awareness to all staff

Operational improvement and corporate objectives shall be set on an annual basis and our performance is monitored through audits and inspections. We pursue a programme of continuous improvement in all aspects of our business to achieve this high level of regulatory compliance and client satisfaction.

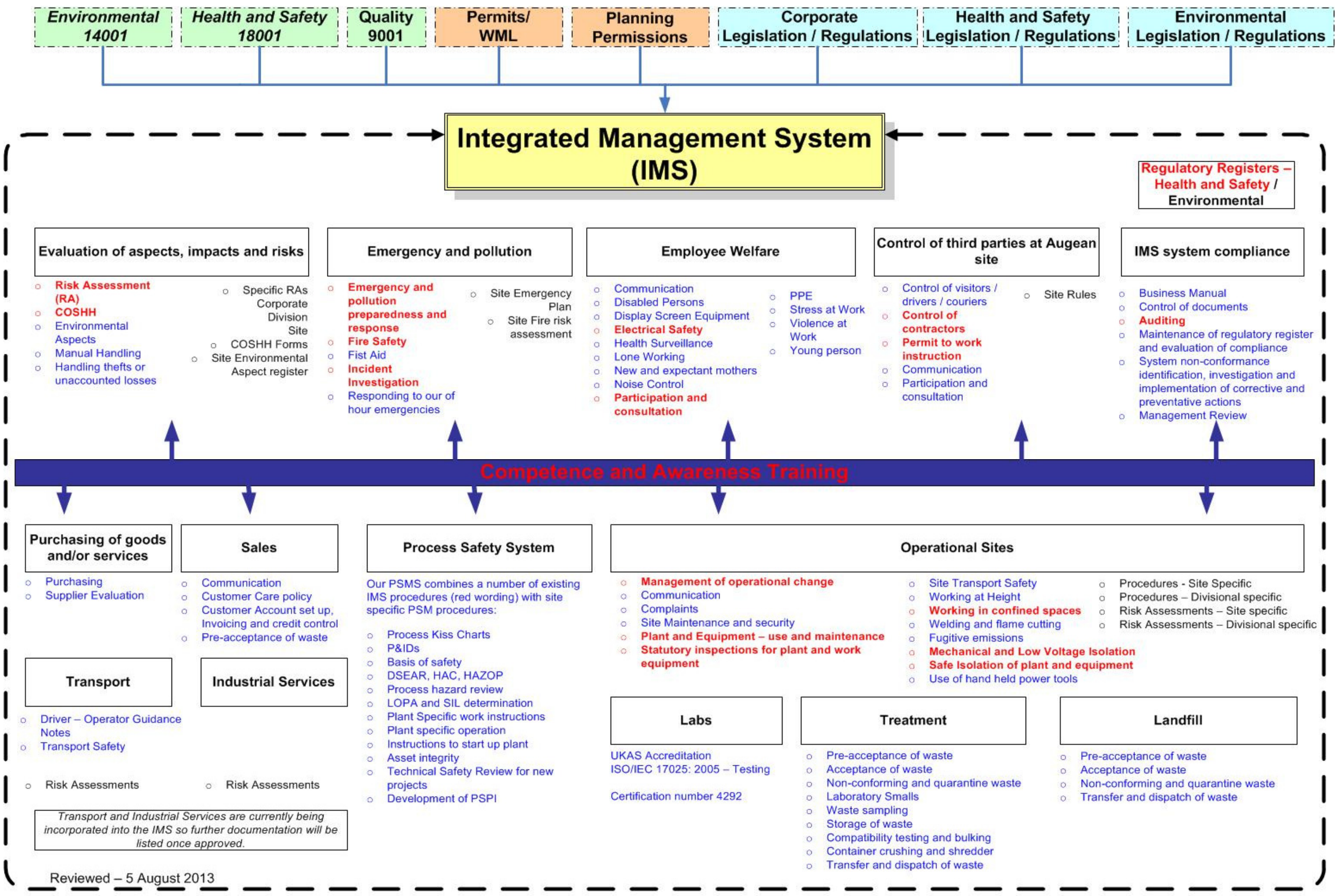
Our Directors are committed to protecting and improving the working environment and employee health and safety by seeking continuous improvements and periodic review of our management policies and objectives.

Each employee has a responsibility for their own safety and that of fellow employees and visitors, along with the obligation to meet environmental regulations, and provide a quality service to our customers

Delivery of this policy is a business priority. Consistent with our whistleblowing policy, we encourage employees who have any concerns regarding compliance with this policy to report this directly, and if necessary anonymously, to Gene Wilson (the Management Board champion for this policy), who will investigate the matter confidentially.

[redacted]

Dr Stewart Davies
Chief Executive Officer
1 July 2014


Environmental 14001
Health and Safety 18001
Quality 9001
Permits/ WML
Planning Permissions
Corporate Legislation / Regulations
Health and Safety Legislation / Regulations
Environmental Legislation / Regulations
Integrated Management System (IMS)
Regulatory Registers – Health and Safety / Environmental
Evaluation of aspects, impacts and risks
Risk Assessment (RA)
COSHH
Environmental Aspects
Manual Handling
Handling thefts or unaccounted losses
Specific RAs Corporate Division Site
COSHH Forms
Site Environmental Aspect register
Emergency and pollution
Emergency and pollution preparedness and response
Fire Safety
Fist Aid
Incident Investigation
Responding to our of hour emergencies
Site Emergency Plan
Site Fire risk assessment
Employee Welfare
Communication
Disabled Persons
Display Screen Equipment
Electrical Safety
Health Surveillance
Lone Working
New and expectant mothers
Noise Control
Participation and consultation
PPE
Stress at Work
Violence at Work
Young person
Control of third parties at Augean site
Control of visitors / drivers / couriers
Control of contractors
Permit to work instruction
Communication
Participation and consultation
Site Rules
IMS system compliance
Business Manual
Control of documents
Auditing
Maintenance of regulatory register and evaluation of compliance
System non-conformance identification, investigation and implementation of corrective and preventative actions
Management Review
Competence and Awareness Training
Purchasing of goods and/or services
Purchasing
Supplier Evaluation
Sales
Communication
Customer Care policy
Customer Account set up, Invoicing and credit control
Pre-acceptance of waste
Process Safety System
Our PSMS combines a number of existing IMS procedures (red wording) with site specific PSM procedures:
Process Kiss Charts
P&IDs
Basis of safety
DSEAR, HAC, HAZOP
Process hazard review
LOPA and SIL determination
Plant Specific work instructions
Plant specific operation
Instructions to start up plant
Asset integrity
Technical Safety Review for new projects
Development of PSPI
Operational Sites
Management of operational change
Communication
Complaints
Site Maintenance and security
Plant and Equipment – use and maintenance
Statutory inspections for plant and work equipment
Site Transport Safety
Working at Height
Working in confined spaces
Welding and flame cutting
Fugitive emissions
Mechanical and Low Voltage Isolation
Safe Isolation of plant and equipment
Use of hand held power tools
Procedures - Site Specific
Procedures – Divisional specific
Risk Assessments – Site specific
Risk Assessments – Divisional specific
Transport
Driver – Operator Guidance Notes
Transport Safety
Risk Assessments
Industrial Services
Risk Assessments
Transport and Industrial Services are currently being incorporated into the IMS so further documentation will be listed once approved.
Labs
UKAS Accreditation
ISO/IEC 17025: 2005 – Testing
Certification number 4292
Treatment
Pre-acceptance of waste
Acceptance of waste
Non-conforming and quarantine waste
Laboratory Smalls
Waste sampling
Storage of waste
Compatibility testing and bulking
Container crushing and shredder
Transfer and dispatch of waste
Landfill
Pre-acceptance of waste
Acceptance of waste
Non-conforming and quarantine waste
Transfer and dispatch of waste
Reviewed – 5 August 2013

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 (Environment Agencies, 2009), para 7.2.8

394. This appendix considers the radiological aspects of an Environmental Safety Case (ESC) for the proposed revision to the permit for receipt and disposal of radioactive waste at East Northants Resource Management Facility (ENRMF), Stamford Road, King's Cliffe, Northamptonshire, PE8 6XX (the centre of the site lies approximately at OS Grid Reference TF 008 000; 0^{O}30ˈ46" W 52^{o}35ˈ18" N).

E.1. Features, events and process

395. 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 - it is not meant to infer a set of possible future conditions as used elsewhere (LLWR, 2011b). 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.

396. The ESC supporting the previous permit application (Augean, 2009a) was reviewed and provided an initial set of scenarios based on consideration of features, events and processes that could lead to exposure of people.

397. The IAEA's Improvement of Safety Assessment Methodologies for Near-Surface Disposal Facilities (ISAM) project (IAEA, 2004) identifies relevant FEPs. This and recent Eden-NE experience with the LLWR safety case and involvement with work on Environment Agency landfill assessment methodologies has been used to supplement the initial set of scenarios.

398. Important features of the ENRMF are described in the rest of this section followed by a summary of the scenarios in Section E.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). Biota exposure is considered in Section E.6. The scenarios that are considered

in the ESC are based on the identified events and the assessment models consider the appropriate processes.

399. 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).

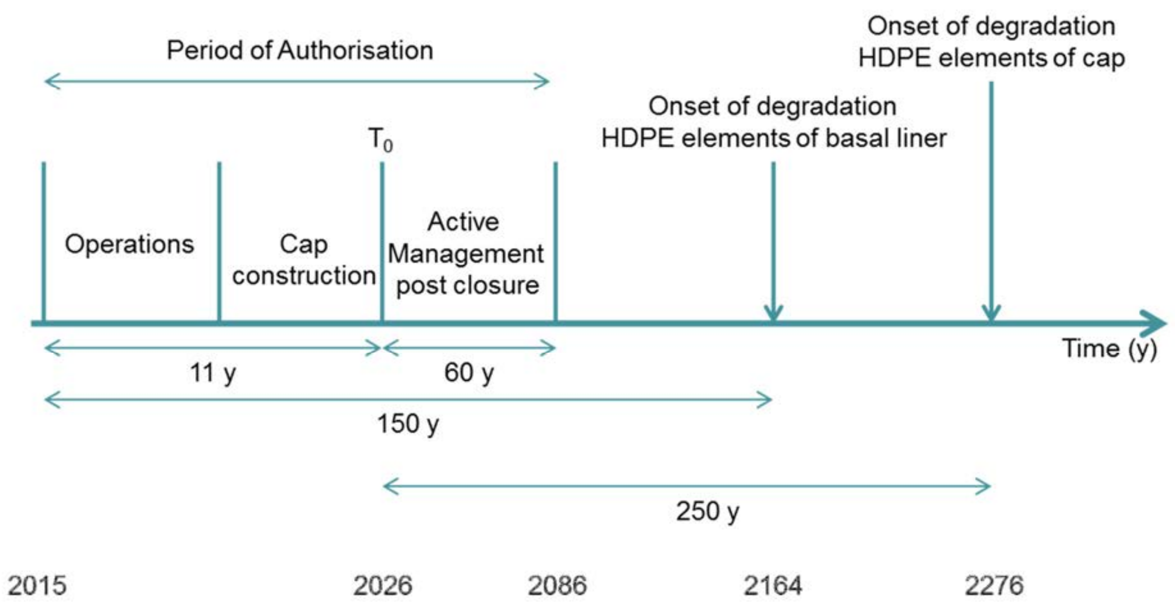
The impact of leaching from the landfill to groundwater is assessed using a model implemented in GoldSim (GoldSim Technology Group, 2013). The models are described in Sections E.3.4.

E.1.1. Period of authorisation for the ENRMF

400. Figure 15 presents the timeline for the ENRMF. This timeline is based on dates from the HRA (Augean, 2014) and the site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013). The figure identifies the Period of Authorisation (POA, the period during which the facility holds a permit), the period of operation, the time of cap construction and the period of active management following cap construction.

401. 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.

Figure 15. Timeline for the ENRMF



402. Figure 15 also illustrates assumptions regarding the onset of degradation of the HDPE elements of the basal liner and the cap based on the HRA (Augean, 2014).

403. During the “operational period” assumed to last until 31/12/2026, 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, 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 2026 to 2086, 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.

404. 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

405. The landfill site will continue to be operated on the principle of containment (Augean, 2012a). This means that the site will be lined with an engineered low permeability barrier designed to retain contaminants within the site. The landfill will be operated in a series of cells which are filled, capped and restored progressively. To separate the restored surface of the site from the wastes and to minimise the infiltration of rainfall the landfill will be capped with low permeability layers overlain with restoration materials.

406. The dimensions of the currently permitted landfill were taken from the 2009 assessment (Augean, 2009a), and the Western extension data were taken from the HRA (Augean, 2014). Site dimensions are shown in Table 33 and a plan of the site is presented in Figure 1. The basal area of the total site is the sum of the basal areas of the currently permitted site and the Western Extension. The surface area over the total site is the sum of the surface areas of the currently permitted site and the Western Extension.

Table 33 Dimensions of the landfill

Component	Basal area (m^2)	Surface area (m^2)	Waste Thickness (m)
Current permitted site (cells 4B, 5A and 5B)	27,775	34,108	15.5
Western extension	82,960	112,000	11.1
Total for LLW permit	110,735	146,108	13.5

407. No distinctions between the disposal cells are made for the radiological assessment. Hence the currently permitted site and the western extension are treated as a single unit that receives the radioactive waste.

E.1.3. Barrier engineering

408. 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;
- an in situ clay barrier (2.0m) of low permeability Rutland Formation; and,
- dilution of the flux of released radionuclides when it enters the Lincolnshire Limestone unit which underlies the facility.

Engineered Cap

409. The engineered cap has a layered construction designed to prevent water from entering the waste cells. In accordance with the HRA (Augean, 2014), 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 4.97 mm y^{-1} (Augean, 2014). 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 74.3 mm y^{-1}) after 1,000 years (Augean, 2014).

Basal Liner and Clay Barrier

410. 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 (Augean, 2014). The same assumptions are used in the radiological assessment.

411. The efficiency of the HDPE component of the basal liner is determined by the number of defects (pinholes, holes and tears) that are present.

412. The engineered clay barrier in the Western Extension is 1 m thick, has a low hydraulic conductivity (ranging from $6.9\ 10^{-11}$ to $1.0\ 10^{-9}$ m s^{-1}) (Augean, 2014), 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 current cell (5B) is 1.5 m thick with a hydraulic conductivity of less than $3\ 10^{-10}$ m s^{-1}.

413. When the HDPE component of the basal liner has degraded, outflow through the base of the landfill is controlled by the clay barrier, comprising the engineered clay barrier and the in situ clay barrier. In the HRA (Augean, 2014), the thickness of the low permeability Rutland Formation left in situ is not taken into account. This is a conservative assumption. The thickness of Rutland Formation clay (hydraulic conductivity of $8.86\ 10^{-11}$ m s^{-1}) below the engineered barrier will be 2 m in the Western landfill area. 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. Consideration is given to both the depth of Rutland Formation and the clay barrier for the radiological assessment as discussed in Section E.3.4.1. The combined clay barrier is modelled as 1.5 m thick with a hydraulic conductivity of 8.86 10^{-11} m s^{-1}.

E.1.4. Landfill drainage

414. 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 (Augean, 2014). Until the end of the period of authorisation, leachate is monitored and managed to ensure that leachate levels remain below the regulatory limits. Excess leachate is pumped off and either used in the on-site soil treatment plant, or transported off-site by tanker for treatment and disposal (Augean, 2014).

415. 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 wastes. Soil and soil-like wastes are effective sorption substrates and soil sorption distribution coefficients (K_d) are applied. LLW is not considered an effective sorption substrate and K_d 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.

416. 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

417. As noted in paragraph 92 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 under the current permit to December 2014 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 or descaling residues;
- 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).

418. It is anticipated that any 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

419. An unsaturated zone underlies the landfill comprising Rutland clay and Lincolnshire Limestone. Flow through this zone will be subvertical. A water table exists within the Lincolnshire Limestone at depths of between 5 m and 15 m below ground level in the western landfill area. Flow within the saturated Lincolnshire Limestone is dominantly fracture flow (Augean, 2009a) and is subhorizontal. Significant dilution occurs when radionuclides enter the saturated zone.

E.1.7. Water abstraction points

420. The following paragraphs are taken from the HRA (Augean, 2014) and describe the water abstraction points (receptors) used for groundwater modelling in the HRA.

In accordance with the Environmental Permitting (England & Wales) Regulations 2010 (EPR), the receptor for hazardous substances is the groundwater beneath the landfill. It is stated in the Environment Agency guidance on HRAs for landfills (reference 9) under the "Compliance points for hazardous substances" sub-heading that:

"An input [of hazardous substances to groundwater] is considered to have been prevented if the substance concerned is not discernible in the groundwater above natural background concentrations or a relevant minimum reporting value (MRV) after the immediate dilution as the discharge enters the groundwater."

The compliance point for hazardous substances in groundwater will be at one or more of the boreholes at the down hydraulic gradient edge of the landfill. To model the effects of dilution only in accordance with the guidance for undertaking HRAs contaminant attenuation, dispersion and degradation are not relied on in the saturated pathway for the purpose of calculating the contaminant concentration at the hazardous substance compliance point.

As described in previous HRAs for the site the primary potential receptor for non-hazardous pollutants migrating from the landfill is the groundwater at the site boundary. The secondary receptor closest to the site is an abstraction borehole located approximately 1.2km down hydraulic gradient of the site (Figure HRA 1). The compliance point for non-hazardous substances is the site boundary.

421. The radiological assessment considered the same two locations i.e. an abstraction borehole at the site boundary and one located approximately 1.2 km down hydraulic gradient of the site boundary (about 1.5 km from the centre of the site).

E.2. Identifying scenarios and exposure groups

422. 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.

423. 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 2086.

424. 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 (Augean, 2009a), (SNIFFER, 2006), (IAEA, 2004) and (LLWR, 2011b). In cases where a scenario has not been assessed, because it will not or is very unlikely to occur at the ENRMF, 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 expected to occur and those where it is hard to quantify the likelihood of occurrence (not certain to occur).

425. 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 be used to 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.

426. The exposure groups considered for the period of authorisation are workers at the landfill site and members of the public living close to the site (see Section E.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).

427. A summary of the scenarios and human exposure groups is given below (Table 34) and in the main text (Table 5, Table 9 and Table 14). Exposure of non-human biota is also considered, see Section E.6.

428. This lists the period and expectation that the case will occur, the scenario and the exposed group. Further details of the assumptions and parameters used to describe the exposed groups are presented in the 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 755).

Table 34 Summary of radiological assessment scenarios considered in the ESC

Scenario	Exposed group
Period of Authorisation – expected to occur	
Direct exposure	Worker
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 – not certain to occur	
Leachate spillage	Farming family
Dropped load	Worker
Aircraft impact	Member of public
Barrier failure*	Member of public
Wound exposure	Worker
Exposure due to fire*	Member of public
After the period of Authorisation – expected to occur	
Recreational user	Member of public
Groundwater abstraction	Farming family
Wildlife exposure	Critical species
After the period of Authorisation – not certain to occur	
Water abstraction at site boundary	Farming family
Bathtubbing	Farming family
Gas release	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
Exposure to discrete items	Worker/Resident
Other unlikely events*	

* Not explicitly assessed.

429. The ESC calculates 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 in all scenarios. However, it is recognised that other age groups could be considered for some scenarios exposing members of the public (children, infants and the developing embryo and foetus) and that habit data and dose conversion factors are available for these different age-groups. The SNIFFER models and the LLWR safety case models only consider doses to adults and although the EA initial radiological assessment methodology considers a range of age groups, it is limited in the range of scenarios and radionuclides that are considered.

430. The LLWR safety case (LLWR ESC, 2010) 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 ranged up to about three times larger than those for an adult. These enhancement factors 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.'

431. The previous radiological assessment for the ENRMF also undertook a comparison of calculated doses for exposed individuals in different age groups (Augean, 2009a). This work showed that for the majority of the radionuclides assessed, specific doses to adults are higher than those to infants or children. It was explained that the adult rates of consumption for foodstuffs grown on contaminated soil are sufficiently greater than those for infants and children to off-set the higher dose coefficients for these age groups. In the case of Cl-36, specific doses to children and infants are higher than those to adults, but the difference is less than a factor of 10.

432. Therefore, the ESC has calculated the radiological capacity of the ENRMF based on the impact to adults since they are expected to be limiting in the majority of cases and any increases in doses for other age groups will be small compared with the uncertainty in the assessed doses.

433. 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). Biota exposure is considered in Section E.6.

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

434. 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.

435. The scenarios and relevant exposure pathways considered in this ESC for the period of authorisation are summarised in Table 35. This is followed by a discussion of three other scenarios that are not considered further in the ESC; these are cell excavation, barrier failure and a waste fire during the period of authorisation.

436. The radiological impact of each of the scenarios in Table 35 has been estimated using the approaches described in Sections E.3.2 to E.3.8.

Table 35 Summary of scenarios and exposure pathways during the period of authorisation

Event/scenario	Exposure pathway	Description
Waste receipt, monitoring, transfer and placement: site worker	External irradiation	A worker is exposed to external radiation whilst accepting and disposing of waste.
Release to atmosphere: operational period	Gas (including radon) inhalation	Workers and members of the public exposed to gases emanating from contaminated material in the landfill.
Release to groundwater: operational period	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.
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.
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.
Aircraft impact: member of the public	Inhalation of contaminated dust	Contaminated dust is released by the impact of the aircraft on uncovered waste.
Dropped load: site worker and member of the public	Inhalation of contaminated dust	Dust released from a dropped container is inadvertently inhaled by a site worker and a member of the public.
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.

Exposure from Cell Excavation

437. 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).

Barrier Failure

438. 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. This is a conservative scenario even for a non-hazardous waste site which has less stringent requirements for engineered barriers than a hazardous waste site such as the ENRMF. 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 in the HRAs. The clay component comprises a natural mineral material and therefore will not degrade other than over geological time. It is considered unreasonable to consider this scenario for a hazardous waste site receiving LLW where the construction, operation and monitoring during the period of authorisation will all reduce the possibility of the complete barrier failing 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 Fire

439. Fire is a potential issue at a landfill site that accepts LLW if the LLW is disposed of alongside municipal and other wastes with large amounts of combustible material.

440. Fires in landfill sites can result from the deposition of hot or burning loads of waste or can be associated with the collection and utilisation of methane in landfill gas at sites which accept significant quantities of biodegradable wastes. There will be insignificant amounts of biodegradable or combustible material in the hazardous waste and LLW deposited at the site and soil treated at the treatment facility hence a fire starting in the site as a result of the ignition of combustible material is considered unlikely. The wastes in the landfill, the cover materials, the drainage materials which include shredded tyres, the hazardous waste including the soils to be treated and LLW have an extremely low combustibility. The current waste acceptance criteria for the landfill excludes material with an organic carbon content greater than 6% and flammable wastes are prohibited. It is considered that the potential for a fire in the hazardous wastes and LLW at the site is negligible.

441. The WAC allows for the total organic carbon (TOC) limit to be exceeded on occasion for loads of LLW, overall the TOC limit for the site will not be exceeded and the assumptions and conclusions above remain valid.

442. Any fire could only occur whilst the wastes are not covered with the final capping layer. The lack of biodegradable wastes in the hazardous waste landfill site makes fires very unlikely after the cap is in place. LLW is also covered on a daily basis (within 8 hours) with 0.3 m of non-radioactive waste or material. This reduces the probability of a landfill fire at the ENRMF to a very low level. Whilst the organic content of the LLW may occasionally be higher than the average specified in the hazardous waste WAC, it will not be high enough to lead to overheating and fire as a result of biodegradation, as can occur in composting facilities or non-hazardous waste landfills. As such it is difficult to conceive that the fire scenario included in the SNIFFER model could occur for a hazardous waste type of landfill. It has therefore not been considered in the assessment.

443. Although an aircraft crash could lead to a fire, the fire would mostly consume aircraft fuel and wreckage. The main feature of an aircraft impact which could lead to exposure would be the physical displacement of material and this is considered, see Section E.3.6.

E.3.1. Presentation of dose assessments

444. 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 ($\mu Sv\ y^{-1}\ MBq^{-1}$).

445. The site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013) restricts LLW disposal at the ENRMF to 448,000 t at a maximum specific activity of 200 Bq g^{-1}. This constrains disposal of LLW at the ENRMF to a maximum total of 89.6 TBq ($8.96\ 10^7$ MBq).

446. The maximum inventory that could be disposed of in the site for each radionuclide is therefore the minimum of 89.6 TBq and the overall radiological capacity and is therefore not necessarily the same as the overall radiological capacity. 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, based on these two constraints, andthe dose ($\mu Sv\ y^{-1}$) from disposal of that maximum 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 308).

447. 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 G.

E.3.2. Direct exposure from waste handling and emplacement

448. 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 the ENRMF are not met, waste may need to be quarantined temporarily while deciding on a course of action.

449. 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 $\mu Sv\ h^{-1}$.

450. The exposed group considered for quarantine, waste handling and emplacement is landfill workers (see also Appendix H, Appendix I and Appendix J). These appendices reproduce the calculations presented in the previous ESC [Annexes C and D, (Augean, 2009a)]. The following paragraphs on waste handling (452 to 460) are an extract from the previous ESC (Section 8.1) updated in *italics* to refer to the relevant Appendix in this ESC. 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.

451. The dose criterion used for this scenario is the site criterion of 1 $mSv\ y^{-1}$ for workers.

E.3.2.1. Waste handling

452. Radiation risks to employees from normal operations were reviewed by the HPA [Annex C, (Augean, 2009a)], and the assessment is included here as *Appendix H*. A conservative estimate of the dose to workers as a result of three work activities suggests an annual dose of about 1.1 mSv if the same worker undertook waste receipt, monitoring, transfer and placement in the landfill and worked in the covered waste area. HPA considered it unlikely that the same person would be exposed during all the listed work activities.

453. The waste handling scenario is the external radiation exposure to workers from their occupancy near to a waste package prior to disposal. The SNIFFER model does not include this scenario and it was therefore assessed by the UKAEA [Annex D of (Augean, 2009a)] reproduced here as *Appendix I*.

454. *Appendix I* considers the external radiation dose for a series of cases and package types. The hypothetical worst case is identified to be a flexible type waste container with 200 Bq g^{-1} of Co-60. A flexible container carrying Co-60 at 200 Bq g^{-1} is an unlikely case and another case is included in *Appendix I* to illustrate more typical exposures.

455. The hypothetical worst case dose identified in *Appendix I* is 14.5 µSv h^{-1} measured at a distance of 1 m from the package face. However, the radiation protection advisor (*Appendix H*) has advised that the maximum dose at 1 m from a package should be less than 10 µSv h^{-1} in order to ensure the occupational dose is considerably less than the dose criterion of 1 mSv y^{-1}. Thus 10 µSv h^{-1} will be used as an acceptance criterion and constrains the contents of the package to this limit.

456. The proposed authorisation condition 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 the ENRMF.

457. Additional ALARA precautions are that dose can be measured directly and managed actively to prevent unnecessary exposure. As illustrated in *Appendix I* the field dose drops quickly with distance from the package and hence the simple precaution of managing occupancy time and distance is practicable.

458. This dose is specific to workers during the operational phase and can be managed through occupational radiation dose protection practices, hence it is not used to constrain overall radiological capacity.

459. There is an additional scenario that a member of the public 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, 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}).

460. This gives an estimated maximum dose rate at 50 metres 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}. These calculations do not take into account the significant shielding afforded by the soil screen bund at the boundary of the site.

E.3.2.2. Waste emplacement

461. The waste emplacement scenario considers the external radiation exposure of workers in the vicinity of the waste emplaced in the landfill after it has been covered.

The assessment is by the UKAEA [Annex H of (Augean, 2009a)] reproduced here as *Appendix J*.

462. *Appendix J* illustrates the dose rate for varying cover thicknesses using two illustrative cases, one of which is a worst case. The advice of the radiation protection advisor (*Appendix H*) is that the maximum radiation dose 1 m above the covered waste should be less than 2 μSv h^{-1} in order to ensure the occupational dose is considerably less than the dose criterion of 1 mSv y^{-1}.

463. *Appendix J* demonstrates that for most cases a 0.3 m thick cover layer will more than achieve the dose rate. For the worst case of waste containing Co-60 at 200 Bq g^{-1}, a cover layer of 0.7 m would be required to achieve the dose rate, but this is exceptional.

464. The proposed authorisation condition is that a minimum cover layer of 0.3 m be utilised and that if the dose rate 1 m above the waste is still greater than 2 μSv h^{-1} then further cover will be added in order to achieve the dose rate. The minimum cover layer of 0.3 m is adequate to ensure daily physical protection of the waste.

465. 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.

466. The original SNIFFER model uses occupational external dose as a constraint to set the radiological capacity of the landfill but since this dose is specific to workers during the operational phase and can be managed through occupational radiation dose protection practices this is not considered necessary. Hence the external dose assessment for waste emplacement has not been used to constrain the overall radiological capacity.

E.3.2.3. Wound exposure

467. Exposure due to radionuclides embedded in a wound is relevant to landfill site workers during the pre-closure phase.

468. The scenario has been separately addressed in *Appendix H*, which is a radiological risk assessment for occupational exposure completed by the HPA; it is not considered in the SNIFFER landfill assessment model. The dose criteria used by HPA are the legal limit to workers of 20 mSv y^{-1} and the site criterion of 1 mSv y^{-1} for workers.

469. The radiation risk assessment undertaken by the HPA [Annex C, (Augean, 2009a)], included here as *Appendix H*, considers internal exposure from contaminated wounds (see Section 3.3). The following extract refers:

> *Under normal circumstances this is not a reasonably foreseeable exposure scenario. However, if contamination does arise, for example because of the spill scenario in 3.2 above, then this additional accident exposure pathway becomes a possibility. It is considered that doses from this pathway would be likely to be the same order of magnitude as from inadvertent ingestion, i.e., less than 0.1 mSv.*

The UKAEA Safety Assessment Handbook (UKAEA/SAH/D9, Issue 1, March 2006) gives dose factors for contaminated wounds. Assuming that 0.1 g of material (at 200 Bq/g) becomes incorporated into a wound, the highest estimated dose is approximately 3 mSv, from actinium-227. As mentioned above, this radionuclide is most unlikely to predominate, and it is concluded that internal doses from a contaminated wound would be very unlikely to exceed 1 mSv in practice.

470. HPA concluded that wound exposures are unlikely and can be further reduced in likelihood and impact through simple precautions. Hence, it is very likely that these precautions will be effective in maintaining individual exposures within the site criterion. This scenario is not used to constrain landfill radiological capacity.

E.3.3. Exposure to gas during site operations

471. The permit application involves no specific authorised gaseous discharge routes. 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.

472. Emission of radioactive gases as a result of combustion for power generation or flaring will not occur. Gas collection and combustion is included in earlier capped cells (Augean, 2009a) due to the earlier practice of co-disposing hazardous materials with other waste (pre-permit) but these do not contain radioactive waste. New cells and existing cells containing radioactive waste will contain insufficient putrescible material to require flaring.

473. An aerosol pathway does not arise as leachate is not sprayed on to the landfill. Where leachate recirculation is carried out this comprises reinjection below ground level.

474. Resuspension of dust has not been assessed as all waste is packaged, covered with suitable material before packaging can degrade and a condition for accepting wastes requires low surface contamination of packages which is monitored (Augean, 2011a).

475. 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.3.3.1. Estimating activity concentrations of gas release from disposed waste

476. 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.

477. Radioactive gas, i.e., $^{14}CO_2$, $^{14}CH_4$, ^{3}H, 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, 2011a). 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.

478. 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 some time after the end of the period of authorisation and this is addressed later (see Section E.5.6 and E.5.7). Table 36 details the habit data assumed for the exposed groups during the period of authorisation.

Table 36 Worker and public habit data for exposure to gas releases: applicable during the Period of Authorisation

Parameter	Value	Comment
Inhalation rate – worker ($m^3\ h^{-1}$)	1.2	
Inhalation rate – public ($m^3\ h^{-1}$)	1.0	
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

479. 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

480. 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).

481. The parameters used in this study are summarised in Table 37 and are from (Augean, 2010). The hazardous waste acceptance criteria at the ENRMF 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 CFA permits LLW to contain a greater amount of organic carbon subject to the overall site limit and this has been considered in the section on uncertainty (see Section E.7.1).

482. 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 37 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} C-14: 6.0 10^{-2}	Fraction of activity associated with gas
τ_{gas}	y	10	Average timescale of gas generation

From (Augean, 2010)

483. 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}).

484. 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}).

485. The dispersion parameter values used in the ESC are given in Table 38, the dose coefficients in Table 170 and the habit data in Table 36.

Table 38 Parameter values used in calculations of doses through the gas pathway during site operations

Parameter	Units	Value	Description
W	m	200	Width of source perpendicular to the wind direction
u	m s^{-1}	6.2	Mean wind speed
h	m	2.0	Height for vertical mixing
s_y	s	3.16 10^7	Seconds in a year

486. The wind data for Wittering RAF base indicates wind direction and speed (Table 39). 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 a West South West wind. It assumes that mixing is limited to a height of 2 m and that the width of the source is limited to a cell width based on a current cell minimum. These assumptions are conservative. Wind data (Augean, 2012b) for the meteorological station closest to the ENRMF (Table 39) show that the peak dose, using a combination of wind speed and the prevailing sector, to a member of the public is about 12% of the value calculated assuming that the exposed group is always downwind of the release point.

Table 39 Wind data from RAF Wittering for 2000 to 2009

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	0.0047	0.0088	0.0109	0.0161	0.0058	0.0012	0.0005
NNE	0.0045	0.0086	0.0101	0.0175	0.0079	0.0020	0.0006
NE	0.0039	0.0076	0.0097	0.0169	0.0088	0.0034	0.0016
ENE	0.0034	0.0063	0.0074	0.0114	0.0052	0.0016	0.0005
E	0.0032	0.0048	0.0050	0.0051	0.0021	0.0003	0.0001
ESE	0.0053	0.0067	0.0049	0.0065	0.0021	0.0003	0.0001
SE	0.0078	0.0084	0.0068	0.0069	0.0018	0.0003	0.0000
SSE	0.0074	0.0075	0.0065	0.0105	0.0061	0.0017	0.0007
S	0.0066	0.0085	0.0106	0.0203	0.0135	0.0055	0.0027
SSW	0.0056	0.0093	0.0124	0.0280	0.0228	0.0123	0.0094
SW	0.0057	0.0089	0.0119	0.0279	0.0262	0.0151	0.0112
WSW	0.0049	0.0088	0.0146	0.0378	0.0286	0.0138	0.0095
W	0.0045	0.0099	0.0182	0.0328	0.0174	0.0089	0.0080
WNW	0.0044	0.0101	0.0139	0.0251	0.0158	0.0064	0.0040
NW	0.0046	0.0093	0.0124	0.0214	0.0092	0.0031	0.0007
NNW	0.0042	0.0082	0.0099	0.0157	0.0069	0.0021	0.0004

Gas generation – Radon

487. 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 Pu-242 decay chain (see Figure 11) will also result in radon gas release.

488. 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).

489. 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, 143,447 m^2;
- 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}) see Table 46;
- 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).

490. The activity concentration of radon in outdoor air is calculated using the equation given in paragraph 483 and the parameters in Table 38. The radon calculations for members of the public are adjusted for the wind direction and speed (see paragraph 486).

491. 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 0.7m 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 40 Radon parameters

Parameter	Units	Value	Description	Comment
τ		0.1	emanation factor	
H_1	m	0.2	effective diffusion relaxation length for the waste	
H_2	m	0.2	effective relaxation length of the cover	
h_2	m	0.3	thickness of cover for first two months	daily cover depth
		1.0	thickness of cover for remaining ten months	

From (HPA, 2007)

E.3.3.2. Assessment calculation for gas releases

492. 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}).

493. The dose coefficients for C-14 and H-3 are in Table 170 and the habit data in Table 36.

494. The dose coefficient for radon (Table 41) 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 IAM, see Section 4 in Appendix B (Environment Agency, 2006b). Habit data for workers and members of the public are presented in Table 36.

Table 41 Inhalation dose coefficient for use in calculation of radon doses

Parameter	Units	Value	Description
D_{inh}	Sv Bq^{-1}	6 10^{-9}	Effective dose from Rn-222 inhalation

E.3.3.3. Doses from atmospheric releases

495. 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 calculation assumes that waste is covered on a daily basis to a depth of 0.3 m, and covered again within 2 months, 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.

Table 42 Dose estimated for exposure from gas released during operations

Radionuclide	Dose (µSv y^{-1} MBq^{-1})	
	Worker	Public
H-3	$1.37\ 10^{-8}$	$5.08\ 10^{-9}$
C-14	$4.70\ 10^{-7}$	$1.74\ 10^{-7}$
Ra-226	$3.37\ 10^{-6}$	$2.08\ 10^{-7}$

496. 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 scenario has the potential to constrain the radiological capacity of the ENRMF. The results are independent of the Ra-226 placement depth in the site.

497. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.3.

E.3.4. Groundwater pathways

498. The release of radionuclides to groundwater during the period of authorisation is limited by controlling the accumulation of leachate at the bottom of the waste cells. Although no doses are expected to members of the public off-site from groundwater pathways during the period of authorisation this is assessed.

499. Exposure of members of the public is assumed to occur as a result of abstracting contaminated groundwater in a well and using it for drinking and for irrigation. Irrigation leads to contamination of soil resulting in doses from ingestion of foodstuffs grown on the soil (including pasture supporting grazing livestock), inhalation of dust from the soil and external exposure to the soil. This scenario considers the exposures resulting from contaminated groundwater taken from the boundary of the landfill site.

500. Doses to workers on site managing the level of leachate is covered by the operational safety case and the assessment of leachate processing off-site. The dose criterion for workers on site is the site criterion of 1 mSv y^{-1} for workers and 20 µSv y^{-1} for members of the public.

Implementation of a groundwater model using GoldSim

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

- it provides a flexible modelling environment;
- it applies a linear distribution model of sorption of radionuclides in an equilibrated environment with water and solid compounds (e.g. leachate in waste cells);
- decay and ingrowth of radionuclides can be modelled in the standard application; and,
- models for well-mixed compartments and one-dimensional contaminant transport are available in the Contaminant Transport Module (GoldSim Technology Group, 2013).

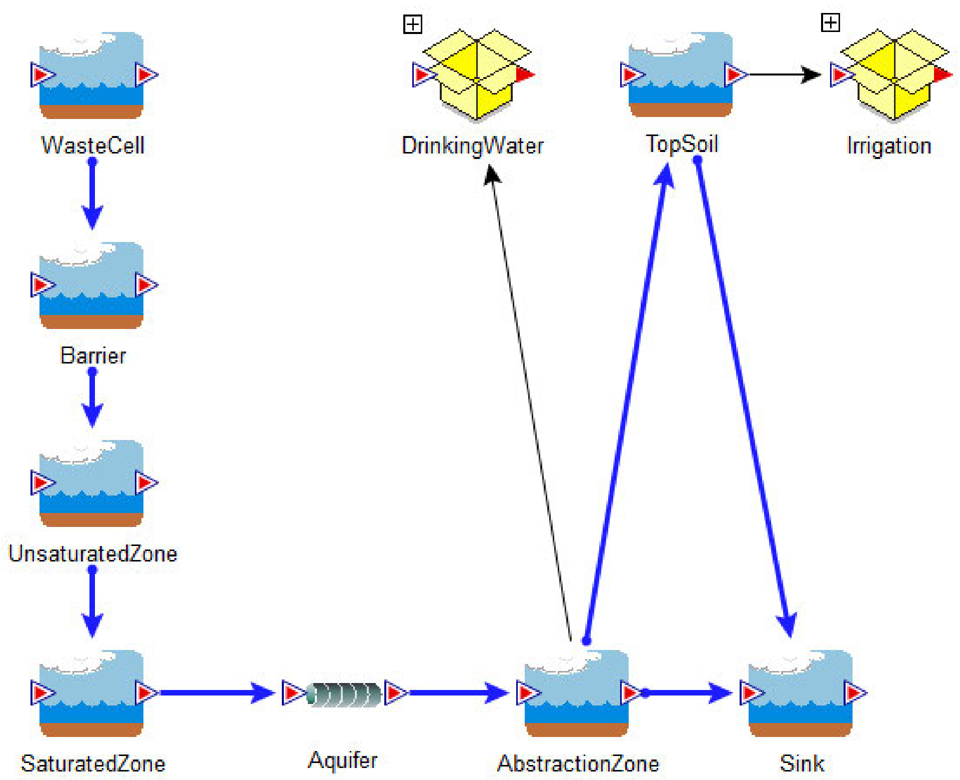
502. The model has been developed on the basis of the conceptual model set out in Section E.1.4. Calculations have been undertaken for an area comprising the currently permitted landfill and the western extension. The flexibility of the GoldSim model allows a range of sensitivity analyses to be performed.

503. Calculations have been undertaken of the activity concentration in the groundwater at two locations:

- at the site boundary, which is 37 m from the edge of the landfill for the western extension; and,
- at 1500 m from the centre of the waste cells used for LLW, corresponding to an existing permitted abstraction well (Augean, 2014).

504. The structure of the GoldSim model is shown in Figure 16. 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 16. Compartments as modelled in GoldSim.



505. Table 1 lists the radionuclides of interest with their half-lives, short-lived daughters where applicable and radioactive daughters considered explicitly. The list is based on those radionuclides included in the current permit for the site with the addition of Ra-228. 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,
- $A_{Rn,Comp}$ is the activity of radionuclide *Rn*.

506. Decay systems corresponding to a number of radionuclide chains are illustrated in Figure 10 to Figure 13. Short-lived daughters that are assumed to be in secular equilibrium with a longer-lived parent radionuclide have been omitted from the figure.

507. In all of 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 four 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; and,
- U-234 ingrown from Pu-242.

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

Confidence building for the GoldSim model

509. A simple model was constructed from first principles to verify the more complex model in GoldSim. This involved development of the differential equations describing mass balance in the waste cell, the clay barrier and the relevant section of the aquifer. The model included sorption to soil and clay, radioactive decay and leaching. The differential equations were then solved numerically in the Gnu Octave environment. The results were very similar to the corresponding calculations in GoldSim (see Appendix F for more details).

510. Internal consistency within GoldSim was verified by comparing the results of models using a “Pipe” model and using an “Aquifer” model. The pipe model uses a Laplace transform to solve the one-dimensional transport equation, while the aquifer model represents the pipe as a series of compartments. GoldSim indicates when the number of cells in the aquifer model is insufficient to represent the length and dispersivity of the pathway. If no issues were raised, both models gave the same results.

E.3.4.1. Waste cells

511. The engineered cap and the waste cell design are discussed in Section E.1.3. In this section, more details are given on the relevant equations and on the parameter values used in the 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

512. Calculations were undertaken for a nominal disposal inventory of 1 MBq of each radionuclide, distributed evenly through the landfill. As all radiological impacts associated with the groundwater pathway scale with the disposed inventory (noting that saturation effects are ignored in the calculations), the results of these calculations serve as a basis for calculation of the radiological capacity for disposal of specific radionuclides. The radiological impact from the disposal of two illustrative waste streams is presented in Appendix G.

Water flux

513. 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.

514. The infiltration flux through the cap, $q_{Infiltration}$, ($m^3\ y^{-1}$) is 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} ($m\ y^{-1}$) represents the effective infiltration into the waste cell, defined as:

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

515. As long as cap degradation has not started (i.e. before $t_{StartCapDegradation}$), the cap design infiltration 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 $t_{EndCapDegradation}$ (Augean, 2014).

516. The parameters used to calculate the effective infiltration have been assigned values as defined in Table 43. All these parameter values are taken from the HRA (Augean, 2014).

Table 43 Parameters to calculate the effective infiltration through the cap.

Parameter	Description	Value
P_{Cap}	Cap design infiltration	4.97 mm y^{-1}
$P_{Grassland}$	Infiltration to grassland (between 4.73 and 74.3 mm y^{-1})	74.3 mm y^{-1}
$t_{StartCapDegradation}$	Start of cap degradation	250 y
$t_{EndCapDegradation}$	End of cap degradation	1,000 y

517. We have adopted the same formula for the potential flux through the HDPE liner as in the LandSim model (Golders Associates, 2003). The maximum water flux q_{Liner} through the basal liner (m^3 y^{-1}) is defined in LandSim as:

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

with:

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

518. The initial number and type of defects are as defined in Table 44 and other parameters defined in Table 45. 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 44 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 45 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	1.5	Leachate head
$K_{Barrier}$	m s^{-1}	8.86 10^{-11}	Hydraulic conductivity of the clay barrier

519. The parameter values in Table 45 were taken from (Augean, 2014), 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).

520. The number of defects is assumed to double every 100 years (Augean, 2014).

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

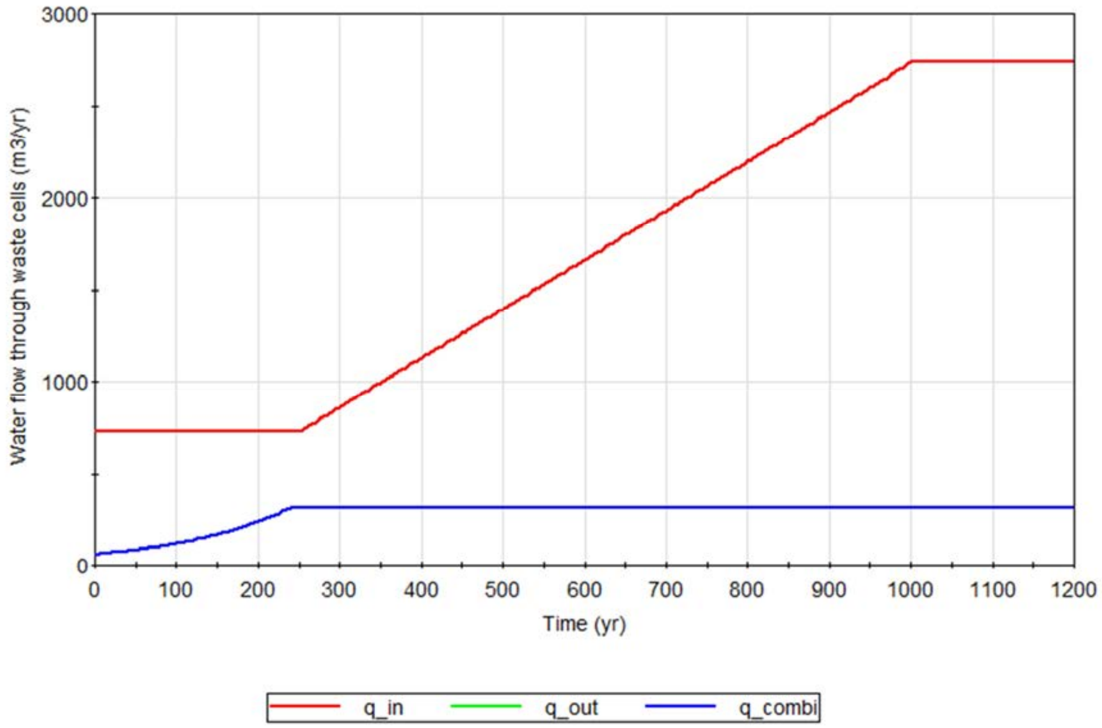
$$q_{Barrier} = K_{Barrier} \cdot A_{Basal} \cdot 3.16E+07$$

with A_{Basal} the basal area (m^2) of the landfill component being considered and $3.16E + 07$ is the number of seconds in a year.

522. We have 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}.

523. Due to differences in surface area, the water flow through the current landfill differs from the water flow through the western extension. The water flow through the whole landfill is shown in Figure 17. In this figure, 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 17. Water flux through the waste cells for the total site



524. Until the end of the management phase, leachate is monitored and managed to ensure that leachate levels remain below the regulatory limits. Excess leachate is pumped off and either used in the adjacent treatment plant or transported off-site by tanker for treatment (Augean, 2014).

525. Properties of waste and filling materials are given in Table 46. The density, porosity and hydraulic conductivity of waste and clay were taken from the HRA (Augean, 2014) and the density of soil was taken from the previous assessment (Augean, 2009a). The porosity of soil was assumed to be 0.5.

Table 46 Proportions and properties of waste and filling materials

Material	Density ($kg\ m^{-3}$)	Porosity	Hydraulic conductivity ($m\ s^{-1}$)
Waste	1,530	0.1	Not used
Soil (incl. soil type waste)	1,300	0.5	Not used
Clay	1,640	0.2	$8.86\ 10^{-11}$

526. The sorption distribution coefficients (K_d's) for the filling materials are given in Table 169. The sorption distribution coefficient (K_d) for Cl-36 was modified; the default value in SNIFFER is high and results in unrealistically low groundwater activity concentrations. The revised values, $2\ 10^{-4}\ m^3\ kg^{-1}$ for clay and $3\ 10^{-4}\ m^3\ kg^{-1}$ for soil are from the review presented in TecDoc 1616 (IAEA, 2009) which supports the IAEA handbook of parameter values (IAEA, 2010).

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

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

where:

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

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

529. 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:

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

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

Clay barrier

531. 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. The clay barrier is a combination of the engineered clay barrier and the underlying in situ Rutland Formation clay.

532. The dimensions of the barrier for the different calculation cases are defined in Table 47. The barrier thicknesses for the currently permitted site and the western extension have been taken from the HRA (Augean, 2014). The clay barrier thickness under the current permitted site is 1.5 m thick. In the western area, 2 m of Rutland Formation clay will be left in situ beneath a clay barrier at least 1 m thick. Tests indicate the hydraulic conductivity of the underlying Rutland Formation is $8.86\ 10^{-11}$ $m\ s^{-1}$ (geometric mean). This is close to the geometric mean for cells constructed using the same clay in other parts of the ENRMF site ($4.84\ 10^{-11}\ m\ s^{-1}$) and at the nearby Thornhaugh Landfill ($6.9\ 10^{-11}\ m\ s^{-1}$).

533. A clay barrier thickness of 1.5 m has been selected for the radiological assessment, which combined with the hydraulic conductivity ($8.86\ 10^{-11}\ m\ s^{-1}$) provides conservative assumptions for water flow through the base of the waste cells.

Table 47 Dimensions of the barrier assumed in the assessment

Site	Basal area (m^2)	Clay barrier Thickness (m)
Current permitted site (cells 4B, 5A and 5B)	27,775	1.5
Western extension	82,960	1.5*
Total site	110,735	1.5
* The clay barrier is actually 1 m of engineered clay and 2 m of natural clay		

534. 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.

535. The barrier is constructed out of boulder clay and clay from the Rutland Formation. The properties and K_d values for clay have been defined in Table 46 and Table 169.

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

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

537. 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 limestone zone. The subscript index *GW* indicates that the equation includes the contributions from groundwater movement, and not radioactive decay. Radioactive decay and ingrowth is 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,
- $V_{Barrier}$ the volume of the clay barrier (m^3).

E.3.4.2. Unsaturated Limestone zone

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

539. The dimensions of the unsaturated zone for the different calculation cases are defined in Table 48. The thickness of the unsaturated zone for the currently permitted site is taken from the HRA (Augean, 2014). For the western extension, the model uses a modified thickness of the unsaturated zone to account for the different assumption regarding the thickness of the clay barrier (our calculations are based on 1.5 m clay; the HRA calculations use 1 m clay). Maintaining the same distance below the liner to the saturated zone results in an unsaturated zone thickness of 8.99 m rather than the HRA value of 9.49 m. The minimum thickness over the site has been selected for the total site for the radiological assessment as a conservative assumption.

Table 48 Dimensions of the unsaturated zone used in the model

Site	Basal area (m^2)	Thickness (m)
Current permitted site (cells 4B, 5A and 5B)	27,775	8.68
Western extension	82,960	8.99
Total site	110,735	8.68

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

541. The unsaturated zone consists of Lincolnshire Limestone. This rock is fractured and fracture flow is assumed to occur. The properties of Lincolnshire Limestone are summarised in Table 49 (Augean, 2014) and are from (British Geological Survey and the Environment Agency, 2006). The porosity given is fracture porosity, and is appropriate to use for fracture flow. It has been assumed that sorption in this zone is insignificant on the basis of the assumed fracture flow.

Table 49 Proportions of Lincolnshire Limestone

Material	Density (kg m^{-3})	Porosity	Hydraulic conductivity (m s^{-1})
Lincolnshire Limestone	2,000	0.007	5 10^{-5}

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

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

543. 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 aquifer. The volume of water in the unsaturated zone ($V_{Water,Unsat}$, m^3) is given by:

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

where:

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

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

E.3.4.3. Aquifer

545. The aquifer consists of Lincolnshire Limestone. This unit is fractured and fracture flow is assumed to occur with no sorption to the rock. In addition to the properties defined in Table 49, the hydraulic gradient in the aquifer is taken as 0.01 [(Augean, 2014) based on (British Geological Survey and the Environment Agency, 2006)].

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

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

where:

- $K_{Limstone}$ is the hydraulic conductivity of Limestone (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).

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

- The saturated zone: this is the volume of saturated limestone 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, e.g. well. The vertical water flux into this zone is assumed to be equal to the water flux out of the unsaturated limestone zone. This contaminated water is also mixed with clean water from the up gradient part of the aquifer. The horizontal water flux in the aquifer is much higher than the vertical water flux into the aquifer.
- 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 edge of the landfill and the well. Given the abstraction zone at a distance D and the initial width of the aquifer W_0, the width of the contaminated zone (W) at the abstraction point is given by the following expression (from SNIFFER).

$$W^2 = W_0^2 + 24 \cdot \frac{W_0}{10} \cdot D$$

- The inflow of contaminated water into the abstraction zone, defined by the width of the aquifer, ensures that all radioactive contaminants pass through the abstraction zone. The outflow is defined by the width calculated from the expression above. GoldSim compensates the difference in flows with uncontaminated water as it assumes a constant volume of water in the cells.
- 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.

Saturated Zone

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

549. The dimensions of the saturated zone for the different calculation cases are defined in Table 50. The thicknesses of the unsaturated zone for the currently permitted site and the western extension have been taken from the HRA (Augean, 2014) and are the same value. This minimum thickness has been selected for the total site.

Table 50 Dimensions of the saturated zone

Site	Basal area (m^2)	Saturated zone Thickness (m)	Width of aquifer perpendicular to flow direction (m)
Current permitted site (cells 4B, 5A and 5B)	27,775	7.42	416
Western extension	82,960	7.42	255
Total site	110,735	7.42	671

550. The water flux into the saturated zone has two components; the vertical water flux through the unsaturated zone, the barrier and the waste cells and the horizontal groundwater flux.

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

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

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

553. The first term relates to the flux from the unsaturated zone into the saturated zone right beneath the waste cell. The second term relates to the flux from the saturation zone horizontally away from the landfill. The volume of water in the saturation zone ($V_{Water,Sat}$, m^3) is given by:

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

where:

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

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

Aquifer Transport Zone

555. The aquifer transport zone is the region of the aquifer between the saturated zone beneath the waste cells and the abstraction zone, where the abstraction well is located.

556. The dimensions of the aquifer transport zone for the different calculation cases are given in Table 51. As noted previously, the length of the pathway has been taken to be equivalent to the maximum length considered in the HRA.

557. The "aquifer" model in GoldSim represents a one-dimensional transport pathway as a series of cells. Advection and dispersion in the aquifer is modelled, together with radioactive decay and ingrowth. A sufficiently high number of cells are required to represent the pathway length and dispersivity appropriately.

Table 51 Dimensions of the aquifer transport zone

Site	Length of Aquifer Pathway (m)	Width of Aquifer Pathway (m)	Thickness of Aquifer (m)
Abstraction well at the site boundary (after regulatory control) (Augean, 2014)			
Current permitted site	40	416	7.42
Western extension	52	255	7.42
Total site	40	671	7.42
Abstraction at an existing permitted abstraction well (Augean, 2009a)			
Current permitted site	1500	416	7.42
Western extension	1500	255	7.42
Total site	1500	671	7.42

Abstraction Zone

558. The abstraction zone is the section of the aquifer in which an abstraction well is 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.

559. The dimensions of the abstraction zone for the different calculation cases are defined in Table 52. 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 does not significantly affect the results because a strong gradient is not expected. An arbitrary value of 10 m is chosen. The width of the abstraction zone is calculated using the transversal diffusion equation given above.

Table 52 Dimensions of the abstraction zone

Site	Length of abstraction zone (m)	Thickness of abstraction zone (m)
Current permitted site	10	7.42
Western extension	10	7.42
Total site	10	7.42

560. 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 flux out is the flux abstracted by the well and the remaining horizontal flux in the aquifer. Radioactive decay and ingrowth is also applied, separately.

E.3.4.4. Assessment calculations for groundwater used for drinking water and irrigation

561. The contamination of groundwater under the landfill is expected to occur at some point in the future. The HRA shows degradation of the landfill liner and cap over time resulting in leachate flows to the underlying substrate and then to groundwater. This scenario considers the exposures resulting from contaminated groundwater taken from a hypothetical abstraction point at the site boundary. Although the presence of an abstraction point at the site boundary is not credible during the period of authorisation, this calculation enables protection of groundwater to be demonstrated.

562. If the 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. However, groundwater does not discharge to a watercourse that is closer to the landfill than the abstraction point; any discharges to a more distant watercourse would be subject to additional dilution by groundwater, surface runoff and drainage water thereby reducing exposure relative to the extraction point.

563. 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).

Exposed group

564. Groundwater abstraction is also expected to continue at the nearest borehole to the site and it is assumed to access groundwater within the Lincolnshire Limestone. The nearest licensed water abstraction point in the direction of groundwater flows is at Law's Lawn, about 1.5 km south east of the centre of waste cells used for LLW disposal. Although this has only been used for farm activities in the past, it is currently licenced for potable water. The same exposure pathways are assumed for both this and the hypothetical site boundary abstraction points.

565. Exposure of members of the public is assumed to occur as a result of using well 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.

566. 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 53).

567. 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.

568. 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 97.5^{th} 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.

569. Table 53 details the habit data assumed for an adult farming irrigated land. 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 is used for the remaining pathways.

Table 53 Habit data for the farming family irrigating soil with groundwater: applicable during the Period of Authorisation

Pathway	Adult average	Adult 97.5th	Comment
Milk consumption ($l\ y^{-1}$)	122.5	240	From (Smith & Jones, 2003).
Cow meat consumption ($kg\ y^{-1}$)	15	45	
Sheep meat consumption ($kg\ y^{-1}$)	8	25	
Offal consumption ($kg\ y^{-1}$)	5.5	20	
Green & other domestic veg consumption ($kg\ y^{-1}$)	35	80	
Root veg & potatoes consumption ($kg\ y^{-1}$)	60	130	
Breathing rate ($m^3\ h^{-1}$)	1		From (Augean, 2009a)
Inadvertent soil ingestion ($kg\ y^{-1}$)	0.03		
Occupancy ($h\ y^{-1}$)	8,760		Standard assumption in (Environment Agency, 2006b).
Indoor shielding factor	0.1		
Fraction of time spent indoors	0.75		From (Augean, 2009a)

570. The GoldSim model used to model the groundwater migration scenario also includes a soil compartment which receives inputs from irrigation water 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 $m\ y^{-1}$ in this area of Northamptonshire. 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.

571. The peak activity concentration in the groundwater over the period of authorisation (60 years) is used to calculate the doses to the exposed group.

Use of Groundwater as Drinking Water

572. 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} is the drinking water consumption rate ($l\ y^{-1}$);

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

573. Drinking water consumption rate for adults taken from (Smith & Jones, 2003) is 600 l y^{-1}. The activity concentrations of radionuclides in irrigation water are determined by the groundwater transport model outlined above (Section E.3.4.3).

574. The dose conversion factors are set out in Table 170 for all radionuclides except Ra-226 and Th-232 which use the values shown below (Table 54). The dose conversion factor for Ra-226 in Table 170 includes the radiological impact of its daughter Pb-210 and subsequent daughters. However, as Pb-210 is modelled explicitly in groundwater, a Ra-226 dose conversion factor is required that does not include a contribution from Pb-210 and subsequent daughters. Similarly, a Th-232 dose conversion factor is required in the GoldSim model that doesn't include contributions from Ra-228 and its daughters. The dose conversion factors for Ra-226 and Th-232 were determined from ICRP data for ingestion and inhalation (ICRP, 1996) and from US EPA data for slab irradiation (US EPA, 1993).

Table 54 Radium 226 and Th-232 dose coefficients used when Pb-210 and Ra-228 are modelled explicitly

Radionuclide	Ingestion (Sv Bq^{-1})	Inhalation (Sv Bq^{-1})	External Irradiation from slab (Sv y^{-1} Bq^{-1} kg)
Ra-226	2.8 10^{-7}	9.53 10^{-6}	3.03 10^{-6}
Th-232	2.3 10^{-7}	1.1 10^{-4}	1.41 10^{-10}

Use of Groundwater for Irrigation of Farmland

575. The compartment in GoldSim that represents the top soil was used to derive the activity concentration in irrigated soil. Groundwater is applied 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 and the infiltration rate of grassland would be more conservative in terms of activity concentration in the soil, the infiltration rate for grassland has been used (Augean, 2014).

576. 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:

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

577. 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:

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

578. 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).

579. Assumptions regarding the top soil compartment, used to calculate the volume of water and the mass of soil, are summarised in Table 55. The area of farmland assumed is arbitrary and doesn't affect the results since it cancels out when the activity concentration in the soil is calculated (see below). Soil properties are taken from the previous assessment (Augean, 2009a).

Table 55 Dimensions and properties of top soil used for farming

Parameter	Units	Value
Area of farmland	m^2	10,000
Depth of soil irrigated	m	1
Top soil porosity	dimensionless	0.3
Top soil saturation	dimensionless	0.5

580. 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 by plants from contaminated soil (Augean, 2009a):

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

where:

- Q_{crop} is the crop consumption rate (kg y^{-1});
- $C_{Rn,water}$ is the activity concentration of radionuclide *Rn* in the irrigation water at time *t* (Bq l^{-1});
- *Irrig* is the irrigation rate (m y^{-1});
- Int_{crop} is the effective interception factor;
- F_{crop} is the fraction remaining after processing;
- $Yield_{crop}$ is the crop yield (kg m^{-2} y^{-1});
- $C_{Rn,soil}$ is the soil 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}).

581. Habit data are discussed above and other parameter values are summarised in Table 56. 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 56 Overview of parameters used for the irrigation scenario

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}	0.15
Infiltration rate [maximum value for grassland from (Augean, 2014)]	All crops	mm y^{-1}	74.3
Crop interception factor	All crops		0.33
Crop processing factor	Green vegetables		0.3
	Root vegetables		1
Yield (crops)	Green vegetables	kg m^{-2} y^{-1}	3.0

Parameter	Substance	Units	Value
	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
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

From (Augean, 2009a)

582. 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.

583. Soil to crop transfer factors are given in Table 172 and dose coefficients for ingestion are given in Table 170 and Table 54.

584. The dose from ingesting animal foodstuffs (e.g. meat and milk) raised on contaminated land is given by (Augean, 2009a), and is adapted here for livestock grazing only pasture (a single crop):

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

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}).

585. Parameter values are summarised in Table 56.

586. The distribution coefficients are defined in Table 169 for soil.

587. Dose from inadvertent ingestion of soil is given by (Augean, 2009a):

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

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}).

588. The soil consumption rate is given in Table 56.

589. 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.

590. Dose coefficients for ingestion are given in Table 170 and Table 54.

591. The dose from external irradiation while living and working on contaminated soil is given by (Augean, 2009a):

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

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.

592. Parameter values are summarised in Table 56.

593. 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.

594. Dose coefficients for irradiation are given in Table 170 and Table 54.

595. The dose from inhalation of contaminated soil is given by (Augean, 2009a):

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

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 ($Sv\ Bq^{-1}$)

596. Parameter values are summarised in Table 56.

597. 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.

598. Dose coefficients for inhalation are given in Table 170 and Table 59.

E.3.4.5. Groundwater doses during the Period of Authorisation

599. Specific dose calculations were undertaken for water extracted at a well located at the site boundary (Table 57) at the end of the period of active management (the time of the peak activity concentration in the groundwater during the period of authorisation). A well at this location does not currently exist but it represents the maximum dose that could occur during the period of authorisation. The second column in the table gives the maximum inventory that can be disposed at the ENRMF (from the DO limits and the radiological capacities for the different scenarios). The third column presents the dose per unit disposal for the drinking water pathway and the fourth column gives the maximum dose per unit disposal due to the irrigation pathway. The fifth column gives the sum of the maximum dose per unit disposal for both pathways and this is used to calculate the dose from the maximum inventory (given in column 6). Doses decrease if the well location is further away from the site, e.g. up to the point of nearest abstraction.

600. GoldSim output has a low value cut-off and reports a lower limit of $1\ 10^{-10}$ 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.

601. The doses are all very low and do not constrain the landfill capacity. The results for Ra-226 are independent of the Ra-226 placement depth in the site

Table 57 Maximum annual doses for adults, based on a unit inventory of 1 MBq for each radionuclide and a well at the site boundary during the period of authorisation

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway (µSv y^{-1} MBq^{-1})	Maximum calculated dose for the irrigation pathway (µSv y^{-1} MBq^{-1})	Sum (µSv y^{-1} MBq^{-1})	Dose from maximum inventory (µSv y^{-1})
H-3	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
C-14	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Maximum calculated dose for the irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Sum ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Cl-36	$1.48\ 10^{6}$	$1.28\ 10^{-8}$	$5.79\ 10^{-8}$	$7.07\ 10^{-8}$	$1.04\ 10^{-1}$
Fe-55	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Co-60	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ni-63	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$1.53\ 10^{-10}$	$1.53\ 10^{-10}$	$1.37\ 10^{-2}$
Nb-94	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Tc-99	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ru-106	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ag-108m	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Sb-125	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Sn-126	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
I-129	$4.17\ 10^{4}$	$1.65\ 10^{-7}$	$6.19\ 10^{-7}$	$7.83\ 10^{-7}$	$3.26\ 10^{-2}$
Ba-133	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Cs-134	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Cs-137	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pm-147	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-152	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-154	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$9.54\ 10^{-10}$	$3.55\ 10^{-9}$	$4.50\ 10^{-9}$	$4.04\ 10^{-1}$
Ra-226	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ra-228	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ac-227	$8.96\ 10^{7}$	$5.62\ 10^{-10}$	$2.09\ 10^{-9}$	$2.65\ 10^{-9}$	$2.38\ 10^{-1}$
Th-229	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Th-230	$6.93\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Th-232	$7.16\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pa-231	$1.86\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
U-232	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$2.11\ 10^{-10}$	$2.11\ 10^{-10}$	$1.89\ 10^{-2}$
U-233	$3.13\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
U-234	$6.41\ 10^{6}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
U-235	$4.92\ 10^{6}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
U-236	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
U-238	$2.53\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Np-237	$4.52\ 10^{5}$	$1.08\ 10^{-9}$	$4.02\ 10^{-9}$	$5.09\ 10^{-9}$	$2.30\ 10^{-3}$
Pu-238	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pu-239	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pu-240	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pu-241	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pu-242	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Am-241	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Maximum calculated dose for the irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Sum ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Cm-243	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$1.11\ 10^{-10}$	$1.11\ 10^{-10}$	$9.98\ 10^{-3}$
Cm-244	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$

602. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.3.

E.3.5. Leachate processing off-site

603. The permit application involves no specific authorised liquid discharge routes as leachate is currently used at the soil treatment facility or treated off-site. An assessment has been made of the radiological impact arising from off-site treatment of contaminated leachate.

604. The GoldSim model provides an estimate of the annual leachate from the facility for the whole site 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 for the ESC considers the treatment of contaminated leachate at an off-site hazardous waste treatment facility, secondary treatment at a sewage treatment works followed by discharge to an estuary. The assessment considers the radiation exposure of workers, anglers fishing in an estuary into which the sewage treatment works discharge and a farming family assumed to grow crops on land fertilised with sludge from the sewage treatment works.

605. The assessment is based on the Environment Agency initial radiological assessment methodology (Environment Agency, 2006b) and applies scaling factors to account for different critical group/potentially exposed group assumptions. 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 doses from sewage treatment. Further dilution occurs at the sewage treatment works receiving discharges from the leachate treatment facility.

606. 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.3.5.1. Estimating activity concentrations in leachate

607. Currently, leachate collected from the landfill is either used on-site at the soil treatment facility or taken for treatment at the Augean Avonmouth hazardous waste water treatment facility. It is assumed that treated leachate is then discharged to sewer and subsequently treated at a sewage works, and sludge from this sewage treatment works is then used to improve agricultural land. The treated sewage effluent is discharged straight to sea (estuary), and hence sewage effluent discharge to river is not considered in the ESC. As explained above, calculations have been undertaken using the methodology set out by the Environment Agency (IAM); (Environment Agency, 2006a) and (Environment Agency, 2006b). The methodology

accounts for radionuclide-specific partitioning of activity between treated sewage effluent and sewage solids, which can be used as a soil conditioner. The Environment Agency methodology for discharge to sea considers exposure of coastal fishermen and their families. The doses from sewage treatment have been calculated for the radionuclides listed in Table 1 that are considered in the IAM.

608. The default IAM calculations are based on generic data and provide a cautious estimate of the radiation dose arising to various exposed groups. The Environment Agency IAM 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 Augean Avonmouth treatment facility has a throughput of about 250 m^3 day^{-1} and the local sewage treatment works has a throughput of about 2.1 10^5 m^3 day^{-1} (Bristol Post, 2009). 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).

609. The flux of radionuclides to the Augean Avonmouth 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 (403.2 m^3 y^{-1}) from the site (based on 28 m^3 monthly loads with an allowance of 20% for peak rainfall). 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 58 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	3.30	1.33 10^3
C-14	7.62 10^{-3}	3.07
Cl-36	1.82	7.35 10^2
Fe-55	2.71 10^{-3}	1.09
Co-60	1.12 10^{-2}	4.50
Ni-63	1.89 10^{-3}	7.63 10^{-1}
Sr-90	5.68 10^{-2}	2.29 10^1
Nb-94	4.76 10^{-3}	1.92
Tc-99	2.95	1.19 10^3
Ru-106	7.40 10^{-3}	2.98
Ag-108m	8.45 10^{-3}	3.41
Sb-125	1.33 10^{-2}	5.34
Sn-126	5.86 10^{-3}	2.36
I-129	6.82 10^{-1}	2.75 10^2
Ba-133	1.69 10^{-1}	6.83 10^1
Cs-134	2.05 10^{-3}	8.25 10^{-1}
Cs-137	2.76 10^{-3}	1.11
Pm-147	2.46 10^{-3}	9.91 10^{-1}
Eu-152	3.02 10^{-3}	1.22

Radionuclide	Leachate activity concentration ($Bq\ m^{-3}/MBq$)	Flux to treatment works ($Bq\ y^{-1}/MBq$)
Eu-154	$2.93\ 10^{-3}$	1.18
Eu-155	$2.75\ 10^{-3}$	1.11
Pb-210	$2.74\ 10^{-3}$	1.10
Ra-226	$1.56\ 10^{-3}$	$6.27\ 10^{-1}$
Ac-227	$1.38\ 10^{-3}$	$5.57\ 10^{-1}$
Th-229	$1.64\ 10^{-3}$	$6.62\ 10^{-1}$
Th-230	$2.54\ 10^{-4}$	$1.02\ 10^{-1}$
Th-232	$2.54\ 10^{-4}$	$1.02\ 10^{-1}$
Pa-231	$2.54\ 10^{-4}$	$1.02\ 10^{-1}$
U-232	$1.41\ 10^{-3}$	$5.69\ 10^{-1}$
U-233	$2.28\ 10^{-2}$	9.19
U-234	$2.30\ 10^{-2}$	9.29
U-235	$2.30\ 10^{-2}$	9.29
U-236	$2.30\ 10^{-2}$	9.29
U-238	$2.30\ 10^{-2}$	9.29
Np-237	$2.30\ 10^{-2}$	9.29
Pu-238	$1.81\ 10^{-1}$	$7.29\ 10^{1}$
Pu-239	$1.40\ 10^{-3}$	$5.65\ 10^{-1}$
Pu-240	$1.41\ 10^{-3}$	$5.69\ 10^{-1}$
Pu-241	$1.41\ 10^{-3}$	$5.69\ 10^{-1}$
Pu-242	$1.35\ 10^{-3}$	$5.43\ 10^{-1}$
Am-241	$1.41\ 10^{-3}$	$5.69\ 10^{-1}$
Cm-243	$3.81\ 10^{-4}$	$1.53\ 10^{-1}$
Cm-243	$1.86\ 10^{-3}$	$7.50\ 10^{-1}$
Cm-244	$1.84\ 10^{-3}$	$7.40\ 10^{-1}$

E.3.5.2. Assessment calculations for off-site leachate treatment

Treatment Facility Worker

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

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

611. The EA methodology provides tables of Dose Per Unit Release (DPUR; μSv y^{-1} per Bq y^{-1} discharge) that can be used to obtain doses from discharges to a sewage treatment plant. The model and parameter assumptions are provided allowing the DPUR to be scaled to account for different circumstances. The worker characteristics used to derive the DPUR are given in Table 59. The assessment model is described below. It uses leachate contamination levels derived from the GoldSim groundwater model (see Section E.3.4) and a realistic throughput for the treatment works.

Table 59 Sewage 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})	1.7 10^{-7}	
Inhalation rate (m^3 h^{-1})	1.2	
Inadvertent sludge ingestion (mg h^{-1})	5	

612. 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 taken from EA methodology and reproduced in Table 60 (adult sewage worker) ; and,
- Dil is a dilution factor that is given by the ratio of the default and actual treatment throughputs, i.e. 60/250.

613. The flux to the leachate treatment works uses a dilution factor that is appropriate to the facility inputs.

Table 60 Dose per unit release factors for sewage treatment workers – sewage release scenario (μSv/y per Bq/y of discharge to sewer) given in the EA IAM methodology

Radionuclide	External irradiation DPUR	Inadvertent ingestion and inhalation DPUR	Total DPUR
H-3	0	3.80 10^{-14}	3.80 10^{-14}
C-14	1.40 10^{-13}	1.30 10^{-12}	1.40 10^{-12}
Cl-36	2.30 10^{-11}	1.60 10^{-12}	2.40 10^{-11}
Fe-55	0	3.60 10^{-12}	3.60 10^{-12}
Co-60	8.90 10^{-7}	3.40 10^{-11}	8.90 10^{-7}
Ni-63	0	9.70 10^{-13}	9.70 10^{-13}
Sr-90	3.70 10^{-10}	4.30 10^{-11}	4.10 10^{-10}

Radionuclide	External irradiation DPUR	Inadvertent ingestion and inhalation DPUR	Total DPUR
Nb-94	nd	nd	nd
Tc-99	$9.90\ 10^{-13}$	$1.10\ 10^{-12}$	$2.10\ 10^{-12}$
Ru-106	$1.10\ 10^{-8}$	$1.10\ 10^{-11}$	$1.10\ 10^{-8}$
Ag-108m	nd	nd	nd
Sb-125	$1.30\ 10^{-7}$	$1.10\ 10^{-11}$	$1.30\ 10^{-7}$
Sn-126	nd	nd	nd
I-129	$1.50\ 10^{-10}$	$2.90\ 10^{-10}$	$4.40\ 10^{-10}$
Ba-133	nd	nd	nd
Cs-134	$2.00\ 10^{-7}$	$7.10\ 10^{-11}$	$2.00\ 10^{-7}$
Cs-137	$7.40\ 10^{-8}$	$4.90\ 10^{-11}$	$7.40\ 10^{-8}$
Pm-147	$1.60\ 10^{-12}$	$2.30\ 10^{-12}$	$3.80\ 10^{-12}$
Eu-152	$2.50\ 10^{-7}$	$1.40\ 10^{-11}$	$2.50\ 10^{-7}$
Eu-154	$2.70\ 10^{-7}$	$2.00\ 10^{-11}$	$2.70\ 10^{-7}$
Eu-155	$6.00\ 10^{-9}$	$2.90\ 10^{-12}$	$6.00\ 10^{-9}$
Pb-210	$4.90\ 10^{-10}$	$7.60\ 10^{-9}$	$8.00\ 10^{-9}$
Ra-226	$3.90\ 10^{-7}$	$2.20\ 10^{-9}$	$4.00\ 10^{-7}$
Ra-228	nd	nd	nd
Ac-227	nd	nd	nd
Th-229	nd	nd	nd
Th-230	$7.00\ 10^{-11}$	$5.80\ 10^{-9}$	$5.80\ 10^{-9}$
Th-232	$3.00\ 10^{-11}$	$8.80\ 10^{-9}$	$8.80\ 10^{-9}$
Pa-231	nd	nd	nd
U-232	nd	nd	nd
U-233	nd	nd	nd
U-234	$3.10\ 10^{-12}$	$2.00\ 10^{-10}$	$2.00\ 10^{-10}$
U-235	$6.30\ 10^{-9}$	$1.80\ 10^{-10}$	$6.50\ 10^{-9}$
U-236	nd	nd	nd
U-238	$1.30\ 10^{-9}$	$1.70\ 10^{-10}$	$1.50\ 10^{-9}$
Np-237	$2.60\ 10^{-9}$	$4.00\ 10^{-9}$	$6.60\ 10^{-9}$
Pu-238	$4.30\ 10^{-12}$	$8.00\ 10^{-9}$	$8.00\ 10^{-9}$
Pu-239	$9.80\ 10^{-12}$	$8.70\ 10^{-9}$	$8.70\ 10^{-9}$
Pu-240	$4.20\ 10^{-12}$	$8.70\ 10^{-9}$	$8.70\ 10^{-9}$
Pu-241	$2.00\ 10^{-13}$	$1.60\ 10^{-10}$	$1.60\ 10^{-10}$
Pu-242	$3.70\ 10^{-12}$	$8.40\ 10^{-9}$	$8.40\ 10^{-9}$
Am-241	$2.40\ 10^{-9}$	$1.30\ 10^{-8}$	$1.50\ 10^{-8}$
Cm-243	$3.50\ 10^{-8}$	$9.40\ 10^{-9}$	$4.40\ 10^{-8}$
Cm-244	$5.80\ 10^{-12}$	$8.10\ 10^{-9}$	$8.10\ 10^{-9}$

614. The radionuclides that are not covered in the Environment Agency methodology are given the letters “nd”.

615. The doses to leachate treatment facility workers are presented in Section E.3.5.3. The doses to workers at the sewage treatment works are assumed to be lower than those at the waste treatment works on account of the increased dilution and are not considered explicitly.

Farming family (soil treated with sewage sludge)

616. Farm land is assumed to be treated repeatedly with contaminated sewage sludge from the sewage treatment works that receives discharges from the hazardous waste treatment plant. 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.

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

618. 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).

619. Parameters characterising the application of treated sewage sludge to agricultural land are summarised in Table 61. 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 53.

Table 61 Parameters characterising the application of treated sewage sludge to agricultural land: applicable during the Period of Authorisation

Parameter	Value	Comment
Rate of application of treated sewage 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	

Parameter	Value	Comment
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}	

620. Habit data for the farming family are summarised in Table 53. The habit data values used in the ESC for the inhalation exposure pathway differ from those in the DPUR since they are equal to the values used in the previous radiological assessment (Augean, 2009a); the Environment Agency DPUR used a lower inhalation rate of 0.92 $m^3\ h^{-1}$ and a lower indoor occupancy factor of 0.5).

621. The leachate treatment assessment model is described above. It uses leachate contamination levels derived from the GoldSim groundwater model (see Section E.3.4).

622. Consumption rates assumed for the farming family using biosolids from the sewage 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. As described above (paragraph 618), the Environment Agency IAM adopted 97.5th percentile consumption rates for all foods and hence they use different values, see Table 62. Scaling factors for consumption (F_P) have been determined by dividing the assumed consumption rates by the EA IAM 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).

623. A biosolids application rate of 8 $kg\ m^{-2}\ y^{-1}$ was used as the default value in the Environment Agency IAM 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 in above (paragraph 618).

Table 62 Food consumption rates ($kg\ y^{-1}$)

Foodstuff	DPUR basis	Mean	97.5th percentile
Green vegetables	80	35	80
Root vegetables	130	60	130
Sheep meat	25	8	25
Sheep liver	10	2.75	10
Cow meat	45	15	45
Cow liver	10	2.75	10
Milk	240	122.5	240

624. 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 sewage 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 64 from (Environment Agency, 2006b);
- *Dil* is a dilution factor that is given by the ratio of the assumed and actual sewage throughputs, i.e. 60/2.1 10^5;
- F_{SAR} is the scaling factor for the sewage application rate;
- F_P is the consumption scaling factor for the specific pathway P; and,
- F_E is the fraction (Table 66) from raw sewage that is disposed in liquid effluent (the rest is disposed with biosolids).

625. The dose per unit flux for inhalation and external irradiation are also scaled to account for the different breathing rate) and fraction of time spent indoors assumed in this assessment (1 m^3 h^{-1} and 0.75, respectively) from the Environment Agency IAM calculations (0.92 m^3 h^{-1} and 0.5, respectively).

626. The doses to an adult of a farming family are presented in Section E.3.5.3.

Anglers (discharge from sewage treatment plant)

627. The assessment of doses to a coastal angler fishing in an estuary that receives discharges from the sewage 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 sewage treatment works is discharged.

628. Habit data assumed for the angler family are summarised in Table 63.

Table 63 Habit data for the angling family: applicable during the Period of Authorisation

Pathway	Adult average	Adult 97.5th	Comment
Fish consumption (kg y^{-1})	61	100	From (Smith & Jones, 2003).
Crustacean consumption (kg y^{-1})	18	20	
Molluscs consumption (kg y^{-1})	14	15	
Occupancy on beach (h y^{-1})	2000		Standard assumption in (Environment Agency, 2006b)

Table 64 Dose per unit release factors for adult farming family – sewage release scenario ($\mu Sv\ y^{-1}$ per $Bq\ y^{-1}$) given in the EA IAM 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	$7.90\ 10^{-8}$	$1.00\ 10^{-7}$	$5.30\ 10^{-8}$	$2.10\ 10^{-8}$	$8.30\ 10^{-8}$	$1.80\ 10^{-8}$	$8.60\ 10^{-8}$	$4.40\ 10^{-10}$	$3.50\ 10^{-12}$	$7.70\ 10^{-14}$
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-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	$1.10\ 10^{-12}$	$6.00\ 10^{-14}$
Sr-90	$1.30\ 10^{-7}$	$3.60\ 10^{-8}$	$1.20\ 10^{-9}$	$4.80\ 10^{-10}$	$3.30\ 10^{-9}$	$7.30\ 10^{-10}$	$8.30\ 10^{-8}$	$3.60\ 10^{-9}$	$8.60\ 10^{-12}$	$1.20\ 10^{-12}$
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	$1.50\ 10^{-10}$	$5.30\ 10^{-11}$	$5.80\ 10^{-11}$	$2.30\ 10^{-11}$	$4.90\ 10^{-11}$	$1.10\ 10^{-11}$	$4.90\ 10^{-13}$	$5.50\ 10^{-8}$	$3.40\ 10^{-12}$	$1.40\ 10^{-13}$
Ag-108m	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-126	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-137	$6.00\ 10^{-9}$	$9.20\ 10^{-9}$	$8.10\ 10^{-8}$	$3.30\ 10^{-8}$	$5.30\ 10^{-8}$	$1.20\ 10^{-8}$	$5.40\ 10^{-8}$	$1.60\ 10^{-6}$	$6.10\ 10^{-12}$	$2.90\ 10^{-12}$
Pm-147	$4.00\ 10^{-11}$	$1.10\ 10^{-11}$	$1.50\ 10^{-11}$	$3.60\ 10^{-11}$	$1.90\ 10^{-11}$	$3.40\ 10^{-11}$	$2.90\ 10^{-12}$	$1.80\ 10^{-11}$	$5.70\ 10^{-12}$	$5.10\ 10^{-14}$
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
Pb-210	$1.10\ 10^{-6}$	$1.40\ 10^{-6}$	$2.90\ 10^{-7}$	$2.20\ 10^{-7}$	$3.50\ 10^{-7}$	$1.60\ 10^{-7}$	$7.00\ 10^{-7}$	$1.10\ 10^{-8}$	$4.20\ 10^{-9}$	$4.50\ 10^{-10}$
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-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	$4.20\ 10^{-8}$	$2.80\ 10^{-9}$	$4.20\ 10^{-9}$	$1.20\ 10^{-7}$	$4.30\ 10^{-9}$	$1.20\ 10^{-7}$	$1.60\ 10^{-9}$	$2.30\ 10^{-10}$	$1.20\ 10^{-7}$	$1.00\ 10^{-10}$
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	$4.00\ 10^{-8}$	$2.60\ 10^{-9}$	$4.00\ 10^{-9}$	$1.10\ 10^{-7}$	$4.10\ 10^{-9}$	$1.10\ 10^{-7}$	$1.60\ 10^{-9}$	$8.80\ 10^{-11}$	$1.20\ 10^{-7}$	$9.90\ 10^{-11}$
Am-241	$6.20\ 10^{-8}$	$6.10\ 10^{-9}$	$6.20\ 10^{-9}$	$1.80\ 10^{-7}$	$7.70\ 10^{-9}$	$2.00\ 10^{-7}$	$3.00\ 10^{-9}$	$5.90\ 10^{-8}$	$1.80\ 10^{-7}$	$1.50\ 10^{-10}$
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	$3.50\ 10^{-8}$	$6.50\ 10^{-10}$	$3.10\ 10^{-9}$	$9.00\ 10^{-8}$	$3.30\ 10^{-9}$	$9.00\ 10^{-8}$	$1.20\ 10^{-9}$	$1.20\ 10^{-10}$	$1.00\ 10^{-7}$	$7.70\ 10^{-11}$

629. 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 sewage treatment works (Bq y^{-1});
- $DF_{Rn,\,fisherman}$ 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 Table 66 (adult fisherman);
- Dil is a dilution factor that is given by the ratio of the assumed and actual treatment throughputs, i.e. 60/2.1 10^5;
- F_E is the fraction from raw sewage 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. 100/3200, to adjust for the assumed exchange rate in the Bristol Channel of 3200 m^3 s^{-1}.

630. The flux to the sewage treatment works assumes all treated leachate is transferred from the treatment facility to the sewage treatment works. Seafood consumption for a coastal group is used based on a recent habit survey (Smith & Jones, 2003) with adjustments for the assumed mean and 97.5th percentile rates as shown in Table 65 for the two most important pathways.

Table 65 Sea food consumption rates (kg y^{-1})

Foodstuff	DPUR basis	Mean	97.5th percentile
Fish	100	61	100
Crustaceans	20	18	20
Molluscs	20	14	15

Table 66 Dose per unit release factors for an adult fisherman – sewage release scenario (µSv/y per Bq/y of discharge to sewer) given in the EA IAM methodology

Radionuclide	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation	Total DPUR
H-3	0.85	2.90 10^{-16}	1.10 10^{-16}	1.10 10^{-16}	0	5.20 10^{-16}
C-14	0.85	1.90 10^{-10}	7.30 10^{-11}	7.30 10^{-11}	1.60 10^{-16}	3.30 10^{-10}
Cl-36	0.90	9.10 10^{-16}	3.60 10^{-16}	3.00 10^{-16}	3.10 10^{-17}	1.60 10^{-15}
Fe-55	0.10	2.10 10^{-14}	1.40 10^{-13}	1.40 10^{-13}	0	3.00 10^{-13}
Co-60	0.20	4.60 10^{-12}	1.80 10^{-11}	5.20 10^{-11}	2.70 10^{-9}	2.80 10^{-9}
Ni-63	0.50	1.70 10^{-12}	6.60 10^{-13}	1.30 10^{-12}	0	3.60 10^{-12}
Sr-90	0.90	1.40 10^{-12}	8.90 10^{-13}	1.80 10^{-12}	1.00 10^{-15}	4.10 10^{-12}

Radionuclide	Fraction to effluent	Fish	Crustacea	Molluscs	External irradiation	Total DPUR
Nb-94	nd	nd	nd	nd	nd	nd
Tc-99	0.90	$8.40\ 10^{-13}$	$4.10\ 10^{-12}$	$2.00\ 10^{-12}$	$1.00\ 10^{-16}$	$7.00\ 10^{-12}$
Ru-106	0.90	$1.10\ 10^{-13}$	$2.20\ 10^{-12}$	$1.10\ 10^{-11}$	$3.50\ 10^{-11}$	$4.80\ 10^{-11}$
Ag-108m	nd	nd	nd	nd	nd	nd
Sb-125	0.20	$1.00\ 10^{-11}$	$2.00\ 10^{-12}$	$2.00\ 10^{-12}$	$1.50\ 10^{-11}$	$2.90\ 10^{-11}$
Sn-126	nd	nd	nd	nd	nd	nd
I-129	0.80	$1.60\ 10^{-11}$	$2.10\ 10^{-12}$	$7.00\ 10^{-12}$	$5.40\ 10^{-15}$	$2.50\ 10^{-11}$
Ba-133	nd	nd	nd	nd	nd	nd
Cs-134	0.70	$2.80\ 10^{-11}$	$5.40\ 10^{-12}$	$6.50\ 10^{-12}$	$8.40\ 10^{-11}$	$1.20\ 10^{-10}$
Cs-137	0.70	$1.90\ 10^{-11}$	$3.80\ 10^{-12}$	$4.50\ 10^{-12}$	$1.20\ 10^{-10}$	$1.50\ 10^{-10}$
Pm-147	0.50	$2.40\ 10^{-14}$	$1.30\ 10^{-13}$	$2.30\ 10^{-13}$	$6.00\ 10^{-15}$	$3.90\ 10^{-13}$
Eu-152	0.50	$1.40\ 10^{-13}$	$7.40\ 10^{-13}$	$1.30\ 10^{-12}$	$2.20\ 10^{-9}$	$2.20\ 10^{-9}$
Eu-154	0.50	$2.00\ 10^{-13}$	$1.00\ 10^{-12}$	$1.80\ 10^{-12}$	$2.00\ 10^{-9}$	$2.00\ 10^{-9}$
Eu-155	0.50	$3.10\ 10^{-14}$	$1.60\ 10^{-13}$	$2.90\ 10^{-13}$	$3.70\ 10^{-11}$	$3.70\ 10^{-11}$
Pb-210	0.10	$6.70\ 10^{-10}$	$1.20\ 10^{-7}$	$6.60\ 10^{-8}$	$2.50\ 10^{-12}$	$1.90\ 10^{-7}$
Ra-226	0.50	$4.40\ 10^{-10}$	$1.70\ 10^{-10}$	$1.70\ 10^{-10}$	$2.60\ 10^{-10}$	$1.00\ 10^{-9}$
Ra-228	nd	nd	nd	nd	nd	nd
Ac-227	nd	nd	nd	nd	nd	nd
Th-229	nd	nd	nd	nd	nd	nd
Th-230	0.10	$3.20\ 10^{-11}$	$2.10\ 10^{-11}$	$2.10\ 10^{-11}$	$3.00\ 10^{-11}$	$1.10\ 10^{-10}$
Th-232	0.10	nd	nd	nd	nd	nd
Pa-231	nd	$7.20\ 10^{-10}$	$4.60\ 10^{-10}$	$4.60\ 10^{-10}$	$5.10\ 10^{-9}$	$6.70\ 10^{-9}$
U-232	nd	nd	nd	nd	nd	nd
U-233	nd	nd	nd	nd	nd	nd
U-234	0.90	$7.80\ 10^{-13}$	$3.10\ 10^{-12}$	$9.20\ 10^{-12}$	$4.90\ 10^{-15}$	$1.30\ 10^{-11}$
U-235	0.90	$7.50\ 10^{-13}$	$2.90\ 10^{-12}$	$8.80\ 10^{-12}$	$9.60\ 10^{-12}$	$2.20\ 10^{-11}$
U-236	nd	nd	nd	nd	nd	nd
U-238	0.90	$1.80\ 10^{-12}$	$6.90\ 10^{-11}$	$2.80\ 10^{-10}$	$1.40\ 10^{-11}$	$3.60\ 10^{-10}$
Np-237	0.50	$1.10\ 10^{-10}$	$9.00\ 10^{-11}$	$1.30\ 10^{-9}$	$5.00\ 10^{-14}$	$1.60\ 10^{-9}$
Pu-238	0.50	$7.20\ 10^{-13}$	$2.80\ 10^{-12}$	$8.50\ 10^{-12}$	$1.80\ 10^{-12}$	$1.40\ 10^{-11}$
Pu-239	0.50	$1.20\ 10^{-10}$	$9.90\ 10^{-11}$	$1.50\ 10^{-9}$	$1.20\ 10^{-13}$	$1.70\ 10^{-9}$
Pu-240	0.50	$1.20\ 10^{-10}$	$9.90\ 10^{-11}$	$1.50\ 10^{-9}$	$5.30\ 10^{-14}$	$1.70\ 10^{-9}$
Pu-241	0.50	$2.30\ 10^{-12}$	$1.80\ 10^{-12}$	$2.70\ 10^{-11}$	$2.40\ 10^{-13}$	$3.20\ 10^{-11}$
Pu-242	0.50	$7.00\ 10^{-12}$	$1.10\ 10^{-11}$	$2.80\ 10^{-11}$	$2.50\ 10^{-11}$	$7.10\ 10^{-11}$
Am-241	0.10	$1.20\ 10^{-10}$	$9.50\ 10^{-11}$	$1.40\ 10^{-9}$	$4.70\ 10^{-14}$	$1.60\ 10^{-9}$
Cm-243	0.10	$5.10\ 10^{-12}$	$8.20\ 10^{-12}$	$2.00\ 10^{-11}$	$2.60\ 10^{-10}$	$3.00\ 10^{-10}$
Cm-244	0.10	$4.00\ 10^{-12}$	$6.40\ 10^{-12}$	$1.60\ 10^{-11}$	$4.00\ 10^{-14}$	$2.70\ 10^{-11}$

E.3.5.3. Doses from leachate treatment

Dose per MBq Deposited at the ENRMF – Leachate Treatment

631. The calculated doses shown below for each of the assessed groups are per MBq input to the ENRMF.

Table 67 Dose per unit disposal at the ENRMF (µSv y^{-1} / MBq) – leachate treatment

Radionuclide	Leachate treatment worker (µSv y^{-1} / MBq)	Farming family (µSv y^{-1} / MBq)	Fisherman (µSv y^{-1} / MBq)
H-3	1.21 10^{-11}	6.12 10^{-13}	1.16 10^{-18}
C-14	1.03 10^{-12}	4.09 10^{-12}	1.76 10^{-15}
Cl-36	4.23 10^{-9}	3.63 10^{-9}	2.14 10^{-18}
Fe-55	9.44 10^{-13}	1.99 10^{-12}	6.04 10^{-20}
Co-60	9.60 10^{-7}	5.36 10^{-9}	5.32 10^{-15}
Ni-63	1.78 10^{-13}	3.47 10^{-14}	2.67 10^{-18}
Sr-90	2.25 10^{-9}	9.56 10^{-11}	1.57 10^{-16}
Nb-94	nd	nd	nd
Tc-99	6.00 10^{-10}	1.92 10^{-8}	1.40 10^{-14}
Ru-106	7.88 10^{-9}	1.75 10^{-12}	2.62 10^{-16}
Ag-108m	nd	nd	nd
Sb-125	1.67 10^{-7}	7.56 10^{-10}	6.50 10^{-17}
Sn-126	nd	nd	nd
I-129	2.90 10^{-8}	6.32 10^{-9}	1.09 10^{-14}
Ba-133	nd	nd	nd
Cs-134	3.96 10^{-8}	5.48 10^{-11}	1.51 10^{-16}
Cs-137	1.98 10^{-8}	6.67 10^{-11}	2.44 10^{-16}
Pm-147	9.04 10^{-13}	1.47 10^{-14}	3.43 10^{-19}
Eu-152	7.30 10^{-8}	3.08 10^{-10}	2.87 10^{-15}
Eu-154	7.66 10^{-8}	3.06 10^{-10}	2.54 10^{-15}
Eu-155	1.60 10^{-9}	5.30 10^{-12}	4.45 10^{-17}
Pb-210	2.12 10^{-9}	5.99 10^{-10}	4.02 10^{-14}
Ra-226	6.20 10^{-8}	8.59 10^{-10}	3.50 10^{-14}
Ra-228	nd	nd	nd
Ac-227	nd	nd	nd
Th-229	nd	nd	nd
Th-230	1.72 10^{-9}	2.22 10^{-11}	6.01 10^{-16}
Th-232	2.16 10^{-10}	2.49 10^{-10}	1.45 10^{-16}
Pa-231	nd	nd	nd
U-232	nd	nd	nd
U-233	nd	nd	nd
U-234	4.46 10^{-10}	2.14 10^{-12}	1.88 10^{-16}
U-235	1.45 10^{-8}	1.38 10^{-11}	3.50 10^{-16}

Radionuclide	Leachate treatment worker ($\mu Sv\ y^{-1}$ / MBq)	Farming family ($\mu Sv\ y^{-1}$ / MBq)	Fisherman ($\mu Sv\ y^{-1}$ / MBq)
U-236	nd	nd	nd
U-238	$3.34\ 10^{-9}$	$4.35\ 10^{-12}$	$2.04\ 10^{-16}$
Np-237	$1.15\ 10^{-7}$	$6.26\ 10^{-9}$	$2.30\ 10^{-14}$
Pu-238	$1.08\ 10^{-9}$	$2.02\ 10^{-11}$	$7.06\ 10^{-16}$
Pu-239	$1.19\ 10^{-9}$	$2.34\ 10^{-11}$	$8.14\ 10^{-16}$
Pu-240	$1.19\ 10^{-9}$	$2.29\ 10^{-11}$	$8.14\ 10^{-16}$
Pu-241	$3.71\ 10^{-11}$	$7.31\ 10^{-13}$	$1.43\ 10^{-17}$
Pu-242	$1.15\ 10^{-9}$	$2.22\ 10^{-11}$	$7.66\ 10^{-16}$
Am-241	$5.55\ 10^{-10}$	$1.85\ 10^{-11}$	$2.44\ 10^{-18}$
Cm-243	$7.92\ 10^{-9}$	$1.16\ 10^{-10}$	$4.68\ 10^{-17}$
Cm-244	$1.44\ 10^{-9}$	$4.28\ 10^{-11}$	$5.26\ 10^{-18}$

Dose from maximum inventory – Leachate Treatment

632. The radionuclide specific dose arising from disposing of the maximum inventory (minimum of 89.6 TBq and the radiological capacity) is shown in Table 68. The results for Ra-226 are independent of the Ra-226 placement depth in the site

633. The highest dose is for Co-60 at the treatment facility (86 $\mu Sv\ y^{-1}$), but this would only occur if it was the only radionuclide to be disposed at the ENRMF. As a percentage of the national inventory Co-60 accounts for about 7% of low level waste (Nuclear Decommissioning Authority, 2013), and it comprised less than 2% of the activity in radioactive waste disposed to June 2015 at the ENRMF.

634. On this basis the dose due to off-site treatment of leachate containing Co-60 would never exceed 0.3 mSv y^{-1} and would be substantially lower under a sum of fractions approach.

635. The estimate also assumes that all leachate is sent for treatment whereas most is used on site. The model does not take into account sorption within waste materials whereas in reality waste received at the ENRMF is likely to provide additional sorption sites within waste cells.

636. These doses have not been used to determine the radiological capacity of the ENRMF. The quantity of leachate collected from the sumps is variable, ranging from 7,468 to 15,812 t between 2011 and 2013 from the active site, and the quantity of leachate will be lower once all cells are capped.

Table 68 Dose for exposure from the off-site treatment of leachate: disposal of maximum inventory

Radionuclide	Maximum inventory (MBq)	Treatment facility worker ($\mu Sv\ y^{-1}$)	Farming family - adult ($\mu Sv\ y^{-1}$)	Fisherman - adult ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$1.09\ 10^{-3}$	$5.48\ 10^{-5}$	$1.04\ 10^{-10}$
C-14	$8.96\ 10^{7}$	$9.24\ 10^{-5}$	$3.67\ 10^{-4}$	$1.57\ 10^{-7}$
Cl-36	$1.48\ 10^{6}$	$6.25\ 10^{-3}$	$5.36\ 10^{-3}$	$3.16\ 10^{-12}$
Fe-55	$8.96\ 10^{7}$	$8.46\ 10^{-5}$	$1.78\ 10^{-4}$	$5.41\ 10^{-12}$
Co-60	$8.96\ 10^{7}$	$8.60\ 10^{1}$	$4.80\ 10^{-1}$	$4.76\ 10^{-7}$
Ni-63	$8.96\ 10^{7}$	$1.59\ 10^{-5}$	$3.11\ 10^{-6}$	$2.40\ 10^{-10}$
Sr-90	$8.96\ 10^{7}$	$2.02\ 10^{-1}$	$8.57\ 10^{-3}$	$1.40\ 10^{-8}$
Nb-94	$8.96\ 10^{7}$	nd	nd	nd
Tc-99	$8.96\ 10^{7}$	$5.37\ 10^{-2}$	1.72	$1.26\ 10^{-6}$
Ru-106	$8.96\ 10^{7}$	$7.06\ 10^{-1}$	$1.57\ 10^{-4}$	$2.35\ 10^{-8}$
Ag-108m	$8.96\ 10^{7}$	nd	nd	nd
Sb-125	$8.96\ 10^{7}$	$1.49\ 10^{1}$	$6.77\ 10^{-2}$	$5.83\ 10^{-9}$
Sn-126	$8.96\ 10^{7}$	nd	nd	nd
I-129	$4.17\ 10^{4}$	$1.21\ 10^{-3}$	$2.63\ 10^{-4}$	$4.55\ 10^{-10}$
Ba-133	$8.96\ 10^{7}$	nd	nd	nd
Cs-134	$8.96\ 10^{7}$	3.55	$4.91\ 10^{-3}$	$1.35\ 10^{-8}$
Cs-137	$8.96\ 10^{7}$	1.77	$5.97\ 10^{-3}$	$2.18\ 10^{-8}$
Pm-147	$8.96\ 10^{7}$	$8.10\ 10^{-5}$	$1.32\ 10^{-6}$	$3.08\ 10^{-11}$
Eu-152	$8.96\ 10^{7}$	6.54	$2.76\ 10^{-2}$	$2.57\ 10^{-7}$
Eu-154	$8.96\ 10^{7}$	6.86	$2.74\ 10^{-2}$	$2.27\ 10^{-7}$
Eu-155	$8.96\ 10^{7}$	$1.43\ 10^{-1}$	$4.75\ 10^{-4}$	$3.98\ 10^{-9}$
Pb-210	$8.96\ 10^{7}$	$1.90\ 10^{-1}$	$5.36\ 10^{-2}$	$3.60\ 10^{-6}$
Ra-226	$8.96\ 10^{7}$	5.56	$7.70\ 10^{-2}$	$3.14\ 10^{-6}$
Ra-228	$8.96\ 10^{7}$	nd	nd	nd
Ac-227	$8.96\ 10^{7}$	nd	nd	nd
Th-229	$8.96\ 10^{7}$	nd	nd	nd
Th-230	$6.93\ 10^{7}$	$1.19\ 10^{-1}$	$1.54\ 10^{-3}$	$4.17\ 10^{-8}$
Th-232	$7.16\ 10^{7}$	$1.55\ 10^{-2}$	$1.78\ 10^{-2}$	$1.04\ 10^{-8}$
Pa-231	$1.86\ 10^{7}$	nd	nd	nd
U-232	$8.96\ 10^{7}$	nd	nd	nd
U-233	$3.13\ 10^{7}$	nd	nd	nd
U-234	$6.41\ 10^{6}$	$2.86\ 10^{-3}$	$1.37\ 10^{-5}$	$1.20\ 10^{-9}$
U-235	$4.92\ 10^{6}$	$7.12\ 10^{-2}$	$6.76\ 10^{-5}$	$1.72\ 10^{-9}$
U-236	$8.96\ 10^{7}$	nd	nd	nd
U-238	$2.53\ 10^{7}$	$8.47\ 10^{-2}$	$1.10\ 10^{-4}$	$5.18\ 10^{-9}$
Np-237	$4.52\ 10^{5}$	$5.22\ 10^{-2}$	$2.83\ 10^{-3}$	$1.04\ 10^{-8}$
Pu-238	$8.96\ 10^{7}$	$9.72\ 10^{-2}$	$1.81\ 10^{-3}$	$6.32\ 10^{-8}$
Pu-239	$8.96\ 10^{7}$	$1.07\ 10^{-1}$	$2.10\ 10^{-3}$	$7.29\ 10^{-8}$

Radionuclide	Maximum inventory (MBq)	Treatment facility worker ($\mu Sv\ y^{-1}$)	Farming family - adult ($\mu Sv\ y^{-1}$)	Fisherman - adult ($\mu Sv\ y^{-1}$)
Pu-240	$8.96\ 10^{7}$	$1.07\ 10^{-1}$	$2.05\ 10^{-3}$	$7.29\ 10^{-8}$
Pu-241	$8.96\ 10^{7}$	$3.32\ 10^{-3}$	$6.55\ 10^{-5}$	$1.28\ 10^{-9}$
Pu-242	$8.96\ 10^{7}$	$1.03\ 10^{-1}$	$1.99\ 10^{-3}$	$6.86\ 10^{-8}$
Am-241	$8.96\ 10^{7}$	$4.97\ 10^{-2}$	$1.66\ 10^{-3}$	$2.19\ 10^{-10}$
Cm-243	$8.96\ 10^{7}$	$7.10\ 10^{-1}$	$1.04\ 10^{-2}$	$4.19\ 10^{-9}$
Cm-244	$8.96\ 10^{7}$	$1.29\ 10^{-1}$	$3.84\ 10^{-3}$	$4.72\ 10^{-10}$

637. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.3.

E.3.6. Aircraft crash

638. This scenario is included due to the proximity of RAF Wittering which was an operational Harrier aircraft base until the fleet was withdrawn from service in December 2010. RAF Wittering remains an active air base supporting a wide range of military flying activities. The ENRMF is located approximately 2.5 km south west of the runway which runs approximately east-west.

639. The frequency of military aircraft crashes in the UK is very low but 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 included for this reason.

640. This scenario is not used to constrain landfill capacity because it is independent of the tonnage disposed at the ENRMF and crashes have a low probability of occurrence.

641. The dose criteria are the legal 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}.

642. The event is assessed for the pre-closure phase but could also apply to the post-closure phase for the public if the landfill closure cap (at least 1.6 m thick) did not provide full protection from the impact. The event could then be considered an intrusion in which case the 3 to 20 mSv y^{-1} dose criteria would apply.

Potentially exposed group

643. The assessment of doses resulting from waste that is released to atmosphere following an aircraft impact is based on that used in the previous assessment (Augean, 2009a). Members of the exposed group are assumed to be adults and to be exposed as a result of inhalation of contaminated dust.

E.3.6.1. Estimating activity releases due to aircraft impact

644. It is assumed that immediate evacuation of the near zone would occur from such an extreme event and that the event has a 30 minute release duration. Within the very near zone immediate fatality due to impact would be likely. Hence, the distance from the impact to the nearest exposed member of the public is assumed to be 200 m.

The atmospheric conditions assumed are worst case still conditions (dispersal and mixing is not assumed to be enhanced by fire).

645. The worker exposure is assumed to be the same as the public exposure because workers would evacuate quickly to the same distance. The worker inhalation rate is used in the assessment (Table 36).

646. The scenario is not contained within the SNIFFER model and has been separately addressed below using the approach described in the previous ESC (Augean, 2009a). The scenario has a very low probability of occurrence (less than 2 10^{-6}).

647. The following gives exposure to both workers and the public under the following assumptions using the UKAEA release methodology from the safety assessment handbook (reference 22 of Augean's 2009 permit application). The approach used is to assume an amount of material is physically displaced by crater formation through impact of a high velocity military aircraft. This is considered a reasonable scenario given the presence of an RAF base close to the landfill when compared to much less likely scenarios involving heavy civilian aircraft.

648. Due to the complexity of such an event this assessment can only be considered as a scoping calculation based on conservative assumptions. The assumptions are as follows.

- The aircraft hits an area of exposed waste and forms a crater.
- The crater size can be estimated from theoretical models for estimated impact parameters such as densities, impact velocity, impact angle, missile dimensions and target density/type (reference 21 of Augean 2009 application). Scoping calculations indicate that crater sizes of 300 m^3 are conceivable. Actual crater sizes from impacts due to Harrier jets (the type of aircraft formerly based at RAF Wittering) reveal a wide variation from virtually no displacement to significant craters dependent on the nature of the event. A record (reference 23 of Augean 2009 application) notes a Harrier jet impact forming a crater of approximately 300 m^3. For comparison, the Lockerbie B747 impact formed a crater of 560 m^3 (reference 24 of Augean 2009 application).
- The displaced waste contains the maximum activity concentration of a single nuclide at 200 Bq g^{-1}.
- The density of the displaced waste is 1.53 t m^{-3}. 300 m^3 or 460 t are displaced, leading to displacement of an inventory of about 9.18 10^4 MBq.
- The distance to the nearest public is 200 m and the event has 30 minute release duration. This is on the basis that immediate evacuation of the near zone would occur from such an extreme event and within the very near zone immediate fatality due to impact would be likely.
- The effect of fire on dispersal is not included.
- The worker exposure is the same as the public exposure because workers would evacuate quickly to the same distance.
- The atmospheric conditions are worst case still conditions and mixing is not assumed to be enhanced by fire.

E.3.6.2. Assessment calculation involving an aircraft crash

649. 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* released (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.

650. The release fraction is given by the ratio:

$$RF_1 = \frac{Mass\ of\ soil\ released\ to\ air\ by\ aircraft\ impact}{Mass\ of\ soil\ displaced\ by\ aircraft\ impact}$$

651. The decontamination factor accounts for the effects of protective equipment and measures. The parameters used in these calculations are given in Table 69.

Table 69 Aircraft impact parameters

Parameter	Units	Value	Description
I	Bq	9 10^{10}	Radionuclide inventory
RF_1		1 10^{-3}	Release fraction
RF_2		0.1	Respirable fraction
C	s m^{-3}	1.5 10^{-3}	Dispersion coefficient
B	m^3 s^{-1}	3.3 10^{-4}	Inhalation rate
DF		1	Decontamination factor

E.3.6.3. Doses from an aircraft crash

652. The dose coefficients and resulting effective doses from inhalation of contaminated displaced material are given in Table 71. The largest dose (approximately 3 mSv) arises from inhalation of Ac-227, followed by Th-229 at about 1 mSv, the remaining alpha emitters about 0.5 mSv or less with much lower doses for the beta and gamma emitters. There is very little Ac-227 reported in the national inventory of LLW (Nuclear Decommissioning Authority, 2013), a total of 13 MBq. The inventory assumed to be displaced by the aircraft crash in the calculations is 9.18 10^4 MBq, i.e. a factor of 7000 greater. The results for Ra-226 would be zero for a 5 m or greater emplacement depth.

653. This calculation uses conservative assumptions and parameter values and will give rise to conservative estimates of doses; further, the complexity of an aircraft impact means that this calculation can only be considered as a scoping calculation. Nevertheless, the scoping calculations indicate that the 3 mSv y^{-1} human intrusion dose guidance level would not be exceeded by this low probability event.

654. This scenario has not been used to constrain the radiological capacity because it has very low probability of occurrence and is independent of the total tonnage and total activity in the waste cells at the ENRMF.

E.3.7. Dropped container

655. The impacted groups during the pre-closure phase are workers and the public.

656. This scenario is considered below and it was also addressed using a radiological risk assessment for occupational exposure completed by the HPA (Annex C, (Augean, 2009a)). 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.

657. This scenario is not used to constrain landfill capacity because it is independent of the tonnage disposed at the ENRMF.

658. The dose criteria are the legal 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

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

660. The exposed groups are the public and workers. Exposure to both workers and the public has been calculated using the UKAEA dropped load methodology from the safety assessment handbook (reference 22 of 2009 Augean application) and the following assumptions.

661. 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 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 36).

E.3.7.1. Estimating activity concentrations following a dropped load

662. 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 2009 Augean application).

663. 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.3.7.2. Assessment calculation involving a dropped load

664. 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.

665. The parameters used in this calculation are given in Table 70. The inhalation dose coefficients are given in Table 170.

Table 70 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	
DF		1	Decontamination factor	

E.3.7.3. Dose from a dropped load

666. The effective doses arising from a dropped container are given in Table 71. The results for Ra-226 are independent of the Ra-226 placement depth in the site

Table 71 Doses from a dropped container and an aircraft crash

Radionuclide	Dropped load dose		Aircraft impact dose (mSv)
	Worker (mSv)	Public (mSv)	
H-3	$8.58\ 10^{-7}$	$2.92\ 10^{-9}$	$1.16\ 10^{-6}$
C-14	$1.91\ 10^{-5}$	$6.51\ 10^{-8}$	$2.58\ 10^{-5}$
Cl-36	$2.41\ 10^{-5}$	$8.19\ 10^{-8}$	$3.25\ 10^{-5}$
Fe-55	$2.54\ 10^{-6}$	$8.64\ 10^{-9}$	$3.43\ 10^{-6}$
Co-60	$1.02\ 10^{-4}$	$3.48\ 10^{-7}$	$1.38\ 10^{-4}$
Ni-63	$4.29\ 10^{-6}$	$1.46\ 10^{-8}$	$5.79\ 10^{-6}$
Sr-90	$5.35\ 10^{-4}$	$1.82\ 10^{-6}$	$7.22\ 10^{-4}$
Nb-94	$1.62\ 10^{-4}$	$5.50\ 10^{-7}$	$2.18\ 10^{-4}$
Tc-99	$4.29\ 10^{-5}$	$1.46\ 10^{-7}$	$5.79\ 10^{-5}$
Ru-106	$2.18\ 10^{-4}$	$7.41\ 10^{-7}$	$2.94\ 10^{-4}$
Ag-108m	$1.22\ 10^{-4}$	$4.15\ 10^{-7}$	$1.65\ 10^{-4}$
Sb-125	$4.28\ 10^{-5}$	$1.45\ 10^{-7}$	$5.77\ 10^{-5}$
Sn-126	$9.39\ 10^{-5}$	$3.19\ 10^{-7}$	$1.27\ 10^{-4}$
I-129	$1.19\ 10^{-4}$	$4.04\ 10^{-7}$	$1.60\ 10^{-4}$
Ba-133	$3.30\ 10^{-5}$	$1.12\ 10^{-7}$	$4.46\ 10^{-5}$
Cs-134	$6.60\ 10^{-5}$	$2.24\ 10^{-7}$	$8.91\ 10^{-5}$
Cs-137	$1.29\ 10^{-4}$	$4.38\ 10^{-7}$	$1.74\ 10^{-4}$
Pm-147	$1.65\ 10^{-5}$	$5.61\ 10^{-8}$	$2.23\ 10^{-5}$
Eu-152	$1.39\ 10^{-4}$	$4.71\ 10^{-7}$	$1.87\ 10^{-4}$
Eu-154	$1.75\ 10^{-4}$	$5.95\ 10^{-7}$	$2.36\ 10^{-4}$
Eu-155	$2.28\ 10^{-5}$	$7.74\ 10^{-8}$	$3.07\ 10^{-5}$
Pb-210	$3.30\ 10^{-2}$	$1.12\ 10^{-4}$	$4.45\ 10^{-2}$
Ra-226	$6.44\ 10^{-2}$	$2.19\ 10^{-4}$	$8.69\ 10^{-2}$
Ra-228	$1.97\ 10^{-1}$	$6.69\ 10^{-4}$	$2.66\ 10^{-1}$
Ac-227	1.88	$6.38\ 10^{-3}$	2.53
Th-229	$8.45\ 10^{-1}$	$2.87\ 10^{-3}$	1.14
Th-230	$3.30\ 10^{-1}$	$1.12\ 10^{-3}$	$4.46\ 10^{-1}$
Th-232	$5.61\ 10^{-1}$	$1.91\ 10^{-3}$	$7.57\ 10^{-1}$
Pa-231	$4.62\ 10^{-1}$	$1.57\ 10^{-3}$	$6.24\ 10^{-1}$
U-232	$1.22\ 10^{-1}$	$4.15\ 10^{-4}$	$1.65\ 10^{-1}$
U-233	$3.17\ 10^{-2}$	$1.08\ 10^{-4}$	$4.28\ 10^{-2}$
U-234	$3.10\ 10^{-2}$	$1.05\ 10^{-4}$	$4.19\ 10^{-2}$
U-235	$2.81\ 10^{-2}$	$9.54\ 10^{-5}$	$3.79\ 10^{-2}$

Radionuclide	Dropped load dose		Aircraft impact dose (mSv)
	Worker (mSv)	Public (mSv)	
U-236	$2.87\ 10^{-2}$	$9.76\ 10^{-5}$	$3.88\ 10^{-2}$
U-238	$2.64\ 10^{-2}$	$8.99\ 10^{-5}$	$3.57\ 10^{-2}$
Np-237	$1.65\ 10^{-1}$	$5.61\ 10^{-4}$	$2.23\ 10^{-1}$
Pu-238	$3.63\ 10^{-1}$	$1.23\ 10^{-3}$	$4.90\ 10^{-1}$
Pu-239	$3.96\ 10^{-1}$	$1.35\ 10^{-3}$	$5.35\ 10^{-1}$
Pu-240	$3.96\ 10^{-1}$	$1.35\ 10^{-3}$	$5.35\ 10^{-1}$
Pu-241	$7.59\ 10^{-3}$	$2.58\ 10^{-5}$	$1.02\ 10^{-2}$
Pu-242	$3.63\ 10^{-1}$	$1.23\ 10^{-3}$	$4.90\ 10^{-1}$
Am-241	$3.17\ 10^{-1}$	$1.08\ 10^{-3}$	$4.28\ 10^{-1}$
Cm-243	$2.28\ 10^{-1}$	$7.74\ 10^{-4}$	$3.07\ 10^{-1}$
Cm-244	$1.88\ 10^{-1}$	$6.40\ 10^{-4}$	$2.54\ 10^{-1}$

667. The doses meet the site criterion for workers for all radionuclides except Ac-227 (the estimated dose for Ac-227 is less than the criterion of 6 mSv for classifying workers as radiation workers), and all doses to the public are below 0.01 mSv. Ac-227 is very unlikely to be present at 200 Bq g^{-1} given the low occurrence of this radionuclide (Nuclear Decommissioning Authority, 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.

668. 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 and these regulations specify isotope specific limits designed to ensure public safety.

669. 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 the ENRMF.

E.3.8. Leachate spillage

670. 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.

671. 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 farm land adjacent to a water body that becomes contaminated by the spillage also becomes contaminated. Members of the exposed group are assumed to be adults. 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.

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

Potentially exposed group

673. 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 farm land 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.

674. Table 53 details the habit data assumed for the farming family, assumptions concerning drinking water and fish consumption are in Table 72.

Table 72 Habit data for the leachate spillage: applicable during the Period of Authorisation

Pathway*	Adult average	Adult 97.5th	Comment
Fish consumption (kg y^{-1})	15	40	From (Smith & Jones, 2003).
Drinking water consumption (m^3 y^{-1})	0.6		

*Other data are the same as presented in Table 53.

E.3.8.1. Estimating activity concentrations after a leachate spillage

675. 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 (Table 58). This is a very conservative set of assumptions.

676. 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).

677. 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.3.8.2. Assessment calculations for a farming family after a leachate spillage

Irrigation and Drinking Water

678. The exposure pathways for irrigation are the same as those detailed for groundwater contamination, see Section E.3.4.4; paragraphs 580 to 598. There is however no allowance for daughter radionuclide ingrowth.

679. 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

680. 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}).

681. The transfer factors for freshwater fish are listed in Table 173. 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).

682. 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.3.8.3. Doses from Leachate spillage

683. 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 73. The public dose constraint is 0.3 mSv.

684. The highest doses occur if the leachate comprises only I-129, at 4 10^{-6} µSv per MBq. The spillage event has a low probability of occurring and clean-up actions would be taken to largely mitigate the event altogether. The scenario does not constrain the radiological capacity even without mitigation measures. The results for Ra-226 are independent of the Ra-226 placement depth in the site

Table 73 Dose to farming family from leachate spillage

Radionuclide	Maximum inventory (MBq)	Dose to Farming family - adult (µSv MBq^{-1})	Dose from maximum inventory (µSv y^{-1})
H-3	8.96 10^{7}	2.62 10^{-9}	2.34 10^{-1}
C-14	8.96 10^{7}	2.24 10^{-8}	2.01
Cl-36	1.48 10^{6}	1.18 10^{-7}	1.75 10^{-1}
Fe-55	8.96 10^{7}	8.31 10^{-11}	7.45 10^{-3}
Co-60	8.96 10^{7}	7.79 10^{-9}	6.98 10^{-1}
Ni-63	8.96 10^{7}	2.68 10^{-11}	2.40 10^{-3}
Sr-90	8.96 10^{7}	1.24 10^{-7}	1.11 10^{1}
Nb-94	8.96 10^{7}	1.66 10^{-9}	1.49 10^{-1}
Tc-99	8.96 10^{7}	9.53 10^{-8}	8.54
Ru-106	8.96 10^{7}	2.24 10^{-9}	2.00 10^{-1}
Ag-108m	8.96 10^{7}	8.03 10^{-10}	7.20 10^{-2}
Sb-125	8.96 10^{7}	1.60 10^{-9}	1.43 10^{-1}
Sn-126	8.96 10^{7}	1.76 10^{-8}	1.58
I-129	4.17 10^{4}	4.14 10^{-6}	1.73 10^{-1}
Ba-133	8.96 10^{7}	1.04 10^{-8}	9.28 10^{-1}
Cs-134	8.96 10^{7}	4.50 10^{-8}	4.03
Cs-137	8.96 10^{7}	4.15 10^{-8}	3.72
Pm-147	8.96 10^{7}	3.42 10^{-11}	3.06 10^{-3}
Eu-152	8.96 10^{7}	2.29 10^{-10}	2.05 10^{-2}

Radionuclide	Maximum inventory (MBq)	Dose to Farming family - adult ($\mu Sv\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Eu-154	$8.96\ 10^{7}$	$3.16\ 10^{-10}$	$2.83\ 10^{-2}$
Eu-155	$8.96\ 10^{7}$	$4.71\ 10^{-11}$	$4.22\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$1.06\ 10^{-6}$	$9.48\ 10^{1}$
Ra-226	$8.96\ 10^{7}$	$2.18\ 10^{-7}$	$1.96\ 10^{1}$
Ra-228	$8.96\ 10^{7}$	$7.46\ 10^{-8}$	6.68
Ac-227	$8.96\ 10^{7}$	$3.39\ 10^{-7}$	$3.04\ 10^{1}$
Th-229	$8.96\ 10^{7}$	$8.32\ 10^{-9}$	$7.46\ 10^{-1}$
Th-230	$6.93\ 10^{7}$	$2.85\ 10^{-9}$	$1.98\ 10^{-1}$
Th-232	$7.16\ 10^{7}$	$1.44\ 10^{-8}$	1.03
Pa-231	$1.86\ 10^{7}$	$4.23\ 10^{-8}$	$7.86\ 10^{-1}$
U-232	$8.96\ 10^{7}$	$3.18\ 10^{-7}$	$2.85\ 10^{1}$
U-233	$3.13\ 10^{7}$	$4.96\ 10^{-8}$	1.55
U-234	$6.41\ 10^{6}$	$4.76\ 10^{-8}$	$3.05\ 10^{-1}$
U-235	$4.92\ 10^{6}$	$4.60\ 10^{-8}$	$2.26\ 10^{-1}$
U-236	$8.96\ 10^{7}$	$4.57\ 10^{-8}$	4.09
U-238	$2.53\ 10^{7}$	$4.71\ 10^{-8}$	1.19
Np-237	$4.52\ 10^{5}$	$8.47\ 10^{-7}$	$3.82\ 10^{-1}$
Pu-238	$8.96\ 10^{7}$	$1.31\ 10^{-8}$	1.17
Pu-239	$8.96\ 10^{7}$	$1.43\ 10^{-8}$	1.28
Pu-240	$8.96\ 10^{7}$	$1.43\ 10^{-8}$	1.28
Pu-241	$8.96\ 10^{7}$	$2.62\ 10^{-10}$	$2.35\ 10^{-2}$
Pu-242	$8.96\ 10^{7}$	$1.37\ 10^{-8}$	1.23
Am-241	$8.96\ 10^{7}$	$4.06\ 10^{-9}$	$3.64\ 10^{-1}$
Cm-243	$8.96\ 10^{7}$	$1.49\ 10^{-8}$	1.33
Cm-244	$8.96\ 10^{7}$	$1.18\ 10^{-8}$	1.05

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

685. As described in Section E.2, the ESC considers the exposure of adults since they are expected to be limiting in the majority of cases and any increases in doses for other age groups will be small compared with the uncertainty in the assessed doses. Hence the ESC calculates the radiological capacity of the ENRMF based on the radiological impact on adults.

686. 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.

687. Under the planning permission requirements the ENRMF site must be restored to wildflower grassland with woodland. The restoration is carried out progressively during the life of the site and will be completed by 31/12/2026 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.

688. At some point in time after site restoration is complete members of the public or farmers will have access to this land for its intended recreational or agricultural 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, also intended to represent agricultural site users.

689. 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. The exposed group for the groundwater migration scenario is members of the public drinking groundwater abstracted from a well and using it for irrigation of land.

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

Table 74 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.

Event/scenario	Exposure pathway	Description
Release to groundwater: abstraction at nearest well and 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.
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.

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

Long Term Climate Change

692. The effects of very long term climate change are not assessed because the site is already permitted as a hazardous site and LLW disposal gives rise to no additional considerations in respect of flooding, coastal erosion or sea level rises. The surface water management system at the site is designed taking into account changes anticipated in rainfall events as a result of climate change (Augean, 2014). 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. 1000s of years in the future).

693. The HRA (Augean, 2014) 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 considered 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. For parts of the landfill where the capping system incorporates a GCL in addition to a LLDPE geomembrane it is likely that the long term infiltration through the capping system will be less than the effective rainfall because GCL will be less susceptible to chemical or physical degradation than the LLDPE geomembrane.

694. The HRA does not consider a potential increased risk of flooding resulting from climate change. A flood risk assessment (FRA) for the proposed western landfill area was included in the application for a Development Consent Order for the site. The FRA referenced the approved surface water management plan (SWMP) for the site. In the SWMP the design for the surface water management system at the site is set out and on this basis it is assumed that the potential for surface water flooding and uncontrolled runoff at the site following restoration will be low. It is stated in the FRA that the SWMP takes into account the changes in rainfall volumes and intensity that are anticipated as a result of climate change in accordance with Environment Agency guidance. A loss of a significant depth of cover materials through 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

695. 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. Hence this scenario is not considered in the ESC.

Transport Accidents

696. 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 E.3.7) and a transport accident involving leachate sent to a hazardous waste treatment facility is specifically considered (see Section E.3.8).

Criticality Event

697. Criticality and heat generation are processes that are mentioned in the guidance (NS-GRA para. 6.4.21 and 7.3.31). An analysis presented in 2009 (Augean, 2009a) showed that this is not an issue given the very low content of fissile material and very low activity concentrations in the waste disposed at the ENRMF.

E.4.1. Presentation of dose assessments

698. 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 ($\mu Sv\ y^{-1}\ MBq^{-1}$).

699. The site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013) restricts LLW disposal at the ENRMF to 448,000 t at a maximum specific activity of 200 Bq g^{-1}. This constrains disposal of LLW at the ENRMF to a maximum total of 89.6 TBq ($8.96\ 10^{7}$ MBq).

700. The maximum inventory that could be disposed of in the site for each radionuclide is therefore the minimum of 89.6 TBq and the radiological capacity and is therefore not necessarily the same as the radiological capacity. The results of the dose assessments presented in Sections E.4.2 to E.4.5 show the maximum inventory that could be disposed of each radionuclide based on these two constraints and the dose ($\mu Sv\ y^{-1}$) from disposal of that maximum 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 308).

701. 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 G.

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

702. 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).

703. The assessment assumes that the waste is shielded by a 1.6 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.

704. Public exposure to gas emanating from disposed waste could also occur if the restored site is used for recreational use. 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).

705. 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

706. The restored site will include grassland and woodland areas and a permissive footpath 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, equivalent to about 2 hours per day (Oatway & Mobbs, 2003).

707. 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 75 details the habit data assumed for the exposed group. Exposure is assessed both immediately after site restoration (in 2026) and 60 years after closure.

Table 75 Habit data for exposure to gas releases: applicable after the Period of Authorisation

Parameter	Value	Comment
Inhalation rate – public (m^3 h^{-1})	1.0	
Time on site – public (h y^{-1})	750	About 2 hours per day

E.4.2.1. Assessment calculations for recreational site use

708. 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.

709. 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

710. The method in Section E.3.3 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 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).

711. The exposure time (Table 75) 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

712. The external irradiation calculation presented above (Section E.3.4.4) 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

713. 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 76 and Table 77, respectively. Note that the results after 60 years include the effects of ingrowth upon the calculated doses. The expected dose if each radionuclide is disposed at the maximum inventory is also shown. The highest dose at site closure is from C-14 gas (15 µSv y^{-1}), and the dose from wastes disposed of at the ENRMF will always be lower than this due to application of the sum of fractions approach.

714. The dose was also assessed assuming that waste at the maximum activity concentration was disposed at the top layer of the landfill (at 2.6 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 76 Doses to recreational users of restored site at site closure

Radionuclide	Maximum inventory (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose from maximum inventory (µSv y^{-1})
		Gas	External	Total	
H-3	$8.96\ 10^{7}$	$4.86\ 10^{-9}$	0	$4.86\ 10^{-9}$	$4.35\ 10^{-1}$
C-14	$8.96\ 10^{7}$	$1.67\ 10^{-7}$	$1.23\ 10^{-73}$	$1.67\ 10^{-7}$	$1.49\ 10^{1}$

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose from maximum inventory ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
Cl-36	$1.48\ 10^{6}$		$2.66\ 10^{-31}$	$2.66\ 10^{-31}$	$3.93\ 10^{-25}$
Fe-55	$8.96\ 10^{7}$		0	0	0
Co-60	$8.96\ 10^{7}$		$5.23\ 10^{-18}$	$5.23\ 10^{-18}$	$4.68\ 10^{-10}$
Ni-63	$8.96\ 10^{7}$		0	0	0
Sr-90	$8.96\ 10^{7}$		$4.01\ 10^{-28}$	$4.01\ 10^{-28}$	$3.59\ 10^{-20}$
Nb-94	$8.96\ 10^{7}$		$2.79\ 10^{-20}$	$2.79\ 10^{-20}$	$2.50\ 10^{-12}$
Tc-99	$8.96\ 10^{7}$		$5.04\ 10^{-53}$	$5.04\ 10^{-53}$	$4.52\ 10^{-45}$
Ru-106	$8.96\ 10^{7}$		$3.78\ 10^{-22}$	$3.78\ 10^{-22}$	$3.39\ 10^{-14}$
Ag-108m	$8.96\ 10^{7}$		$1.64\ 10^{-21}$	$1.64\ 10^{-21}$	$1.47\ 10^{-13}$
Sb-125	$8.96\ 10^{7}$		$1.21\ 10^{-22}$	$1.21\ 10^{-22}$	$1.08\ 10^{-14}$
Sn-126	$8.96\ 10^{7}$		$1.09\ 10^{-21}$	$1.09\ 10^{-21}$	$9.80\ 10^{-14}$
I-129	$4.17\ 10^{4}$		$3.01\ 10^{-155}$	$3.01\ 10^{-155}$	$1.25\ 10^{-150}$
Ba-133	$8.96\ 10^{7}$		$1.37\ 10^{-25}$	$1.37\ 10^{-25}$	$1.22\ 10^{-17}$
Cs-134	$8.96\ 10^{7}$		$9.88\ 10^{-21}$	$9.88\ 10^{-21}$	$8.85\ 10^{-13}$
Cs-137	$8.96\ 10^{7}$		$1.85\ 10^{-21}$	$1.85\ 10^{-21}$	$1.66\ 10^{-13}$
Pm-147	$8.96\ 10^{7}$		$4.95\ 10^{-52}$	$4.95\ 10^{-52}$	$4.44\ 10^{-44}$
Eu-152	$8.96\ 10^{7}$		$1.58\ 10^{-19}$	$1.58\ 10^{-19}$	$1.42\ 10^{-11}$
Eu-154	$8.96\ 10^{7}$		$2.28\ 10^{-19}$	$2.28\ 10^{-19}$	$2.04\ 10^{-11}$
Eu-155	$8.96\ 10^{7}$		$2.07\ 10^{-44}$	$2.07\ 10^{-44}$	$1.85\ 10^{-36}$
Pb-210	$8.96\ 10^{7}$		$6.86\ 10^{-26}$	$6.86\ 10^{-26}$	$6.15\ 10^{-18}$
Ra-226	$8.96\ 10^{7}$	$1.49\ 10^{-16}$	$1.58\ 10^{-35}$	$1.49\ 10^{-16}$	$1.33\ 10^{-8}$
Ra-228	$8.96\ 10^{7}$		$4.48\ 10^{-16}$	$4.48\ 10^{-16}$	$4.02\ 10^{-8}$
Ac-227	$8.96\ 10^{7}$		$1.48\ 10^{-24}$	$1.48\ 10^{-24}$	$1.32\ 10^{-16}$
Th-229	$8.96\ 10^{7}$		$6.88\ 10^{-22}$	$6.88\ 10^{-22}$	$6.17\ 10^{-14}$
Th-230	$6.93\ 10^{7}$		$1.25\ 10^{-41}$	$1.25\ 10^{-41}$	$8.68\ 10^{-34}$
Th-232	$7.16\ 10^{7}$		$8.39\ 10^{-19}$	$8.39\ 10^{-19}$	$6.01\ 10^{-11}$
Pa-231	$1.86\ 10^{7}$		$6.04\ 10^{-28}$	$6.04\ 10^{-28}$	$1.12\ 10^{-20}$
U-232	$8.96\ 10^{7}$		$1.01\ 10^{-41}$	$1.01\ 10^{-41}$	$9.08\ 10^{-34}$
U-233	$3.13\ 10^{7}$		$2.18\ 10^{-34}$	$2.18\ 10^{-34}$	$6.84\ 10^{-27}$
U-234	$6.41\ 10^{6}$		$1.82\ 10^{-49}$	$1.82\ 10^{-49}$	$1.17\ 10^{-42}$
U-235	$4.92\ 10^{6}$		$5.83\ 10^{-32}$	$5.83\ 10^{-32}$	$2.87\ 10^{-25}$
U-236	$8.96\ 10^{7}$		$4.98\ 10^{-45}$	$4.98\ 10^{-45}$	$4.46\ 10^{-37}$
U-238	$2.53\ 10^{7}$		$1.21\ 10^{-24}$	$1.21\ 10^{-24}$	$3.07\ 10^{-17}$
Np-237	$4.52\ 10^{5}$		$2.96\ 10^{-41}$	$2.96\ 10^{-41}$	$1.34\ 10^{-35}$
Pu-238	$8.96\ 10^{7}$		$1.32\ 10^{-51}$	$1.32\ 10^{-51}$	$1.18\ 10^{-43}$
Pu-239	$8.96\ 10^{7}$		$8.55\ 10^{-34}$	$8.55\ 10^{-34}$	$7.66\ 10^{-26}$
Pu-240	$8.96\ 10^{7}$		$1.18\ 10^{-59}$	$1.18\ 10^{-59}$	$1.06\ 10^{-51}$
Pu-241	$8.96\ 10^{7}$		$2.84\ 10^{-44}$	$2.84\ 10^{-44}$	$2.55\ 10^{-36}$
Pu-242	$8.96\ 10^{7}$		$1.39\ 10^{-72}$	$1.39\ 10^{-72}$	$1.25\ 10^{-64}$

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose from maximum inventory ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
Am-241	$8.96\ 10^{7}$		$1.38\ 10^{-67}$	$1.38\ 10^{-67}$	$1.23\ 10^{-59}$
Cm-243	$8.96\ 10^{7}$		$8.95\ 10^{-32}$	$8.95\ 10^{-32}$	$8.02\ 10^{-24}$
Cm-244	$8.96\ 10^{7}$		0	0	0

Table 77 Doses to recreational users of restored site 60 years after site closure

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose at maximum inventory ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
H-3	$8.96\ 10^{7}$	$6.65\ 10^{-11}$	0	$6.65\ 10^{-11}$	$5.95\ 10^{-3}$
C-14	$8.96\ 10^{7}$	$3.31\ 10^{-8}$	$1.23\ 10^{-73}$	$3.31\ 10^{-8}$	$2.97\ 10^{0}$
Cl-36	$1.48\ 10^{6}$		$2.66\ 10^{-31}$	$2.66\ 10^{-31}$	$3.93\ 10^{-25}$
Fe-55	$8.96\ 10^{7}$		0	0	0
Co-60	$8.96\ 10^{7}$		$1.96\ 10^{-21}$	$1.96\ 10^{-21}$	$1.75\ 10^{-13}$
Ni-63	$8.96\ 10^{7}$		0	0	0
Sr-90	$8.96\ 10^{7}$		$9.46\ 10^{-29}$	$9.46\ 10^{-29}$	$8.48\ 10^{-21}$
Nb-94	$8.96\ 10^{7}$		$2.79\ 10^{-20}$	$2.79\ 10^{-20}$	$2.50\ 10^{-12}$
Tc-99	$8.96\ 10^{7}$		$5.04\ 10^{-53}$	$5.04\ 10^{-53}$	$4.52\ 10^{-45}$
Ru-106	$8.96\ 10^{7}$		$8.36\ 10^{-40}$	$8.36\ 10^{-40}$	$7.49\ 10^{-32}$
Ag-108m	$8.96\ 10^{7}$		$1.49\ 10^{-21}$	$1.49\ 10^{-21}$	$1.33\ 10^{-13}$
Sb-125	$8.96\ 10^{7}$		$3.43\ 10^{-29}$	$3.43\ 10^{-29}$	$3.08\ 10^{-21}$
Sn-126	$8.96\ 10^{7}$		$1.09\ 10^{-21}$	$1.09\ 10^{-21}$	$9.80\ 10^{-14}$
I-129	$4.17\ 10^{4}$		$3.01\ 10^{-155}$	$3.01\ 10^{-155}$	$1.25\ 10^{-150}$
Ba-133	$8.96\ 10^{7}$		$2.62\ 10^{-27}$	$2.62\ 10^{-27}$	$2.35\ 10^{-19}$
Cs-134	$8.96\ 10^{7}$		$1.77\ 10^{-29}$	$1.77\ 10^{-29}$	$1.59\ 10^{-21}$
Cs-137	$8.96\ 10^{7}$		$4.67\ 10^{-22}$	$4.67\ 10^{-22}$	$4.19\ 10^{-14}$
Pm-147	$8.96\ 10^{7}$		$6.44\ 10^{-59}$	$6.44\ 10^{-59}$	$5.77\ 10^{-51}$
Eu-152	$8.96\ 10^{7}$		$7.33\ 10^{-21}$	$7.33\ 10^{-21}$	$6.57\ 10^{-13}$
Eu-154	$8.96\ 10^{7}$		$1.80\ 10^{-21}$	$1.80\ 10^{-21}$	$1.62\ 10^{-13}$
Eu-155	$8.96\ 10^{7}$		$3.32\ 10^{-48}$	$3.32\ 10^{-48}$	$2.98\ 10^{-40}$
Pb-210	$8.96\ 10^{7}$		$1.05\ 10^{-26}$	$1.05\ 10^{-26}$	$9.45\ 10^{-19}$
Ra-226	$8.96\ 10^{7}$	$1.45\ 10^{-16}$	$1.26\ 10^{-35}$	$1.45\ 10^{-16}$	$1.30\ 10^{-8}$
Ra-228	$8.96\ 10^{7}$		$3.24\ 10^{-19}$	$3.24\ 10^{-19}$	$2.90\ 10^{-11}$
Ac-227	$8.96\ 10^{7}$		$2.19\ 10^{-25}$	$2.19\ 10^{-25}$	$1.96\ 10^{-17}$
Th-229	$8.96\ 10^{7}$		$6.84\ 10^{-22}$	$6.84\ 10^{-22}$	$6.13\ 10^{-14}$
Th-230	$6.93\ 10^{7}$		$3.31\ 10^{-37}$	$3.31\ 10^{-37}$	$2.29\ 10^{-29}$
Th-232	$7.16\ 10^{7}$		$8.39\ 10^{-19}$	$8.39\ 10^{-19}$	$6.01\ 10^{-11}$

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose at maximum inventory ($\mu Sv\ y^{-1}$)
		Gas	External	Total	
Pa-231	$1.86\ 10^{7}$		$1.25\ 10^{-24}$	$1.25\ 10^{-24}$	$2.33\ 10^{-17}$
U-232	$8.96\ 10^{7}$		$5.54\ 10^{-42}$	$5.54\ 10^{-42}$	$4.96\ 10^{-34}$
U-233	$3.13\ 10^{7}$		$3.89\ 10^{-24}$	$3.89\ 10^{-24}$	$1.22\ 10^{-16}$
U-234	$6.41\ 10^{6}$		$6.76\ 10^{-45}$	$6.76\ 10^{-45}$	$4.33\ 10^{-38}$
U-235	$4.92\ 10^{6}$		$8.26\ 10^{-31}$	$8.26\ 10^{-31}$	$4.06\ 10^{-24}$
U-236	$8.96\ 10^{7}$		$2.49\ 10^{-27}$	$2.49\ 10^{-27}$	$2.23\ 10^{-19}$
U-238	$2.53\ 10^{7}$		$1.21\ 10^{-24}$	$1.21\ 10^{-24}$	$3.07\ 10^{-17}$
Np-237	$4.52\ 10^{5}$		$5.75\ 10^{-38}$	$5.75\ 10^{-38}$	$2.60\ 10^{-32}$
Pu-238	$8.96\ 10^{7}$		$8.45\ 10^{-52}$	$8.45\ 10^{-52}$	$7.57\ 10^{-44}$
Pu-239	$8.96\ 10^{7}$		$8.53\ 10^{-34}$	$8.53\ 10^{-34}$	$7.65\ 10^{-26}$
Pu-240	$8.96\ 10^{7}$		$8.82\ 10^{-51}$	$8.82\ 10^{-51}$	$7.90\ 10^{-43}$
Pu-241	$8.96\ 10^{7}$		$1.57\ 10^{-45}$	$1.57\ 10^{-45}$	$1.40\ 10^{-37}$
Pu-242	$8.96\ 10^{7}$		$1.13\ 10^{-32}$	$1.13\ 10^{-32}$	$1.01\ 10^{-24}$
Am-241	$8.96\ 10^{7}$		$5.49\ 10^{-46}$	$5.49\ 10^{-46}$	$4.92\ 10^{-38}$
Cm-243	$8.96\ 10^{7}$		$2.14\ 10^{-32}$	$2.14\ 10^{-32}$	$1.92\ 10^{-24}$
Cm-244	$8.96\ 10^{7}$		$2.93\ 10^{-62}$	$2.93\ 10^{-62}$	$2.62\ 10^{-54}$

715. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.4.

E.4.3. Post PoA Groundwater abstraction – existing borehole

716. The contamination of groundwater under the landfill is expected to occur at some point in the future. The HRA shows degradation of the landfill liner and cap over time resulting in leachate flows to the underlying substrate and then to groundwater. This scenario considers the exposures resulting from contaminated groundwater taken from the nearest licensed abstraction point, taken to be 1.5 km from the site.

717. If the 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. However, groundwater does not discharge to a watercourse that is closer to the landfill than the abstraction point; any discharges to a more distant watercourse would be subject to additional dilution by groundwater, surface runoff and drainage water thereby reducing exposure relative to the extraction point.

718. 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

719. Groundwater abstraction is expected to continue at the nearest borehole to the site and it is assumed to access groundwater within the Lincolnshire Limestone. The

nearest licensed water abstraction point in the direction of groundwater flows is at Law's Lawn, about 1.5 km south east of the centre of waste cells used for LLW disposal. Although this has only been used for farm activities in the past, it is currently licenced for potable water.

720. Exposure of members of the public is assumed to occur as a result of using well 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 food produced on irrigated land including milk, green vegetables, root vegetables and meat products;
- consumption of drinking water from the borehole;
- external irradiation from radionuclides incorporated in contaminated soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

721. The same habit assumptions applied in Section E.3.4.4 for an adult in a farming family have been used for this scenario (see Table 53). The drinking water consumption rate for adults used in the assessment is 600 l y^{-1} (Smith & Jones, 2003).

E.4.3.1. Assessment calculations for use of groundwater at nearest licensed abstraction point

722. The GoldSim model used to model the groundwater migration scenario also includes a soil compartment which receives inputs from irrigation water 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} in this area of Northamptonshire. 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.

723. The groundwater and exposure models are described in Section E.3.4. The peak activity concentration in the groundwater over the assessment time period (100,000 years) is used to calculate the doses to the exposed group.

E.4.3.2. Dose to farming family exposed to groundwater extracted at existing well

724. Specific dose calculations were undertaken for water extracted at an existing well location. Table 78 sets out the calculated results. The third column in the table gives the maximum dose for the drinking water pathway and the fourth column gives the maximum dose due to the irrigation pathway. The fifth column gives the sum of the maximum doses for both pathways. The sixth column gives the point in time when the maximum dose occurs. The maximum inventory for each radionuclide, and the corresponding dose from disposal of the maximum inventory are also shown. The results for Ra-226 are independent of the Ra-226 placement depth in the site.

725. GoldSim output has a low value cut-off and shows a lower limit of 1 10^{-10} µSv y^{-1} MBq^{-1}, which occurs for short lived radionuclides (half-life of less than about 5 years) where radioactive decay reduces activity to very low levels or or where there is limited radionuclide transport in groundwater.

726. The quantities of long-lived daughters that have ingrown from specific parents or were directly disposed are distinguished in these calculations. For example, the model considers four 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; and,
- U-234 ingrown from Pu-242.

727. Dose factors are presented in Table 170 that include the contribution of all short-lived daughters assuming that those daughters are in secular equilibrium. For example, the dose factor for U-238 includes the contributions from Th-234, Pa-234m and Pa-234.

728. Reported doses of parent radionuclides include the contributions of all daughters, ingrown from the parent. For example, the dose from U-238 includes contributions from the variants of U-234, Th-230, Ra-226 and Pb-210, specific for ingrowth from U-238.

Table 78 Doses for adults based on a unit inventory of 1 MBq and the maximum inventory - well at 1500 metres from the edge of the landfill.

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway (µSv y^{-1} MBq^{-1})	Maximum calculated dose for the irrigation pathway (µSv y^{-1} MBq^{-1})	Sum (µSv y^{-1} MBq^{-1})	Time of Maximum (y)	Dose from maximum inventory (µSv y^{-1})
H-3	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
C-14	8.96 10^{7}	1.23 10^{-10}	1.20 10^{-9}	1.32 10^{-9}	7,930	1.18 10^{-1}
Cl-36	1.48 10^{6}	1.03 10^{-6}	4.76 10^{-6}	5.79 10^{-6}	759	8.56
Fe-55	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Co-60	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Ni-63	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Sr-90	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Nb-94	8.96 10^{7}	<1.0 10^{-10}	9.55 10^{-10}	9.55 10^{-10}	28,755	8.56 10^{-2}
Tc-99	8.96 10^{7}	1.02 10^{-8}	4.38 10^{-8}	5.40 10^{-8}	5,205	4.84
Ru-106	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Ag-108m	8.96 10^{7}	<1.0 10^{-10}	3.29 10^{-10}	3.29 10^{-10}	746	2.95 10^{-2}
Sb-125	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}
Sn-126	8.96 10^{7}	2.70 10^{-9}	3.62 10^{-8}	3.89 10^{-8}	100,000	3.49
I-129	4.17 10^{4}	4.31 10^{-5}	1.62 10^{-4}	2.05 10^{-4}	2,100	8.56
Ba-133	8.96 10^{7}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}		<9.0 10^{-3}

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Maximum calculated dose for the irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Sum ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of Maximum (y)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Cs-134	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Cs-137	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Pm-147	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-152	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-154	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Ra-226	$8.96\ 10^{7}$	$4.31\ 10^{-10}$	$1.76\ 10^{-9}$	$2.19\ 10^{-9}$	2,450	$1.96\ 10^{-1}$
Ra-228	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Ac-227	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Th-229	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$1.57\ 10^{-8}$	$1.57\ 10^{-8}$	10,725	1.41
Th-230	$6.93\ 10^{7}$	$4.06\ 10^{-9}$	$2.53\ 10^{-8}$	$2.94\ 10^{-8}$	100,000	2.04
Th-232	$7.16\ 10^{7}$	$2.07\ 10^{-9}$	$5.04\ 10^{-8}$	$5.24\ 10^{-8}$	100,000	3.76
Pa-231	$1.86\ 10^{7}$	$6.26\ 10^{-9}$	$3.21\ 10^{-8}$	$3.84\ 10^{-8}$	46,800	$7.13\ 10^{-1}$
U-232	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
U-233	$3.13\ 10^{7}$	$5.20\ 10^{-8}$	$2.21\ 10^{-7}$	$2.73\ 10^{-7}$	100,000	8.56
U-234	$6.41\ 10^{6}$	$1.87\ 10^{-7}$	$1.14\ 10^{-6}$	$1.33\ 10^{-6}$	100,000	8.53
U-235	$4.92\ 10^{6}$	$2.86\ 10^{-7}$	$1.45\ 10^{-6}$	$1.73\ 10^{-6}$	100,000	8.52
U-236	$8.96\ 10^{7}$	$1.26\ 10^{-8}$	$4.69\ 10^{-8}$	$5.95\ 10^{-8}$	100,000	5.33
U-238	$2.53\ 10^{7}$	$5.32\ 10^{-8}$	$2.83\ 10^{-7}$	$3.37\ 10^{-7}$	100,000	8.53
Np-237	$4.52\ 10^{5}$	$4.00\ 10^{-6}$	$1.49\ 10^{-5}$	$1.89\ 10^{-5}$	26,095	8.55
Pu-238	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$2.86\ 10^{-10}$	$2.86\ 10^{-10}$	271	$2.56\ 10^{-2}$
Pu-239	$8.96\ 10^{7}$	$5.80\ 10^{-10}$	$2.20\ 10^{-9}$	$2.78\ 10^{-9}$	34,575	$2.49\ 10^{-1}$
Pu-240	$8.96\ 10^{7}$	$1.57\ 10^{-10}$	$4.90\ 10^{-10}$	$6.48\ 10^{-10}$	9,590	$5.80\ 10^{-2}$
Pu-241	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Pu-242	$8.96\ 10^{7}$	$3.55\ 10^{-9}$	$1.38\ 10^{-8}$	$1.73\ 10^{-8}$	100,000	1.55
Am-241	$8.96\ 10^{7}$	$8.06\ 10^{-10}$	$3.01\ 10^{-9}$	$3.81\ 10^{-9}$	768	$3.42\ 10^{-1}$
Cm-243	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Cm-244	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$

729. The time to peak dose varies from 271 to 100,000 years and daughter ingrowth is calculated at the time of peak dose.

730. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.4.

E.4.4. Groundwater abstraction – borehole at site boundary

731. Groundwater abstraction at the site boundary has been assessed after the period of authorisation. The groundwater and exposure models are described in Section E.3.4.

The peak activity concentration in the groundwater over the assessment time period (100,000 years) is used to calculate the doses to the exposed group.

E.4.4.1. Assessment calculation of groundwater abstracted at the boundary of the landfill

732. The construction of a water abstraction borehole at the boundary of the site is not expected to penetrate or damage the integrity of the liner or cap. Where a geological barrier contributed to environmental safety, the 2009 guidance gave some discretion in determining to what distance it was appropriate to apply the dose guidance level for human intrusion to a well that penetrated the geological barrier (NS-GRA (Environment Agencies, 2009); paragraph 6.3.44). In 2012, the Environment Agency issued further guidance to incorporate requirements of the Groundwater Directive (Environment Agency, 2012a).

733. Specifically:

- (i) Requirement R5: Dose constraints during the period of authorisation

We shall require the developer or operator of a radioactive waste disposal facility in all cases to show that the radiation dose to members of the public through the groundwater pathway during the period of authorisation of the facility is consistent with, or lower than, a dose guidance level of 20 µSv y^{-1}. The means of doing so may be proportionate to the radiological hazard presented by the waste at these times.

- (ii) Requirement R6: Risk guidance level after the period of authorisation

We shall require the developer or operator of a radioactive waste disposal facility in all cases to show that the radiological risk to members of the public through the groundwater pathway after the period of authorisation of the facility is consistent with, or lower than, a risk guidance level of 10^{-6} per year. The means of doing so may be proportionate to the radiological hazard presented by the waste at these times.

734. This modification to the guidance removes the discretion that could be applied in applying the dose guidance level for human intrusion (NS-GRA (Environment Agencies, 2009); paragraph 6.3.44) to a well that penetrates a geological barrier that contributes to environmental safety. Hence a well at the site boundary after the period of authorisation is assessed against the criteria given in the above paragraph.

735. The scenario considering abstraction at a well at the site boundary differs from the groundwater scenario using the nearest abstraction point (a scenario that is expected to occur) due to less travel time and less dispersion within the aquifer.

736. Exposure of members of the public is assumed to occur as a result of using well 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. This scenario considers the exposures resulting from water taken from a hypothetical new abstraction point at the site boundary.

737. 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).

E.4.4.2. Dose to farming family exposed to groundwater extracted at site boundary

738. The results of the dose calculations for water extracted at a well located at the site boundary are given in Table 79. The third column in the table gives the maximum dose for the drinking water pathway and the fourth column gives the maximum dose due to the irrigation pathway. The fifth column gives the sum of the maximum doses for both pathways. The sixth column gives the point in time when the maximum dose occurs. The maximum inventory for each radionuclide, and the corresponding dose from disposal of the maximum inventory, are also shown. The results for Ra-226 are independent of the Ra-226 placement depth in the site.Doses are greater than those calculated at the location of the existing well. This is due to the assumed dispersion within the Lincolnshire limestone. The results for Ra-226 are independent of the Ra-226 placement depth in the site.

Table 79 Maximum annual doses for adults, based on a unit inventory of 1 MBq for each radionuclide and a well at the site boundary.

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Maximum calculated dose for the irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Sum ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of Max (y)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
C-14	$8.96\ 10^{7}$	$2.87\ 10^{-10}$	$3.21\ 10^{-9}$	$3.49\ 10^{-9}$	7,930	$3.13\ 10^{-1}$
Cl-36	$1.48\ 10^{6}$	$2.42\ 10^{-6}$	$1.11\ 10^{-5}$	$1.35\ 10^{-5}$	759	$2.00\ 10^{1}$
Fe-55	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Co-60	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Ni-63	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Nb-94	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$2.23\ 10^{-9}$	$2.23\ 10^{-9}$	28,755	$2.00\ 10^{-1}$
Tc-99	$8.96\ 10^{7}$	$2.37\ 10^{-8}$	$1.02\ 10^{-7}$	$1.26\ 10^{-7}$	5,205	$1.13\ 10^{1}$
Ru-106	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Ag-108m	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$9.03\ 10^{-10}$	$9.03\ 10^{-10}$	746	$8.10\ 10^{-2}$
Sb-125	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Sn-126	$8.96\ 10^{7}$	$6.32\ 10^{-9}$	$8.47\ 10^{-8}$	$9.10\ 10^{-8}$	100,000	8.16
I-129	$4.17\ 10^{4}$	$1.01\ 10^{-4}$	$3.79\ 10^{-4}$	$4.80\ 10^{-4}$	2,100	$2.00\ 10^{1}$
Ba-133	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Cs-134	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Cs-137	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Pm-147	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-152	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-154	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Ra-226	$8.96\ 10^{7}$	$1.01\ 10^{-9}$	$4.13\ 10^{-9}$	$5.15\ 10^{-9}$	2,450	$4.61\ 10^{-1}$
Ra-228	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose for the drinking water pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Maximum calculated dose for the irrigation pathway ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Sum ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Time of Max (y)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Ac-227	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Th-229	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$3.68\ 10^{-8}$	$3.68\ 10^{-8}$	10,725	3.30
Th-230	$6.93\ 10^{7}$	$9.53\ 10^{-9}$	$5.99\ 10^{-8}$	$6.94\ 10^{-8}$	100,000	4.81
Th-232	$7.16\ 10^{7}$	$4.88\ 10^{-9}$	$1.18\ 10^{-7}$	$1.23\ 10^{-7}$	100,000	8.81
Pa-231	$1.86\ 10^{7}$	$1.47\ 10^{-8}$	$7.55\ 10^{-8}$	$9.02\ 10^{-8}$	46,800	1.68
U-232	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
U-233	$3.13\ 10^{7}$	$1.21\ 10^{-7}$	$5.17\ 10^{-7}$	$6.38\ 10^{-7}$	100,000	$2.00\ 10^{1}$
U-234	$6.41\ 10^{6}$	$4.39\ 10^{-7}$	$2.68\ 10^{-6}$	$3.12\ 10^{-6}$	100,000	$2.00\ 10^{1}$
U-235	$4.92\ 10^{6}$	$6.73\ 10^{-7}$	$3.40\ 10^{-6}$	$4.07\ 10^{-6}$	100,000	$2.00\ 10^{1}$
U-236	$8.96\ 10^{7}$	$2.94\ 10^{-8}$	$1.10\ 10^{-7}$	$1.39\ 10^{-7}$	100,000	$1.25\ 10^{1}$
U-238	$2.53\ 10^{7}$	$1.25\ 10^{-7}$	$6.64\ 10^{-7}$	$7.89\ 10^{-7}$	100,000	$2.00\ 10^{1}$
Np-237	$4.52\ 10^{5}$	$9.35\ 10^{-6}$	$3.49\ 10^{-5}$	$4.43\ 10^{-5}$	26,095	$2.00\ 10^{1}$
Pu-238	$8.96\ 10^{7}$	$1.57\ 10^{-10}$	$6.71\ 10^{-10}$	$8.28\ 10^{-10}$	271	$7.42\ 10^{-2}$
Pu-239	$8.96\ 10^{7}$	$1.36\ 10^{-9}$	$5.26\ 10^{-9}$	$6.62\ 10^{-9}$	34,575	$5.93\ 10^{-1}$
Pu-240	$8.96\ 10^{7}$	$3.68\ 10^{-10}$	$1.15\ 10^{-9}$	$1.51\ 10^{-9}$	9,590	$1.36\ 10^{-1}$
Pu-241	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$1.92\ 10^{-10}$	$1.92\ 10^{-10}$	53	$1.72\ 10^{-2}$
Pu-242	$8.96\ 10^{7}$	$8.31\ 10^{-9}$	$3.23\ 10^{-8}$	$4.06\ 10^{-8}$	100,000	3.64
Am-241	$8.96\ 10^{7}$	$1.89\ 10^{-9}$	$7.03\ 10^{-9}$	$8.91\ 10^{-9}$	768	$7.99\ 10^{-1}$
Cm-243	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$
Cm-244	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$	$<1.0\ 10^{-10}$		$<9.0\ 10^{-3}$

739. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.4.

E.4.5. Bathtubbing

740. This scenario has been included to account for the possibility of excessive infiltration through the cap at a time when the engineered barrier still prevents leakage to the underlying formation. Monitoring and control of leachate levels will ensure that bathtubbing does not occur during the period of authorisation. For a hazardous waste site, it is envisaged that controls on cap construction and leachate monitoring and management would prevent releases through this pathway, and the following explanation is provided in the HRA for hazardous waste landfill.

Leachate level monitoring will continue following completion of filling, capping and placement of the restoration materials. Leachate levels will be controlled as necessary so that compliance limits are not exceeded. The control of leachate levels at the site will continue until it is considered by the Environment Agency that the landfill is unlikely to present a significant risk to the environment if leachate management ceases.

Based on the HRA even following the cessation of active leachate management regulatory control at the site will be maintained through the Environmental

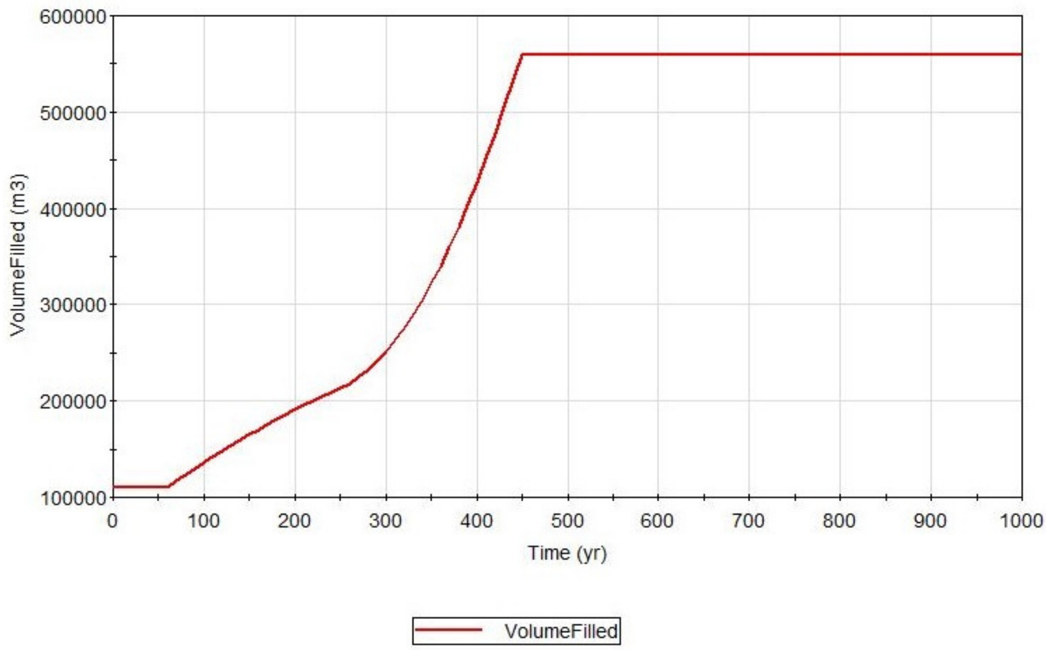
Permit. The Environmental Permit cannot be surrendered until the Environment Agency consider that the site no longer presents a potential risk to groundwater. On this basis the potential for overtopping of leachate at a stage when the leachate could have an unacceptable impact on the environment is unlikely to occur.

741. However, the pathway has been assessed in the ESC in order to illustrate what could occur. The Goldsim model assumes that cap degradation begins after 250 years and the cap is fully degraded after 1000 years. The assessment models potential filling of the waste cells after the period of authorisation, as the infiltration rate (through the cap) is higher than the leaching rate (through the basal liner). The waste cells are estimated to be filled completely with leachate after 450 years, at which time the occurrence of a bathtubbing event is modelled (see figure below).

742. The scenario is based on characteristics similar to those for the residential occupation group considered above. We have conservatively assumed that any water that overspills from the landfill is not diluted by any other standing or draining water around the site.

743. 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). Hence use of this dose criterion is conservative since this scenario is not expected to occur.

Figure 18. Filling of the waste cells with leachate prior to the potential occurrence of a bathtubbing event.



Potentially exposed group

744. Bathtubbing results in leachate spilling over the top of the landfill liner at the sides of the landfill. The release is assumed to inundate sub-soil in 3 ha of surrounding land, with a proportion of the release accumulating in the root zone of plants and the remainder draining to groundwater.

745. The water input rate to land is assumed to be 600 m^3 ha^{-1}. This is comparable to an irrigation rate. Contamination of freshwater streams or water bodies is not considered. It is assumed that the bathtubbing event occurs 450 years after site closure.

746. 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;
- external irradiation from radionuclides incorporated in contaminated soil;
- inadvertent inhalation of contaminated dust; and,
- inadvertent ingestion of contaminated soil.

747. The relevant habit assumptions applied in Section E.3.4.4 for an adult in a farming family have been used for these exposure pathways (see Table 53). Assessment calculations for a residential family exposed as a result of bathtubbing

748. The groundwater and exposure models are described in Section E.3.4. A bathtubbing scenario is assumed to occur at a nominal time, 450 years after the start of cap degradation. During the 450 years the landfill gradually fills up with water once the inflow from precipitation through the degrading cap is larger than the outflow through the liner. After 450 years, the landfill is assumed to be saturated to the height of the wall liners. 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.

749. The model for 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 none of the crops intercept water through the leaves.

750. There are no local hydrological features that suggest there will be a build-up of surface water following overtopping, the local fields are well drained and there is one minor surface drainage water channel to the south and east of the site (downslope). The restored site will have drainage channels near the boundary to collect excess surface water and direct this to constructed ponds and then to natural drainage channels to the northwest and southeast of the site. It is considered likely that overtopping will drain to sub-soil rather than flood and saturate an extensive area or percolate to the site drainage channels which may have degraded after 450 years.

751. The scenario assumes that 3 ha around the site is subject to an inundation event due to bathtubbing; this is a small area relative to the size of the landfill and all activity is assumed to accumulate in the affected area. Seepage will occur at the top of the

side liner and this will be at least 1 m below ground level. It is also assumed that 1% of the activity introduced at depth (>1 m) reaches the cultivated surface soils (Shaw, et al., 2004). The remainder is assumed to drain to sub-strata based on the good drainage observed in the surrounding area. Absorption within the sub-strata is expected to result in a lower concentration in groundwater than modelled for the abstraction scenarios and this has therefore been ignored in the bathtubbing scenario. No account is taken of potential dilution by rain falling in the surrounding area and draining to the same point. The doses are calculated for a household.

752. 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 and 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.

Element and Radionuclide Specific Parameters

753. Radionuclide specific dose coefficients are shown in Table 170 and element specific parameters for plant and animal uptake are specified in Table 172 and Table 173. Note that the Ra-226 and Th-232 dose coefficients used in the groundwater model are shown in Table 54 since their daughter radionuclides are modelled explicitly.

E.4.5.1. Dose to residential family as a result of bathtubbing

754. The results of the dose calculations for inundation of land next to the landfill site following failure of the cap leading to saturation of a waste cell and overtopping of the side liner are given below. The maximum inventory for each radionuclide and the dose from disposal of the maximum inventory are also shown.A time of 450 years post closure was the time suggested from the Goldsim model outputs. The results for Ra-226 are independent of the Ra-226 placement depth in the site.

Table 80 Maximum annual doses for adults, based on a unit inventory of 1 MBq for each radionuclide and overtopping of the side liner

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
C-14	$8.96\ 10^{7}$	$1.10\ 10^{-11}$	$9.86\ 10^{-4}$
Cl-36	$1.48\ 10^{6}$	$1.25\ 10^{-7}$	$1.85\ 10^{-1}$
Fe-55	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Co-60	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ni-63	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$1.58\ 10^{-12}$	$1.41\ 10^{-4}$

Radionuclide	Maximum inventory (MBq)	Maximum calculated dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Nb-94	$8.96\ 10^{7}$	$3.45\ 10^{-9}$	$3.09\ 10^{-1}$
Tc-99	$8.96\ 10^{7}$	$2.02\ 10^{-7}$	$1.81\ 10^{1}$
Ru-106	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ag-108m	$8.96\ 10^{7}$	$2.93\ 10^{-9}$	$2.63\ 10^{-1}$
Sb-125	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Sn-126	$8.96\ 10^{7}$	$1.21\ 10^{-9}$	$1.08\ 10^{-1}$
I-129	$4.17\ 10^{4}$	$1.59\ 10^{-7}$	$6.64\ 10^{-3}$
Ba-133	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Cs-134	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Cs-137	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pm-147	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-152	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-154	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ra-226	$8.96\ 10^{7}$	$2.74\ 10^{-9}$	$2.46\ 10^{-1}$
Ra-228	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Ac-227	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Th-229	$8.96\ 10^{7}$	$5.26\ 10^{-11}$	$4.71\ 10^{-3}$
Th-230	$6.93\ 10^{7}$	$5.74\ 10^{-10}$	$3.98\ 10^{-2}$
Th-232	$7.16\ 10^{7}$	$3.08\ 10^{-9}$	$2.21\ 10^{-1}$
Pa-231	$1.86\ 10^{7}$	$1.55\ 10^{-9}$	$2.88\ 10^{-2}$
U-232	$8.96\ 10^{7}$	$5.16\ 10^{-12}$	$4.63\ 10^{-4}$
U-233	$3.13\ 10^{7}$	$1.18\ 10^{-10}$	$3.70\ 10^{-3}$
U-234	$6.41\ 10^{6}$	$8.76\ 10^{-11}$	$5.62\ 10^{-4}$
U-235	$4.92\ 10^{6}$	$1.42\ 10^{-9}$	$6.99\ 10^{-3}$
U-236	$8.96\ 10^{7}$	$7.95\ 10^{-11}$	$7.12\ 10^{-3}$
U-238	$2.53\ 10^{7}$	$3.09\ 10^{-10}$	$7.83\ 10^{-3}$
Np-237	$4.52\ 10^{5}$	$1.88\ 10^{-8}$	$8.50\ 10^{-3}$
Pu-238	$8.96\ 10^{7}$	$7.99\ 10^{-13}$	$7.16\ 10^{-5}$
Pu-239	$8.96\ 10^{7}$	$3.47\ 10^{-11}$	$3.11\ 10^{-3}$
Pu-240	$8.96\ 10^{7}$	$3.35\ 10^{-11}$	$3.00\ 10^{-3}$
Pu-241	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Pu-242	$8.96\ 10^{7}$	$3.28\ 10^{-11}$	$2.94\ 10^{-3}$
Am-241	$8.96\ 10^{7}$	$6.89\ 10^{-12}$	$6.17\ 10^{-4}$
Cm-243	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$
Cm-244	$8.96\ 10^{7}$	$<1.0\ 10^{-10}$	$<9.0\ 10^{-3}$

755. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.4.

E.5. Human intrusion scenarios {R7}

756. 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.

757. 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, 2009a), (Environment Agencies, 2009), (Environment Agency, 2012a)] has identified five potential intrusion scenarios and nine potentially exposed groups likely to be of relevance to the ENRMF. 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).

758. 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;
- Laboratory analyst working with the borehole or trial pit samples: 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
- Smallholder excavating on the site (at 200 years): dose to smallholder.

759. In Table 81 descriptions of these human intrusion cases based on LLWR assessments (Hicks & Baldwin, 2011) are presented.

Table 81 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.

760. The impact assessment undertaken on behalf of the LLWR (LLWR, 2011b) suggests that house occupancy and a smallholding on site are likely to offer the highest doses to exposed persons, followed by the borehole laboratory analyst and 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 the ENRMF in this assessment.

761. For simplicity, and as an additionally cautious approach, it is assumed that the smallholder is also resident on site, thus combining the house occupancy and smallholding food consumption pathways.

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

763. 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 a number of ways, including: the quantity of excavated waste, habit data and the time when intrusion is assumed to occur.

764. 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. 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 wastes, the overall specific activity (activity concentration) may be less than 200 Bq g^{-1} but the activity concentration within certain fractions of the waste may exceed 200 Bq g^{-1}.

765. A number of different large items 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). Sensitivity to assumed depth profiles for distribution of activity is explored and recommendations on waste acceptance criteria are presented.

766. 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.

767. 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, possibly because they were disposed of under earlier VLLW authorisations, and excavates waste during final site restoration works. In the case of the ENRMF, 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.

768. 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 (ref IAEA Tecdoc 1380). 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.

769. In Table 82 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.5.11.

Table 82 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.
Borehole drilling: laboratory analyst	Inhalation of contaminated dust	Laboratory inspection of contaminated soil samples generates dust. Analyst inhales dust during analysis.
	Ingestion of contaminated material	Analyst ingests contaminated material during laboratory analysis.
	External irradiation	The analyst is exposed to external irradiation while analysing contaminated samples.
	Radon inhalation	The analyst is exposed to radon gas emanating from the sample.
Trial pit excavation	Inhalation of contaminated dust	Dust generated by trial pit intrusion into waste includes radioactive material. Operative 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 drilling activities. A worker in close proximity to this material is exposed to external irradiation.
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 drilling activities. A worker in close proximity to this material is exposed to external irradiation.
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.

Event/scenario	Exposure pathway	Description
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.
	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.
Excavation of particles and objects	Ingestion of contaminated material	Inadvertent ingestion of a particle or contaminated dust by a site occupant or worker.
	External irradiation	Exposure to a particle on the skin or from large contaminated objects lying on the surface.

E.5.1. Presentation of dose assessments

770. 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}).

771. The site Development Consent Order (The East Northamptonshire Resource Management Facility Order, 2013) restricts LLW disposal at the ENRMF to 448,000 t at a maximum specific activity of 200 Bq g^{-1}. This constrains disposal of LLW at the ENRMF to a maximum total of 89.6 TBq (8.96 10^7 MBq).

772. The maximum inventory that could be disposed of in the site for each radionuclide is therefore the minimum of 89.6 TBq and the radiological capacity and is therefore not necessarily the same as the radiological capacity. The results of the dose assessments presented in Sections E.5.2 to E.5.7 show the maximum inventory that could be disposed of each radionuclide based on these two constraints, and the dose (µSv y^{-1}) from disposal of that maximum 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 308).

773. 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 G.

E.5.2. Borehole drilling – Drill Operative

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

774. 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):

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

where:

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

- 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}).

775. The waste density and porosity are given in Table 46. Note that there is a site constraint that LLW tonnage is not to exceed 448,000 t of the total disposed tonnage up to 31st December 2026 or its earlier closure date (The East Northamptonshire Resource Management Facility Order, 2013). On this basis LLW will comprise about 20% of the waste disposed at the ENRMF.

776. 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}).

777. Assumptions regarding the liner are given in Table 44. During the landfill's operational period, $q_{barrier}$ is set equal to q_{out}.

778. 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

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

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

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

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});
- $D_{irr,slab}$, 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 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

780. 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.

781. 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, 2009a)];
- 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, 2009a)];
- 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).

782. For the face, this is given by:

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

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

783. Note that Borehole driller assessment calculation is cautious as no account is taken of non-contaminated cap material that may also be excavated and will act so as to dilute the radioactive material.

Table 83 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

Parameter	Units	Value	Description
			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 33	Volume of landfill (cells) in which activity is homogeneously distributed
ρ_{waste}	$kg\ m^{-3}$	See Table 46	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 (Hicks & Baldwin, 2011).

** Values from (US EPA, 2014).

Values taken from (Augean, 2009a), unless otherwise stated.

784. 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

785. In Table 84 the dose rates to borehole drill operatives ($\mu Sv\ y^{-1}\ MBq^{-1}$) involved in excavating waste at the ENRMF 60 years after capping are presented. The 60 years after capping is immediately at the end of the period of authorisation. The maximum inventory for each radionuclide and the dose from disposal of that maximum inventory are also given.

786. The largest dose rates per MBq disposal are for Ra-226, Nb-94, Pa-231 and Th-232. These radionuclides will correspondingly have the smallest radiological capacities under this scenario. Radiological capacity calculations are presented in Section 7.4. The impact of Radium placement depth within the ENRMF on intrusion and radon release is discussed in Section E.5.8. No Ra-226 emplacement depth restrictions are assumed in the calculation of the doses to the borehole drill operative.

Table 84 Dose to Borehole driller excavating at the site

Radionuclide	Maximum inventory (MBq)	Dose to Borehole excavator (60y) ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$5.91\ 10^{-13}$	$5.29\ 10^{-5}$

Radionuclide	Maximum inventory (MBq)	Dose to Borehole excavator (60y) ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
C-14	$8.96\ 10^{7}$	$5.88\ 10^{-10}$	$5.27\ 10^{-2}$
Cl-36	$1.48\ 10^{6}$	$4.29\ 10^{-9}$	$6.34\ 10^{-3}$
Fe-55	$8.96\ 10^{7}$	$5.15\ 10^{-17}$	$4.61\ 10^{-9}$
Co-60	$8.96\ 10^{7}$	$7.84\ 10^{-9}$	$7.02\ 10^{-1}$
Ni-63	$8.96\ 10^{7}$	$7.90\ 10^{-11}$	$7.08\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$1.27\ 10^{-8}$	1.14
Nb-94	$8.96\ 10^{7}$	$1.25\ 10^{-5}$	$1.12\ 10^{3}$
Tc-99	$8.96\ 10^{7}$	$1.07\ 10^{-9}$	$9.55\ 10^{-2}$
Ru-106	$8.96\ 10^{7}$	$3.70\ 10^{-24}$	$3.32\ 10^{-16}$
Ag-108m	$8.96\ 10^{7}$	$1.13\ 10^{-5}$	$1.01\ 10^{3}$
Sb-125	$8.96\ 10^{7}$	$9.00\ 10^{-13}$	$8.06\ 10^{-5}$
Sn-126	$8.96\ 10^{7}$	$3.29\ 10^{-6}$	$2.95\ 10^{2}$
I-129	$4.17\ 10^{4}$	$7.55\ 10^{-8}$	$3.15\ 10^{-3}$
Ba-133	$8.96\ 10^{7}$	$4.91\ 10^{-8}$	4.40
Cs-134	$8.96\ 10^{7}$	$2.19\ 10^{-14}$	$1.97\ 10^{-6}$
Cs-137	$8.96\ 10^{7}$	$1.11\ 10^{-6}$	$9.95\ 10^{1}$
Pm-147	$8.96\ 10^{7}$	$6.32\ 10^{-17}$	$5.67\ 10^{-9}$
Eu-152	$8.96\ 10^{7}$	$4.19\ 10^{-7}$	$3.75\ 10^{1}$
Eu-154	$8.96\ 10^{7}$	$7.86\ 10^{-8}$	7.05
Eu-155	$8.96\ 10^{7}$	$3.79\ 10^{-11}$	$3.40\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$2.00\ 10^{-7}$	$1.79\ 10^{1}$
Ra-226*	$8.96\ 10^{7}$	$1.93\ 10^{-5}$	$1.73\ 10^{3}$
Ra-228	$8.96\ 10^{7}$	$1.67\ 10^{-8}$	1.50
Ac-227	$8.96\ 10^{7}$	$2.95\ 10^{-6}$	$2.64\ 10^{2}$
Th-229	$8.96\ 10^{7}$	$1.01\ 10^{-5}$	$9.01\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$3.54\ 10^{-6}$	$2.46\ 10^{2}$
Th-232	$7.16\ 10^{7}$	$2.66\ 10^{-5}$	$1.90\ 10^{3}$
Pa-231	$1.86\ 10^{7}$	$2.17\ 10^{-5}$	$4.04\ 10^{2}$
U-232	$8.96\ 10^{7}$	$7.06\ 10^{-7}$	$6.32\ 10^{1}$
U-233	$3.13\ 10^{7}$	$3.75\ 10^{-7}$	$1.18\ 10^{1}$
U-234	$6.41\ 10^{6}$	$3.11\ 10^{-7}$	2.00
U-235	$4.92\ 10^{6}$	$1.27\ 10^{-6}$	6.23
U-236	$8.96\ 10^{7}$	$2.87\ 10^{-7}$	$2.57\ 10^{1}$
U-238	$2.53\ 10^{7}$	$4.39\ 10^{-7}$	$1.11\ 10^{1}$
Np-237	$4.52\ 10^{5}$	$2.99\ 10^{-6}$	1.35
Pu-238	$8.96\ 10^{7}$	$2.14\ 10^{-6}$	$1.92\ 10^{2}$
Pu-239	$8.96\ 10^{7}$	$3.74\ 10^{-6}$	$3.36\ 10^{2}$
Pu-240	$8.96\ 10^{7}$	$3.73\ 10^{-6}$	$3.34\ 10^{2}$
Pu-241	$8.96\ 10^{7}$	$9.38\ 10^{-8}$	8.41
Pu-242	$8.96\ 10^{7}$	$3.44\ 10^{-6}$	$3.09\ 10^{2}$
Am-241	$8.96\ 10^{7}$	$2.78\ 10^{-6}$	$2.49\ 10^{2}$

Radionuclide	Maximum inventory (MBq)	Dose to Borehole excavator (60y) ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Cm-243	$8.96\ 10^{7}$	$7.02\ 10^{-7}$	$6.29\ 10^{1}$
Cm-244	$8.96\ 10^{7}$	$1.88\ 10^{-7}$	$1.69\ 10^{1}$

* Assumes Ra-226 distributed with other LLW.

787. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.5.

E.5.3. Trial pit excavation

E.5.3.1. Assessment calculations for Trial Pit Excavator

788. 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 85. 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 85 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)

789. This scenario has also been used to consider both the consignment tonnage limit and the specific activity limits applied in the CFA. 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.

E.5.3.2. Dose to Trial Pit Excavator on site after 60 years

790. The largest dose rates per MBq disposal for the trial pit excavator under this scenario are Ra-226, Nb-94, Pa-231 and Th-232. Note that the specific doses calculated for this scenario are smaller than those calculated for the borehole excavation scenario (see Table 86). This is because the borehole excavation is of longer duration than the trial pit excavation and the borehole driller 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 scenario.

791. 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 84. The largest dose is from a consignment containing the 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, no additional restrictions on the activity concentration in a consignment are required. It is possible that a package within a consignment of 10 t may contain a higher activity concentration than the average for that consignment. The impact of variations in activity concentrations between packages is considered for this scenario in Section E.7.3.

Table 86 Dose to Trial pit excavator at the site

Radionuclide	Maximum inventory (MBq)	Dose to Trial pit excavator at 60y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)	Dose to Trial pit excavator – 10 t waste at 200 Bq g^{-1} ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$1.48\ 10^{-13}$	$1.32\ 10^{-5}$	$5.63\ 10^{-5}$
C-14	$8.96\ 10^{7}$	$1.47\ 10^{-10}$	$1.32\ 10^{-2}$	$5.61\ 10^{-2}$
Cl-36	$1.48\ 10^{6}$	$1.07\ 10^{-9}$	$1.59\ 10^{-3}$	$4.09\ 10^{-1}$
Fe-55	$8.96\ 10^{7}$	$1.29\ 10^{-17}$	$1.15\ 10^{-9}$	$4.91\ 10^{-9}$
Co-60	$8.96\ 10^{7}$	$1.96\ 10^{-9}$	$1.76\ 10^{-1}$	$7.47\ 10^{-1}$
Ni-63	$8.96\ 10^{7}$	$1.98\ 10^{-11}$	$1.77\ 10^{-3}$	$7.53\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$3.19\ 10^{-9}$	$2.86\ 10^{-1}$	1.22
Nb-94	$8.96\ 10^{7}$	$3.13\ 10^{-6}$	$2.80\ 10^{2}$	$1.19\ 10^{3}$
Tc-99	$8.96\ 10^{7}$	$2.66\ 10^{-10}$	$2.39\ 10^{-2}$	$1.02\ 10^{-1}$
Ru-106	$8.96\ 10^{7}$	$9.25\ 10^{-25}$	$8.29\ 10^{-17}$	$3.53\ 10^{-16}$
Ag-108m	$8.96\ 10^{7}$	$2.82\ 10^{-6}$	$2.53\ 10^{2}$	$1.08\ 10^{3}$
Sb-125	$8.96\ 10^{7}$	$2.25\ 10^{-13}$	$2.02\ 10^{-5}$	$8.58\ 10^{-5}$
Sn-126	$8.96\ 10^{7}$	$8.22\ 10^{-7}$	$7.37\ 10^{1}$	$3.14\ 10^{2}$
I-129	$4.17\ 10^{4}$	$1.89\ 10^{-8}$	$7.87\ 10^{-4}$	7.20
Ba-133	$8.96\ 10^{7}$	$1.23\ 10^{-8}$	1.10	4.68
Cs-134	$8.96\ 10^{7}$	$5.49\ 10^{-15}$	$4.92\ 10^{-7}$	$2.09\ 10^{-6}$
Cs-137	$8.96\ 10^{7}$	$2.78\ 10^{-7}$	$2.49\ 10^{1}$	$1.06\ 10^{2}$
Pm-147	$8.96\ 10^{7}$	$1.58\ 10^{-17}$	$1.42\ 10^{-9}$	$6.03\ 10^{-9}$
Eu-152	$8.96\ 10^{7}$	$1.05\ 10^{-7}$	9.38	$3.99\ 10^{1}$
Eu-154	$8.96\ 10^{7}$	$1.97\ 10^{-8}$	1.76	7.50
Eu-155	$8.96\ 10^{7}$	$9.48\ 10^{-12}$	$8.49\ 10^{-4}$	$3.61\ 10^{-3}$
Pb-210	$8.96\ 10^{7}$	$4.99\ 10^{-8}$	4.47	$1.90\ 10^{1}$
Ra-226*	$8.96\ 10^{7}$	$4.82\ 10^{-6}$	$4.32\ 10^{2}$	$1.50\ 10^{3}$
Ra-228	$8.96\ 10^{7}$	$4.18\ 10^{-9}$	$3.74\ 10^{-1}$	1.59
Ac-227	$8.96\ 10^{7}$	$7.37\ 10^{-7}$	$6.61\ 10^{1}$	$2.81\ 10^{2}$
Th-229	$8.96\ 10^{7}$	$2.51\ 10^{-6}$	$2.25\ 10^{2}$	$9.59\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$8.85\ 10^{-7}$	$6.14\ 10^{1}$	$3.38\ 10^{2}$
Th-232	$7.16\ 10^{7}$	$6.64\ 10^{-6}$	$4.76\ 10^{2}$	$2.53\ 10^{3}$
Pa-231	$1.86\ 10^{7}$	$5.44\ 10^{-6}$	$1.01\ 10^{2}$	$2.07\ 10^{3}$
U-232	$8.96\ 10^{7}$	$1.76\ 10^{-7}$	$1.58\ 10^{1}$	$6.73\ 10^{1}$
U-233	$3.13\ 10^{7}$	$9.38\ 10^{-8}$	2.94	$3.58\ 10^{1}$
U-234	$6.41\ 10^{6}$	$7.78\ 10^{-8}$	$4.99\ 10^{-1}$	$2.97\ 10^{1}$
U-235	$4.92\ 10^{6}$	$3.17\ 10^{-7}$	1.56	$1.21\ 10^{2}$
U-236	$8.96\ 10^{7}$	$7.18\ 10^{-8}$	6.44	$2.74\ 10^{1}$

Radionuclide	Maximum inventory (MBq)	Dose to Trial pit excavator at 60y (µSv y^{-1} MBq^{-1})	Dose from maximum inventory (µSv y^{-1})	Dose to Trial pit excavator – 10 t waste at 200 Bq g^{-1} (µSv y^{-1})
U-238	2.53 10^{7}	1.10 10^{-7}	2.78	4.19 10^{1}
Np-237	4.52 10^{5}	7.47 10^{-7}	3.37 10^{-1}	2.85 10^{2}
Pu-238	8.96 10^{7}	5.35 10^{-7}	4.79 10^{1}	2.04 10^{2}
Pu-239	8.96 10^{7}	9.36 10^{-7}	8.39 10^{1}	3.57 10^{2}
Pu-240	8.96 10^{7}	9.32 10^{-7}	8.35 10^{1}	3.55 10^{2}
Pu-241	8.96 10^{7}	2.35 10^{-8}	2.10	8.95
Pu-242	8.96 10^{7}	8.61 10^{-7}	7.71 10^{1}	3.28 10^{2}
Am-241	8.96 10^{7}	6.94 10^{-7}	6.22 10^{1}	2.65 10^{2}
Cm-243	8.96 10^{7}	1.75 10^{-7}	1.57 10^{1}	6.69 10^{1}
Cm-244	8.96 10^{7}	4.71 10^{-8}	4.22	1.80 10^{1}

* Assumes Ra-226 distributed with other LLW.

E.5.4. Laboratory Analyst

792. The laboratory analyst is assumed to be exposed to radioactive material when analysing samples recovered from a borehole or trial pit. The exposure pathways considered are inhalation of dust particles, inadvertent ingestion of contaminated material, external irradiation and exposure to radon generated from the sample. The following methodology applies to a single sample.

E.5.4.1. Estimating activity concentration of contaminated dust for exposure calculations

793. The activity concentration of radionuclides on dust in the air, $C_{air,sample}$ (Bq m^{-3}), is given by (Hicks & Baldwin, 2011):

$$C_{air,sample} = \frac{C_w \rho_{dust}}{\rho_w}$$

where:

- C_w is the activity concentration of the radionuclide in the waste volume (Bq m^{-3});
- ρ_w is the waste density (kg m^{-3}); and,
- ρ_{dust} is the dust density (kg m^{-3}).

E.5.4.2. Assessment calculations for Laboratory Analyst

Inhalation of contaminated dust

794. The effective dose from inhalation of contaminated dust is then given by:

$$Dose_{inh} = I_{inh} C_{air,sample} t_{inh} f_{inh} D_{inh}$$

where:

- I_{inh} is the inhalation rate (m^3 h^{-1});
- t_{inh} is the exposure time (h y^{-1});
- f_{inh} is the inhalable fraction of dust; and,
- D_{inh} is the inhalation dose coefficient (Sv Bq^{-1}) (see Table 170).

795. The parameter values used in this work are given in Table 87.

Table 87 Inhalation parameters

Parameter	Units	Value	Comments
ρ_{dust}	kg m^{-3}	6.00 10^{-7}	
I_{inh}	m^3 h^{-1}	1.2	Corresponds to light work
t_{inh}	h y^{-1}	2	
f_{inh}		1	Assume all dust inhalable

Values taken from (Hicks & Baldwin, 2011).

Ingestion of contaminated material

796. The dose due to inadvertent ingestion of radioactive material is given by:

$$Dose_{ing} = \frac{I_{ing} C_w t_{ing} D_{ing}}{\rho_w}$$

where:

- I_{ing} is the ingestion rate of contaminated material (kg d^{-1});
- t_{ing} is the exposure time (d y^{-1}); and,
- D_{ing} is the ingestion dose coefficient (Sv Bq^{-1}) (see Table 170).

797. The parameter values used in this work are given in Table 88.

Table 88 Ingestion parameters

Parameter	Units	Value	Comments
I_{ing}	kg d^{-1}	1.00 10^{-5}	
t_{ing}	d y^{-1}	1	Assumes one inspection a day

Values taken from (Hicks & Baldwin, 2011).

External irradiation

798. The dose for irradiation exposure from a semi-infinite slab of radioactively contaminated material may be scaled to yield an expression for external irradiation resulting from inspection of a sample by introducing a suitable scaling factor to account for the reduction in dose associated with the finite geometry of the sample (Hicks & Baldwin, 2011). The dose is given by:

$$Dose_{ext} = \frac{C_w t_{ext} f_{ext} D_{ext}}{\rho_w}$$

where:

- t_{ext} is the exposure time (h y^{-1});
- D_{ext} is the dose coefficient from external irradiation (Sv h^{-1} Bq^{-1} kg) for an individual 1 m from a semi-infinite slab of radioactive material (see Table 170); and,
- f_{ext} is the reduction factor that accounts for the restricted area of the source of radiation relative to the semi-infinite slab geometry and any shielding.

799. The parameter values used in this work are given in Table 89. The sample is assumed to approximate to a sphere and Thorne (2010) shows that the dose coefficient at a distance of 0.3 m from a sphere of radius 2 m approximates to a semi-infinite slab dose coefficient. Scaling factors (f_{ext}) accounting for distance from and radius of the sphere are calculated for a distance of 0.3 to 3 m from the source of radius 0.1 to 2 m (Thorne, 2010). Hicks and Baldwin (2011) assume that the waste examined by the laboratory analyst has 0.2 m radius and is considered from a distance of 0.3 m; Table 3 from Thorne (2010) indicates a scaling factor of 0.0833 should be used, rounded to 0.08 by Hicks and Baldwin.

Table 89 External irradiation parameters

Parameter	Units	Value	Comments
t_{ext}	h y^{-1}	2	Assumes one inspection a day, 2 hours per inspection
f_{ext}		0.08	Scales from slab to spherical configuration

Values taken from (Hicks & Baldwin, 2011)

Radon exposure

800. The radon activity concentration is given by:

$$C_{Rn} = \frac{\lambda_{Rn} V_{samp} C_{w,Rn} S_{Rn}}{V_b (Q_b + \lambda_{Rn})}$$

where:

- V_{samp} is the volume of the sample (m^3);
- $C_{w,Rn}$ is the source concentration (of Ra-226) in the sample (Bq m^{-3});
- S_{Rn} is the emanation coefficient (Rn-222) from the sample;
- C_{Rn} is the equilibrium activity concentration of radon in the lab (Bq m^{-3});
- Q_b is the ventilation rate of the building (y^{-1});
- V_b is the volume of the laboratory in which the analysis is taking place (m^3); and,

- λ_{Rn} is the radon decay constant (y^{-1}).

801. The resulting dose from radon (and its progeny) is given by:

$$Dose_{Rn} = C_{Rn} t_{Rn} D_{Rn}$$

where:

- t_{Rn} is the exposure time ($h\ y^{-1}$); and,
- D_{Rn} is a dose conversion factor (for radon) ($Sv\ h^{-1}\ Bq^{-1}\ m^3$) which takes daughter radionuclides in secular equilibrium into account.

802. The parameter values used in this work are given in Table 90.

Table 90 Parameters for radon inhalation by laboratory analyst

Parameter	Units	Value	Comments
V_{samp}	m^3	0.03	
S_{radon}		0.2	
Q_b	y^{-1}	44000	Cautious estimate for laboratory with mechanical ventilation
V_b	m^3	100	
t_{Rn}	$h\ y^{-1}$	2	
λ_{Rn}	y^{-1}	66.28	
D_{Radon}	$Sv\ h^{-1}\ Bq^{-1}\ m^3$	$3.60\ 10^{-9}$	Takes account of progeny in secular equilibrium

Values taken from (Hicks & Baldwin, 2011)

803. The LLWR human intrusion assessment (Hicks & Baldwin, 2011) suggests that a reasonable assumption is the analysis of 25 samples and this has been used in the ESC.

E.5.4.3. Dose to Laboratory Analyst on site after 60 years

804. The dose to a Laboratory Analyst processing 25 samples in a year is presented in Table 91. The maximum inventory for each radionuclide and the dose from disposal of that maximum inventory are also shown.The largest dose rates per MBq disposal in this scenario are Th-229, Pa-231 and Th-232 (see Table 91). These radionuclides will correspondingly have the smallest radiological capacities under this scenario. Radiological capacity calculations are presented in Section 7.4.

Table 91 Dose to Laboratory Analyst processing 25 samples

Radionuclide	Maximum inventory (MBq)	Dose to Laboratory analyst at 60y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$2.49\ 10^{-13}$	$2.23\ 10^{-5}$
C-14	$8.96\ 10^{7}$	$1.85\ 10^{-10}$	$1.66\ 10^{-2}$
Cl-36	$1.48\ 10^{6}$	$4.14\ 10^{-10}$	$6.12\ 10^{-4}$

Radionuclide	Maximum inventory (MBq)	Dose to Laboratory analyst at 60y (μSv y^{-1} MBq^{-1})	Dose from maximum inventory (μSv y^{-1})
Fe-55	$8.96\ 10^{7}$	$1.45\ 10^{-17}$	$1.30\ 10^{-9}$
Co-60	$8.96\ 10^{7}$	$3.93\ 10^{-10}$	$3.52\ 10^{-2}$
Ni-63	$8.96\ 10^{7}$	$2.92\ 10^{-11}$	$2.61\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$2.05\ 10^{-9}$	$1.83\ 10^{-1}$
Nb-94	$8.96\ 10^{7}$	$6.27\ 10^{-7}$	$5.62\ 10^{1}$
Tc-99	$8.96\ 10^{7}$	$3.37\ 10^{-10}$	$3.02\ 10^{-2}$
Ru-106	$8.96\ 10^{7}$	$1.89\ 10^{-25}$	$1.70\ 10^{-17}$
Ag-108m	$8.96\ 10^{7}$	$5.66\ 10^{-7}$	$5.07\ 10^{1}$
Sb-125	$8.96\ 10^{7}$	$4.52\ 10^{-14}$	$4.05\ 10^{-6}$
Sn-126	$8.96\ 10^{7}$	$1.66\ 10^{-7}$	$1.48\ 10^{1}$
I-129	$4.17\ 10^{4}$	$1.59\ 10^{-8}$	$6.64\ 10^{-4}$
Ba-133	$8.96\ 10^{7}$	$2.47\ 10^{-9}$	$2.21\ 10^{-1}$
Cs-134	$8.96\ 10^{7}$	$1.10\ 10^{-15}$	$9.88\ 10^{-8}$
Cs-137	$8.96\ 10^{7}$	$5.61\ 10^{-8}$	5.03
Pm-147	$8.96\ 10^{7}$	$1.71\ 10^{-17}$	$1.53\ 10^{-9}$
Eu-152	$8.96\ 10^{7}$	$2.10\ 10^{-8}$	1.88
Eu-154	$8.96\ 10^{7}$	$3.95\ 10^{-9}$	$3.54\ 10^{-1}$
Eu-155	$8.96\ 10^{7}$	$1.92\ 10^{-12}$	$1.72\ 10^{-4}$
Pb-210	$8.96\ 10^{7}$	$6.71\ 10^{-8}$	6.01
Ra-226*	$8.96\ 10^{7}$	$1.66\ 10^{-6}$	$1.48\ 10^{2}$
Ra-228	$8.96\ 10^{7}$	$1.65\ 10^{-9}$	$1.48\ 10^{-1}$
Ac-227	$8.96\ 10^{7}$	$1.63\ 10^{-6}$	$1.46\ 10^{2}$
Th-229	$8.96\ 10^{7}$	$4.99\ 10^{-6}$	$4.47\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$1.96\ 10^{-6}$	$1.36\ 10^{2}$
Th-232	$7.16\ 10^{7}$	$4.39\ 10^{-6}$	$3.15\ 10^{2}$
Pa-231	$1.86\ 10^{7}$	$1.21\ 10^{-5}$	$2.25\ 10^{2}$
U-232	$8.96\ 10^{7}$	$4.06\ 10^{-7}$	$3.64\ 10^{1}$
U-233	$3.13\ 10^{7}$	$2.16\ 10^{-7}$	6.78
U-234	$6.41\ 10^{6}$	$1.85\ 10^{-7}$	1.19
U-235	$4.92\ 10^{6}$	$2.19\ 10^{-7}$	1.08
U-236	$8.96\ 10^{7}$	$1.70\ 10^{-7}$	$1.53\ 10^{1}$
U-238	$2.53\ 10^{7}$	$1.66\ 10^{-7}$	4.21
Np-237	$4.52\ 10^{5}$	$1.03\ 10^{-6}$	$4.65\ 10^{-1}$
Pu-238	$8.96\ 10^{7}$	$1.31\ 10^{-6}$	$1.18\ 10^{2}$
Pu-239	$8.96\ 10^{7}$	$2.29\ 10^{-6}$	$2.06\ 10^{2}$
Pu-240	$8.96\ 10^{7}$	$2.28\ 10^{-6}$	$2.05\ 10^{2}$
Pu-241	$8.96\ 10^{7}$	$5.66\ 10^{-8}$	5.07
Pu-242	$8.96\ 10^{7}$	$2.11\ 10^{-6}$	$1.89\ 10^{2}$
Am-241	$8.96\ 10^{7}$	$1.67\ 10^{-6}$	$1.50\ 10^{2}$
Cm-243	$8.96\ 10^{7}$	$3.28\ 10^{-7}$	$2.94\ 10^{1}$
Cm-244	$8.96\ 10^{7}$	$1.15\ 10^{-7}$	$1.03\ 10^{1}$

Radionuclide	Maximum inventory (MBq)	Dose to Laboratory analyst at 60y (μSv y^{-1} MBq^{-1})	Dose from maximum inventory (μSv y^{-1})

* Assumes Ra-226 distributed with other LLW.

805. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.5.

E.5.5. Excavation for housing or road – excavator

E.5.5.1. Assessment calculations for Housing or Road Excavator

806. The exposure pathways for the 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 92. 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 92 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.5.5.2. Dose to Excavator for Housing or Road on site after 150 years

807. 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 93). 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. The maximum inventory for each radionuclide and the dose from disposal of this maximum inventory is also shown in the table.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 93) 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 concentrationn, 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 the ENRMF on these intrusion doses and on radon release is discussed in the next section (see paragraph 861).

Table 93 Dose to Housing site/road excavator at the site

Radionuclide	Maximum inventory (MBq)	Dose to Housing site/road excavator at 150y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
H-3	$8.96\ 10^{7}$	$3.73\ 10^{-15}$	$3.34\ 10^{-7}$
C-14	$8.96\ 10^{7}$	$5.81\ 10^{-10}$	$5.20\ 10^{-2}$
Cl-36	$1.48\ 10^{6}$	$4.28\ 10^{-9}$	$6.33\ 10^{-3}$
Fe-55	$8.96\ 10^{7}$	$6.49\ 10^{-27}$	$5.81\ 10^{-19}$
Co-60	$8.96\ 10^{7}$	$5.67\ 10^{-14}$	$5.08\ 10^{-6}$
Ni-63	$8.96\ 10^{7}$	$4.23\ 10^{-11}$	$3.79\ 10^{-3}$
Sr-90	$8.96\ 10^{7}$	$1.46\ 10^{-9}$	$1.31\ 10^{-1}$
Nb-94	$8.96\ 10^{7}$	$1.24\ 10^{-5}$	$1.11\ 10^{3}$
Tc-99	$8.96\ 10^{7}$	$1.06\ 10^{-9}$	$9.53\ 10^{-2}$
Ru-106	$8.96\ 10^{7}$	$1.21\ 10^{-50}$	$1.09\ 10^{-42}$
Ag-108m	$8.96\ 10^{7}$	$9.71\ 10^{-6}$	$8.70\ 10^{2}$
Sb-125	$8.96\ 10^{7}$	$1.36\ 10^{-22}$	$1.22\ 10^{-14}$
Sn-126	$8.96\ 10^{7}$	$3.28\ 10^{-6}$	$2.94\ 10^{2}$
I-129	$4.17\ 10^{4}$	$7.53\ 10^{-8}$	$3.14\ 10^{-3}$
Ba-133	$8.96\ 10^{7}$	$1.30\ 10^{-10}$	$1.17\ 10^{-2}$
Cs-134	$8.96\ 10^{7}$	$1.66\ 10^{-27}$	$1.49\ 10^{-19}$
Cs-137	$8.96\ 10^{7}$	$1.40\ 10^{-7}$	$1.26\ 10^{1}$
Pm-147	$8.96\ 10^{7}$	$2.96\ 10^{-27}$	$2.65\ 10^{-19}$
Eu-152	$8.96\ 10^{7}$	$4.17\ 10^{-9}$	$3.74\ 10^{-1}$
Eu-154	$8.96\ 10^{7}$	$5.52\ 10^{-11}$	$4.95\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$7.72\ 10^{-17}$	$6.91\ 10^{-9}$
Pb-210	$8.96\ 10^{7}$	$1.20\ 10^{-8}$	1.07
Ra-226*	$8.96\ 10^{7}$	$1.85\ 10^{-5}$	$1.66\ 10^{3}$
Ra-228	$8.96\ 10^{7}$	$3.24\ 10^{-13}$	$2.90\ 10^{-5}$
Ac-227	$8.96\ 10^{7}$	$1.68\ 10^{-7}$	$1.50\ 10^{1}$
Th-229	$8.96\ 10^{7}$	$9.95\ 10^{-6}$	$8.92\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$4.13\ 10^{-6}$	$2.87\ 10^{2}$
Th-232	$7.16\ 10^{7}$	$2.65\ 10^{-5}$	$1.90\ 10^{3}$
Pa-231	$1.86\ 10^{7}$	$2.44\ 10^{-5}$	$4.52\ 10^{2}$
U-232	$8.96\ 10^{7}$	$2.85\ 10^{-7}$	$2.55\ 10^{1}$
U-233	$3.13\ 10^{7}$	$4.59\ 10^{-7}$	$1.44\ 10^{1}$
U-234	$6.41\ 10^{6}$	$3.13\ 10^{-7}$	2.01
U-235	$4.92\ 10^{6}$	$1.27\ 10^{-6}$	6.26
U-236	$8.96\ 10^{7}$	$2.87\ 10^{-7}$	$2.57\ 10^{1}$
U-238	$2.53\ 10^{7}$	$4.38\ 10^{-7}$	$1.11\ 10^{1}$
Np-237	$4.52\ 10^{5}$	$2.98\ 10^{-6}$	1.35
Pu-238	$8.96\ 10^{7}$	$1.05\ 10^{-6}$	$9.40\ 10^{1}$
Pu-239	$8.96\ 10^{7}$	$3.73\ 10^{-6}$	$3.34\ 10^{2}$
Pu-240	$8.96\ 10^{7}$	$3.68\ 10^{-6}$	$3.30\ 10^{2}$
Pu-241	$8.96\ 10^{7}$	$8.27\ 10^{-8}$	7.41

Radionuclide	Maximum inventory (MBq)	Dose to Housing site/road excavator at 150y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Pu-242	$8.96\ 10^{7}$	$3.44\ 10^{-6}$	$3.08\ 10^{2}$
Am-241	$8.96\ 10^{7}$	$2.40\ 10^{-6}$	$2.15\ 10^{2}$
Cm-243	$8.96\ 10^{7}$	$8.61\ 10^{-8}$	7.71
Cm-244	$8.96\ 10^{7}$	$1.59\ 10^{-8}$	1.42

* Assumes Ra-226 distributed with other LLW.

E.5.6. Site Resident – no cap damage

E.5.6.1. Assessment calculations for Site Residents (no cap damage)

808. 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.5.8. External irradiation from the buried wastes and inhalation of radioactive gases released through the cap are considered. Habit data are presented in Table 94.

Table 94 Habit data for site resident

Parameter	Units	Value	Description
B	$m^3\ h^{-1}$	1	Inhalation rate
O_{in}	$h\ y^{-1}$	7012.8	Indoor occupancy (80% indoors)

809. 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

810. The gas pathway is considered in the same way for tritium and C-14. The release rate of radioactive gas is given in paragraph 480 using the release fractions and initial activity values in Table 37.

811. 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.

812. The effective doses arising from inhalation of generated gases are calculated for site residents post-closure (at t = 150 years), assuming that 80% of a resident's time is spent indoors. The dose is calculated according to:

$$Dose_{gas,indoors} = D_{inh}^{Rn} \cdot B \cdot O_{in} \left[R_{Rn,gas}(t) \cdot \frac{a_H}{a} \cdot \left(\frac{1}{kV} \right) \right]$$

where:

- D_{inh}^{Rn} is the inhalation dose coefficient of radionuclide *Rn* (Sv Bq^{-1});
- B is the inhalation rate (m^3 h^{-1});
- O_{in} is the occupancy indoors (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).

813. The gas dispersion parameters used in this work are summarised in Table 95, the dimensions of the landfill are given in Table 33, the dose coefficients in Table 170 and habit data in Table 94.

Table 95 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

814. This section considers migration of radon gas from a waste cell into a building constructed on the intact cap.

815. 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 489. The parameters in Table 40 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.

816. 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, 143,447 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).

817. The values of the quantities used in this work are given in Table 96, except for the landfill area (see Table 33).

Table 96 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)

818. The resultant inhalation dose (Sv y^{-1}) to a resident of the house is then given by:

$$Dose = D_{inh} \cdot C_{Rn-222,house} \cdot B_{inh} \cdot O_{indoor}$$

where:

- D_{inh} is the inhalation dose coefficient (Sv Bq^{-1});
- B_{inh} is the breathing rate (m^3 h^{-1}); and,
- O_{indoor} is the indoor occupancy (h y^{-1}).

819. The dose coefficient is presented in Table 41 and habit data in Table 94.

External irradiation

820. The dose to a future site resident from external irradiation is also calculated through the intact cap assuming that 80% of a resident's time is spent indoors. The dose is calculated according to:

$$Dose_{irr} = D_{irr,slab}^{Rn} \cdot (O_{out} + O_{in} sf) \cdot \left(\frac{A_{Rn,waste}(t)}{V_{waste}\rho_{waste}}\right) \cdot e^{-\mu^{Rn} \cdot x}$$

where:

- $D_{irr,slab}^{Rn}$ is the dose conversion factor for irradiation from radionuclide *Rn* (see Table 170), 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} is the outdoor occupancy;
- O_{in} is the indoor occupancy;
- 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 170); and,
- x is the thickness of the cap and cover material (m).

821. The values of these parameters employed in this work are summarised in Table 97 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 97 External irradiation parameters

Parameter	Units	Value	Description
O_{out}		0.2	Outdoor occupancy
O_{in}		0.8	Indoor occupancy
sf		0.1	Shielding factor
V_{waste}	m^3	1246215	Waste volume
ρ_{waste}	kg m^{-3}	1530	Waste density
x	m	2.6	Cap plus cover thickness

E.5.6.2. Dose to Site Resident – no cap damage

822. In Table 98 radon fluxes, indoor Rn-222 activity concentrations and inhalation doses at 150 years are given arising from a nominal 1 MBq of Ra-226. These calculations assume that the released radon is in secular equilibrium with the parent radium. They take into account decay of Rn-222 within the waste before release. This Table also shows the sensitivity of radon emissions to the depth of radium placement beneath the surface of the landfill.

823. 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 98 Radon inhalation doses for a dwelling built on a capped landfill – unit inventory

Depth below cap	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})
5	6.6	$9.56\ 10^{-10}$	$1.03\ 10^{-18}$	$4.31\ 10^{-17}$
4	5.6	$1.28\ 10^{-7}$	$1.37\ 10^{-16}$	$5.77\ 10^{-15}$
3.4	5	$2.43\ 10^{-6}$	$2.60\ 10^{-15}$	$1.09\ 10^{-13}$
3	4.6	$1.73\ 10^{-5}$	$1.85\ 10^{-14}$	$7.80\ 10^{-13}$
2	3.6	$2.35\ 10^{-3}$	$2.53\ 10^{-12}$	$1.06\ 10^{-10}$
1	2.6*	$3.23\ 10^{-1}$	$3.46\ 10^{-10}$	$1.46\ 10^{-8}$
0	1.6	$4.45\ 10^{1}$	$4.78\ 10^{-8}$	$2.01\ 10^{-6}$

* Minimum depth since LLW is disposed of at a minimum depth of 1 m below the top of the cell and the cap is 1.6 m thick.

824. 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 99. Note that these results include the effects of ingrowth after 150 years upon the calculated doses.

825. The expected dose if each radionuclide is disposed at the maximum inventory is shown in the right hand column. The highest dose is from C-14 gas (384 µSv y^{-1}), but the highest dose from waste disposed of at the ENRMF will always be lower than this due to application of the sum of fractions approach. All other doses are below 1 µSv y^{-1}. Ra-226 results are given for two different activity concentrations, reflecting the emplacement strategy.

Table 99 Site resident exposure

Radionuclide	Maximum inventory (MBq)	Dose (µSv y^{-1} MBq^{-1})			Dose (µSv y^{-1}) from maximum inventory
		Gas	External	Total	
H-3	$8.96\ 10^{7}$	$3.30\ 10^{-10}$	0	$3.30\ 10^{-10}$	$2.95\ 10^{-2}$
C-14	$8.96\ 10^{7}$	$4.28\ 10^{-6}$	$3.97\ 10^{-73}$	$4.28\ 10^{-6}$	$3.84\ 10^{2}$
Cl-36	$1.48\ 10^{6}$		$8.71\ 10^{-31}$	$8.71\ 10^{-31}$	$1.29\ 10^{-24}$
Fe-55	$8.96\ 10^{7}$		0	0	0
Co-60	$8.96\ 10^{7}$		$4.64\ 10^{-26}$	$4.64\ 10^{-26}$	$4.16\ 10^{-18}$
Ni-63	$8.96\ 10^{7}$		0	0	0
Sr-90	$8.96\ 10^{7}$		$3.55\ 10^{-29}$	$3.55\ 10^{-29}$	$3.18\ 10^{-21}$
Nb-94	$8.96\ 10^{7}$		$9.09\ 10^{-20}$	$9.09\ 10^{-20}$	$8.14\ 10^{-12}$
Tc-99	$8.96\ 10^{7}$		$1.65\ 10^{-52}$	$1.65\ 10^{-52}$	$1.48\ 10^{-44}$
Ru-106	$8.96\ 10^{7}$		$8.99\ 10^{-66}$	$8.99\ 10^{-66}$	$8.05\ 10^{-58}$
Ag-108m	$8.96\ 10^{7}$		$4.19\ 10^{-21}$	$4.19\ 10^{-21}$	$3.76\ 10^{-13}$
Sb-125	$8.96\ 10^{7}$		$1.70\ 10^{-38}$	$1.70\ 10^{-38}$	$1.52\ 10^{-30}$
Sn-126	$8.96\ 10^{7}$		$3.58\ 10^{-21}$	$3.58\ 10^{-21}$	$3.20\ 10^{-13}$
I-129	$4.17\ 10^{4}$		$9.84\ 10^{-155}$	$9.84\ 10^{-155}$	$4.10\ 10^{-150}$

Radionuclide	Maximum inventory (MBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			Dose ($\mu Sv\ y^{-1}$) from maximum inventory
		Gas	External	Total	
Ba-133	$8.96\ 10^{7}$		$2.28\ 10^{-29}$	$2.28\ 10^{-29}$	$2.04\ 10^{-21}$
Cs-134	$8.96\ 10^{7}$		$4.40\ 10^{-42}$	$4.40\ 10^{-42}$	$3.94\ 10^{-34}$
Cs-137	$8.96\ 10^{7}$		$1.93\ 10^{-22}$	$1.93\ 10^{-22}$	$1.73\ 10^{-14}$
Pm-147	$8.96\ 10^{7}$		$9.88\ 10^{-69}$	$9.88\ 10^{-69}$	$8.85\ 10^{-61}$
Eu-152	$8.96\ 10^{7}$		$2.40\ 10^{-22}$	$2.40\ 10^{-22}$	$2.15\ 10^{-14}$
Eu-154	$8.96\ 10^{7}$		$4.15\ 10^{-24}$	$4.15\ 10^{-24}$	$3.72\ 10^{-16}$
Eu-155	$8.96\ 10^{7}$		$2.22\ 10^{-53}$	$2.22\ 10^{-53}$	$1.99\ 10^{-45}$
Pb-210	$8.96\ 10^{7}$		$2.08\ 10^{-27}$	$2.08\ 10^{-27}$	$1.86\ 10^{-19}$
Ra-226	$8.96\ 10^{7}$	$1.09\ 10^{-13}$	$4.83\ 10^{-35}$	$1.09\ 10^{-13}$	$9.81\ 10^{-6}$
Ra-228	$8.96\ 10^{7}$		$2.06\ 10^{-23}$	$2.06\ 10^{-23}$	$1.84\ 10^{-15}$
Ac-227	$8.96\ 10^{7}$		$4.07\ 10^{-26}$	$4.07\ 10^{-26}$	$3.65\ 10^{-18}$
Th-229	$8.96\ 10^{7}$		$2.22\ 10^{-21}$	$2.22\ 10^{-21}$	$1.99\ 10^{-13}$
Th-230	$6.93\ 10^{7}$		$3.24\ 10^{-36}$	$3.24\ 10^{-36}$	$2.25\ 10^{-28}$
Th-232	$7.16\ 10^{7}$		$2.74\ 10^{-18}$	$2.74\ 10^{-18}$	$1.97\ 10^{-10}$
Pa-231	$1.86\ 10^{7}$		$4.75\ 10^{-24}$	$4.75\ 10^{-24}$	$8.83\ 10^{-17}$
U-232	$8.96\ 10^{7}$		$7.33\ 10^{-42}$	$7.33\ 10^{-42}$	$6.57\ 10^{-34}$
U-233	$3.13\ 10^{7}$		$3.17\ 10^{-23}$	$3.17\ 10^{-23}$	$9.93\ 10^{-16}$
U-234	$6.41\ 10^{6}$		$5.53\ 10^{-44}$	$5.53\ 10^{-44}$	$3.54\ 10^{-37}$
U-235	$4.92\ 10^{6}$		$6.46\ 10^{-30}$	$6.46\ 10^{-30}$	$3.18\ 10^{-23}$
U-236	$8.96\ 10^{7}$		$2.04\ 10^{-26}$	$2.04\ 10^{-26}$	$1.83\ 10^{-18}$
U-238	$2.53\ 10^{7}$		$3.97\ 10^{-24}$	$3.97\ 10^{-24}$	$1.01\ 10^{-16}$
Np-237	$4.52\ 10^{5}$		$4.70\ 10^{-37}$	$4.70\ 10^{-37}$	$2.12\ 10^{-31}$
Pu-238	$8.96\ 10^{7}$		$1.47\ 10^{-51}$	$1.47\ 10^{-51}$	$1.31\ 10^{-43}$
Pu-239	$8.96\ 10^{7}$		$2.79\ 10^{-33}$	$2.79\ 10^{-33}$	$2.50\ 10^{-25}$
Pu-240	$8.96\ 10^{7}$		$7.18\ 10^{-50}$	$7.18\ 10^{-50}$	$6.44\ 10^{-42}$
Pu-241	$8.96\ 10^{7}$		$6.64\ 10^{-47}$	$6.64\ 10^{-47}$	$5.95\ 10^{-39}$
Pu-242	$8.96\ 10^{7}$		$9.22\ 10^{-32}$	$9.22\ 10^{-32}$	$8.26\ 10^{-24}$
Am-241	$8.96\ 10^{7}$		$4.19\ 10^{-45}$	$4.19\ 10^{-45}$	$3.75\ 10^{-37}$
Cm-243	$8.96\ 10^{7}$		$8.23\ 10^{-33}$	$8.23\ 10^{-33}$	$7.37\ 10^{-25}$
Cm-244	$8.96\ 10^{7}$		$1.05\ 10^{-61}$	$1.05\ 10^{-61}$	$9.42\ 10^{-54}$

E.5.7. Excavation for housing – Residential Occupant

826. 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 1.6 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).

827. 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.

828. Contaminated material may be left at the surface, although it is more likely that such materials would be disposed of given the hazardous nature of material in the landfill. The non-radioactive waste disposed of at the ENRMF largely comprises treated residues (grey coloured) and asbestos. This material is not biodegradable and will essentially remain the same over geological timescales.

829. 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 let alone produce quantities of edible crops which could sustain a smallholding or farm is inconceivable without significant dilution of the waste by clean soil.

830. Exposure pathways for occupancy of a smallholding on contaminated material include those used for the housing development case, but includes additional exposure pathways that are associated with the consumption of contaminated foodstuffs that require a larger area for both cultivation and animal husbandry. Occupancy of a smallholding (see Section E.5.9) is included in the human intrusion assessment; this is more cautious than a larger farm because it assumes more crops are grown on a relatively small area.

E.5.7.1. Estimating activity concentration in waste for exposure calculations

Dilution factors

831. 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.

832. 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.

833. 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 1.6 m capping materials, 1.0 m cover and 2.4 m waste. The mixed spoil therefore comprises 48% waste. Radioactive waste input to the landfill is on average limited to approximately 20% of total inputs to the ENRMF, the rest comprising other hazardous wastes and emplacement cover material.

834. 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 excavated spoil is assumed to contain 9.6% 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.

835. 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 value of 90% clean soil was suggested by the EA (applied to waste) (Environment Agency, 2011a) and has been used for the LLWR assessment (applied to spoil) [(Hicks & Baldwin, 2011); (Thorne, 2009)]. The basis for the 10% value for the fraction of contaminated soil in soil used for crops is not clear but given the nature of the hazardous wastes disposed at the site it is likely that this value would be lower. It is also unclear how spreading spoil to a few centimetres depth over a substantial area and then ploughing would be achieved in practice. It seems more likely that spoil would be used as in-fill, giving a deeper cross-section of waste, and then covering with clean soil to support crop growth. This would reduce both mixing with clean soil and the contaminated area [*FAREA* in (Oatway & Mobbs, 2003)].

836. Other dilution factors have been suggested:

- The SNIFFER default (SNIFFER, 2006) uses the IAEA TecDoc 1380 (IAEA, 2003) value of 0.3. This value is based on excavating a trench to a depth of 3 m from the surface, with waste mixing with a cap of 1 m and cover material of 1 m depth.

- A value of 0.5 is applied in NRPB W36 (Oatway & Mobbs, 2003), based on 15 cm clean soil mixed with underlying waste in their contaminated land assessment, and a factor of 0.67 is applied in the HPA landfill assessment (HPA, 2007) to the dose factors from (Oatway & Mobbs, 2003) to account for 1 m cap materials in a 3 m excavation - this dilution factor (0.34) is not appropriate for plant growth on hazardous waste materials.
- LLWR (Hicks & Baldwin, 2011) applies a factor of 0.04 for a smallholding, based on cap dilution (0.4) and mixing with clean soil (0.1).

837. This assessment considers two potentially exposed groups with similar assumptions:

- A smallholder (200 years after closure) who requires 1 to 3 hectares of land to produce meat, milk and a mixture of crops. The smallholder lives over the site and 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 (1.6 m capping layer, 1.0 m cover, 2.4 m waste) during excavation, site average radioactive waste content and mixing with clean soil (0.1), an overall dilution factor of 0.0096 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 = 0.096).
- A housing development (150 years after closure) with residents growing their own vegetables. The development excavates 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 0.0096 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 = 0.096).

838. 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.

839. 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}.

840. 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 assumes 3 adults live on the site. 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.

841. The long-term occupant is an adult living at a residential site built on top of the ENRMF 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

842. 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:

$$C_{Rn,soil,excavate} = \frac{A_{Rn}(t) \, . \, Dil}{V_{landfill} \cdot \rho_{landfill}}$$

843. 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 0.0096 is used for LLW in the garden as discussed above (see paragraph 837) and a factor of 0.096 for exposure inside the house.

E.5.7.2. Assessment calculations for Residential Occupant

844. 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

845. Dose from ingesting crops grown on contaminated soil is given by (Augean, 2009a):

$$Dose_{ing,crops} = \sum_{crop} \{Q_{crop} \cdot [C_{Rn,soil}(t) \cdot UF_{Rn,crop}]\} \cdot D_{Rn,ing}$$

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}).

846. Parameter values are summarised in Table 100, dose coefficients for ingestion are given in Table 170 and soil to crop transfer factors are given in Table 172.

Table 100 Parameters used in the long-term occupant scenario

Parameter	Substance	Units	Value
Consumption rate (adult)	Green vegetables*	kg y^{-1}	17.5
	Root vegetables*	kg y^{-1}	30
	Soil	kg y^{-1}	0.03
Occupancy Indoors		y y^{-1}	0.80
Occupancy outdoors*		y y^{-1}	0.20
Shielding factor indoors*			0.1
Occupancy dust		h y^{-1}	1753.2
Dustload		kg m^{-3}	1 10^{-7}
Breathing rate adult		m^3 h^{-1}	1
Dilution factor	Soil in garden		0.0096
Dilution factor	Soil under house		0.096

*Taken from NRPB/HPA W36 (Oatway & Mobbs, 2003)

Values from (Augean, 2009a), unless otherwise stated

External irradiation

847. Dose from external irradiation while living and working on contaminated soil is given by (Augean, 2009a):

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

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});
- $Clean$ 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.

848. Parameter values are summarised in Table 100. Note that the NRPB/HPA W36 consumption rates (Oatway & Mobbs, 2003) are smaller than the consumption rates employed in the previous site assessment (Augean, 2009a). This is because it is assumed that a residential occupant 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.

849. Dose conversion factors for irradiation are given in Table 170.

Inhalation of contaminated soil

850. Dose from inhalation of contaminated soil is given by (Augean, 2009a):

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

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 100 and dose coefficients for inhalation are given in Table 170.

Inhalation of gases

851. The assessment calculations presented for the residential housing scenario include a contribution based on gas migration from underlying waste (see Section E.5.6) 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 600, respectively.

852. 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.5.7.3. Dose to Residential Occupant on site after 150 years

853. In Table 101 the dose rates to residents on the site following construction of houses 150 years after site capping are presented. The largest contributions to dose arise from Pa-231, Th-232, Ag-110m, I-129 and Nb-94. The impact of Radium placement depth within the ENRMF on intrusion and radon release is discussed in the next section (see paragraph 861).

Table 101 Dose to site residents after 150 years

Radionuclide	Maximum inventory (MBq)	Dose to site Resident at 150 y (μSv y^{-1} MBq^{-1})	Dose from maximum inventory (μSv y^{-1})
H-3	$8.96\ 10^{7}$	$3.53\ 10^{-10}$	$3.16\ 10^{-2}$
C-14	$8.96\ 10^{7}$	$4.35\ 10^{-6}$	$3.90\ 10^{2}$
Cl-36	$1.48\ 10^{6}$	$5.47\ 10^{-6}$	8.08
Fe-55	$8.96\ 10^{7}$	$1.11\ 10^{-26}$	$9.97\ 10^{-19}$
Co-60	$8.96\ 10^{7}$	$8.26\ 10^{-14}$	$7.40\ 10^{-6}$
Ni-63	$8.96\ 10^{7}$	$1.91\ 10^{-9}$	$1.71\ 10^{-1}$

Radionuclide	Maximum inventory (MBq)	Dose to site Resident at 150 y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Sr-90	$8.96\ 10^{7}$	$1.13\ 10^{-6}$	$1.02\ 10^{2}$
Nb-94	$8.96\ 10^{7}$	$1.81\ 10^{-5}$	$1.62\ 10^{3}$
Tc-99	$8.96\ 10^{7}$	$7.52\ 10^{-6}$	$6.73\ 10^{2}$
Ru-106	$8.96\ 10^{7}$	$1.80\ 10^{-50}$	$1.62\ 10^{-42}$
Ag-108m	$8.96\ 10^{7}$	$1.41\ 10^{-5}$	$1.26\ 10^{3}$
Sb-125	$8.96\ 10^{7}$	$1.98\ 10^{-22}$	$1.77\ 10^{-14}$
Sn-126	$8.96\ 10^{7}$	$5.35\ 10^{-6}$	$4.80\ 10^{2}$
I-129	$4.17\ 10^{4}$	$1.30\ 10^{-5}$	$5.43\ 10^{-1}$
Ba-133	$8.96\ 10^{7}$	$1.90\ 10^{-10}$	$1.70\ 10^{-2}$
Cs-134	$8.96\ 10^{7}$	$2.50\ 10^{-27}$	$2.24\ 10^{-19}$
Cs-137	$8.96\ 10^{7}$	$2.18\ 10^{-7}$	$1.95\ 10^{1}$
Pm-147	$8.96\ 10^{7}$	$7.47\ 10^{-27}$	$6.69\ 10^{-19}$
Eu-152	$8.96\ 10^{7}$	$6.06\ 10^{-9}$	$5.43\ 10^{-1}$
Eu-154	$8.96\ 10^{7}$	$8.02\ 10^{-11}$	$7.18\ 10^{-3}$
Eu-155	$8.96\ 10^{7}$	$1.12\ 10^{-16}$	$1.01\ 10^{-8}$
Pb-210	$8.96\ 10^{7}$	$2.19\ 10^{-7}$	$1.96\ 10^{1}$
Ra-226*	$8.96\ 10^{7}$	$4.75\ 10^{-2}$	$1.46\ 10^{-3}$
Ra-228	$8.96\ 10^{7}$	$9.87\ 10^{-13}$	$8.84\ 10^{-5}$
Ac-227	$8.96\ 10^{7}$	$6.63\ 10^{-8}$	5.94
Th-229	$8.96\ 10^{7}$	$4.85\ 10^{-6}$	$4.35\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$8.55\ 10^{-6}$	$5.93\ 10^{2}$
Th-232	$7.16\ 10^{7}$	$3.24\ 10^{-5}$	$2.32\ 10^{3}$
Pa-231	$1.86\ 10^{7}$	$4.25\ 10^{-5}$	$7.89\ 10^{2}$
U-232	$8.96\ 10^{7}$	$1.76\ 10^{-7}$	$1.57\ 10^{1}$
U-233	$3.13\ 10^{7}$	$2.11\ 10^{-7}$	6.62
U-234	$6.41\ 10^{6}$	$1.36\ 10^{-7}$	$8.74\ 10^{-1}$
U-235	$4.92\ 10^{6}$	$1.66\ 10^{-6}$	8.15
U-236	$8.96\ 10^{7}$	$1.28\ 10^{-7}$	$1.15\ 10^{1}$
U-238	$2.53\ 10^{7}$	$3.76\ 10^{-7}$	9.52
Np-237	$4.52\ 10^{5}$	$2.92\ 10^{-6}$	1.32
Pu-238	$8.96\ 10^{7}$	$2.53\ 10^{-7}$	$2.27\ 10^{1}$
Pu-239	$8.96\ 10^{7}$	$8.99\ 10^{-7}$	$8.05\ 10^{1}$
Pu-240	$8.96\ 10^{7}$	$8.88\ 10^{-7}$	$7.96\ 10^{1}$
Pu-241	$8.96\ 10^{7}$	$2.39\ 10^{-8}$	2.14
Pu-242	$8.96\ 10^{7}$	$8.44\ 10^{-7}$	$7.56\ 10^{1}$
Am-241	$8.96\ 10^{7}$	$6.93\ 10^{-7}$	$6.21\ 10^{1}$
Cm-243	$8.96\ 10^{7}$	$4.66\ 10^{-8}$	4.18
Cm-244	$8.96\ 10^{7}$	$3.84\ 10^{-9}$	$3.44\ 10^{-1}$

* Assumes Ra-226 distributed at any depth with other LLW.

854. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.5.

E.5.8. Excavation for Housing – radon exposure from a house on spoil

855. This case considers building a house on a spoil/waste mix.

856. 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.

857. This case considers long-term occupation of the former landfill site, and thus long-term potential exposure to contaminated wastes.

E.5.8.1. Assessment calculation for radon exposure

858. 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 46;
- 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 see Table 40; and,
- h_{soil} is the thickness of the soil (m);

859. 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.

860. The effective relaxation length for soil is 0.2 m and the thickness of soil is assumed to be 0.1 m.

E.5.8.2. Dose from radon when building on a waste/spoil mix

861. In Table 102 the results of assessment calculations for radon gas and a dilution factor of 0.096 are presented for waste containing 5 Bq g^{-1} of Ra-226.

Table 102 Radon inhalation doses and radiological capacities for a dwelling built on a waste/spoil mix

Case	Indoor Rn-222 activity concentration ($Bq\ m^{-3}\ MBq^{-1}$)	Adult inhalation effective dose ($mSv\ y^{-1}$ for 5 $Bq\ g^{-1}$)	Radiological capacity based on radon dose and 3 mSv/y (TBq)
Buried with other LLW at any depth	$2.82\ 10^{-5}$	2.66	2.53

862. The calculations imply that the activity concentration of Ra-226 in the wastes that are excavated that will meet the 3 mSv dose criterion is about 5.6 $Bq\ g^{-1}$.However, this restriction only applies to the activity concentration of Ra-226 in the wastes that are excavated as the scenario is only relevant if a dwelling is built on a spoil/waste mixture containing the radium bearing waste. It does not impose restrictions on the Ra-226 activity concentration of wastes that remain in the site. 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.

863. 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.3.3 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.

864. 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 radium imposed constraints upon the site's capacity to be minimised.

865. 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 2.6 m since LLW is not placed within the top 1 m of a cell and the cap is 1.6 m thick. A value of 5 $Bq\ g^{-1}$, corresponding to the activity concentration specified in the NORM exemption level (see paragraph 96), has been used to limit disposals in the upper layers of waste cells.

866. 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 East Northamptonshire is 45 $Bq\ m^{-3}$ (Augean, 2009c) with a maximum recorded background radon level of 2000 $Bq\ m^{-3}$. If the quantities of radium emplaced in the ENRMF are equal to the radiological capacity given in Table 102 above, then this would result in an indoor Rn-222 activity concentration of approximately 70 $Bq\ m^{-3}$ (below the HPA action level).

E.5.9. Excavation for a smallholding

E.5.9.1. Assessment calculations for the Smallholder

867. The smallholding case is conceptually similar to the long-term residential occupant described in Section E.5.7, 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, 2009a):

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

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}).

868. 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 0.0096 as discussed above (see paragraph 837). 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 172 and dose coefficients for ingestion are given in Table 170. Relevant parameters for the smallholding scenario are given in Table 103 and animal produce transfer factors are given in Table 173.

Table 103 Parameters for smallholding scenario

Parameter	Substance	Value
Yield (crops)	Green vegetables	3.0 kg m^{-2} y^{-1}
	Root vegetables	3.5 kg m^{-2} y^{-1}
	Pasture	1.7 kg m^{-2} y^{-1}
Consumption rate (animal)	Pasture	55 kg d^{-1}
	Soils	0.6 kg d^{-1}
Occupancy Indoors		0.75 y y^{-1}
Occupancy outdoors		0.25 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}
Dilution factor	Soil on land	0.0096
Dilution factor	Soil under house	0.096

Values taken from (Augean, 2009a)

869. The assessment calculations presented for the smallholding scenario also include a gas contribution based on gas migration from underlying waste (see Section E.5.6) 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.

870. 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 0.03 m for the small holding.

E.5.9.2. Dose to Smallholder on site after 200 years

871. In Table 104 the dose rates to a smallholder on the site following construction of a slurry pit 200 years after site capping are presented. The largest dose rates arise from I-129 and Pa-231.

Table 104 Dose to smallholders after 200 years

Radionuclide	Maximum inventory (MBq)	Dose to Smallholder at 200 y (µSv y^{-1} MBq^{-1})	Dose from maximum inventory (µSv y^{-1})
H-3	8.96 10^{7}	2.81 10^{-11}	2.52 10^{-3}
C-14	8.96 10^{7}	3.71 10^{-6}	3.32 10^{2}
Cl-36	1.48 10^{6}	5.55 10^{-5}	8.20 10^{1}
Fe-55	8.96 10^{7}	7.76 10^{-31}	6.95 10^{-23}
Co-60	8.96 10^{7}	1.35 10^{-16}	1.21 10^{-8}
Ni-63	8.96 10^{7}	1.08 10^{-8}	9.69 10^{-1}
Sr-90	8.96 10^{7}	2.42 10^{-6}	2.17 10^{2}
Nb-94	8.96 10^{7}	2.09 10^{-5}	1.87 10^{3}
Tc-99	8.96 10^{7}	3.31 10^{-5}	2.96 10^{3}
Ru-106	8.96 10^{7}	6.64 10^{-65}	5.95 10^{-57}

Radionuclide	Maximum inventory (MBq)	Dose to Smallholder at 200 y ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Dose from maximum inventory ($\mu Sv\ y^{-1}$)
Ag-108m	$8.96\ 10^{7}$	$1.50\ 10^{-5}$	$1.34\ 10^{3}$
Sb-125	$8.96\ 10^{7}$	$8.08\ 10^{-28}$	$7.24\ 10^{-20}$
Sn-126	$8.96\ 10^{7}$	$8.33\ 10^{-6}$	$7.47\ 10^{2}$
I-129	$4.17\ 10^{4}$	$1.12\ 10^{-4}$	4.65
Ba-133	$8.96\ 10^{7}$	$8.22\ 10^{-12}$	$7.37\ 10^{-4}$
Cs-134	$8.96\ 10^{7}$	$1.92\ 10^{-34}$	$1.72\ 10^{-26}$
Cs-137	$8.96\ 10^{7}$	$1.23\ 10^{-7}$	$1.10\ 10^{1}$
Pm-147	$8.96\ 10^{7}$	$5.81\ 10^{-32}$	$5.21\ 10^{-24}$
Eu-152	$8.96\ 10^{7}$	$5.41\ 10^{-10}$	$4.85\ 10^{-2}$
Eu-154	$8.96\ 10^{7}$	$1.64\ 10^{-12}$	$1.47\ 10^{-4}$
Eu-155	$8.96\ 10^{7}$	$9.02\ 10^{-20}$	$8.08\ 10^{-12}$
Pb-210	$8.96\ 10^{7}$	$1.97\ 10^{-7}$	$1.77\ 10^{1}$
Ra-226*	$8.96\ 10^{7}$	$1.40\ 10^{-2}$	$1.34\ 10^{-3}$
Ra-228	$8.96\ 10^{7}$	$7.39\ 10^{-15}$	$6.62\ 10^{-7}$
Ac-227	$8.96\ 10^{7}$	$2.37\ 10^{-8}$	2.13
Th-229	$8.96\ 10^{7}$	$8.17\ 10^{-6}$	$7.32\ 10^{2}$
Th-230	$6.93\ 10^{7}$	$4.33\ 10^{-5}$	$3.00\ 10^{3}$
Th-232	$7.16\ 10^{7}$	$4.19\ 10^{-5}$	$3.00\ 10^{3}$
Pa-231	$1.86\ 10^{7}$	$1.61\ 10^{-4}$	$3.00\ 10^{3}$
U-232	$8.96\ 10^{7}$	$3.27\ 10^{-7}$	$2.93\ 10^{1}$
U-233	$3.13\ 10^{7}$	$5.57\ 10^{-7}$	$1.75\ 10^{1}$
U-234	$6.41\ 10^{6}$	$3.88\ 10^{-7}$	2.48
U-235	$4.92\ 10^{6}$	$2.63\ 10^{-6}$	$1.29\ 10^{1}$
U-236	$8.96\ 10^{7}$	$3.67\ 10^{-7}$	$3.29\ 10^{1}$
U-238	$2.53\ 10^{7}$	$6.59\ 10^{-7}$	$1.67\ 10^{1}$
Np-237	$4.52\ 10^{5}$	$5.32\ 10^{-6}$	2.40
Pu-238	$8.96\ 10^{7}$	$3.18\ 10^{-7}$	$2.85\ 10^{1}$
Pu-239	$8.96\ 10^{7}$	$1.67\ 10^{-6}$	$1.50\ 10^{2}$
Pu-240	$8.96\ 10^{7}$	$1.65\ 10^{-6}$	$1.47\ 10^{2}$
Pu-241	$8.96\ 10^{7}$	$4.50\ 10^{-8}$	4.03
Pu-242	$8.96\ 10^{7}$	$1.59\ 10^{-6}$	$1.42\ 10^{2}$
Am-241	$8.96\ 10^{7}$	$1.30\ 10^{-6}$	$1.17\ 10^{2}$
Cm-243	$8.96\ 10^{7}$	$2.13\ 10^{-8}$	1.91
Cm-244	$8.96\ 10^{7}$	$4.95\ 10^{-9}$	$4.43\ 10^{-1}$

* Assumes Ra-226 distributed at any depth with other LLW.

872. 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 105. There are a small number of cases where animal products result in larger dose rates (e.g. Cl-36, Cs-134 and Cs-137).

Table 105 Contributing foodstuff doses in the diet of a smallholder

Radionuclide	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)			
	Root vegetables	Green vegetables	Meat	Milk
H-3	$3.73\ 10^{-18}$	$1.01\ 10^{-18}$	$1.06\ 10^{-18}$	$3.80\ 10^{-18}$
C-14	$8.35\ 10^{-14}$	$4.87\ 10^{-14}$	$7.13\ 10^{-13}$	$2.04\ 10^{-13}$
Cl-36	$6.86\ 10^{-12}$	$4.00\ 10^{-12}$	$1.90\ 10^{-11}$	$2.57\ 10^{-11}$
Fe-55	$3.19\ 10^{-38}$	$5.72\ 10^{-39}$	$7.11\ 10^{-37}$	$1.87\ 10^{-39}$
Co-60	$1.23\ 10^{-24}$	$7.59\ 10^{-25}$	$6.77\ 10^{-26}$	$1.08\ 10^{-26}$
Ni-63	$3.60\ 10^{-15}$	$9.70\ 10^{-16}$	$1.81\ 10^{-16}$	$6.03\ 10^{-15}$
Sr-90	$3.30\ 10^{-14}$	$1.47\ 10^{-12}$	$1.86\ 10^{-13}$	$7.30\ 10^{-13}$
Nb-94	$5.40\ 10^{-14}$	$3.32\ 10^{-14}$	$3.29\ 10^{-19}$	$2.40\ 10^{-18}$
Tc-99	$2.04\ 10^{-11}$	$1.26\ 10^{-11}$	$1.99\ 10^{-14}$	$2.44\ 10^{-14}$
Ru-106	$3.14\ 10^{-72}$	$3.39\ 10^{-73}$	$2.37\ 10^{-71}$	$2.74\ 10^{-75}$
Ag-108m	$6.86\ 10^{-15}$	$3.84\ 10^{-16}$	$2.48\ 10^{-16}$	$4.31\ 10^{-15}$
Sb-125	$6.26\ 10^{-36}$	$3.85\ 10^{-36}$	$5.09\ 10^{-39}$	$1.70\ 10^{-38}$
Sn-126	$1.61\ 10^{-12}$	$9.90\ 10^{-13}$	$6.27\ 10^{-14}$	$1.76\ 10^{-13}$
I-129	$1.62\ 10^{-11}$	$9.47\ 10^{-12}$	$4.62\ 10^{-11}$	$3.96\ 10^{-11}$
Ba-133	$9.07\ 10^{-20}$	$9.77\ 10^{-21}$	$2.25\ 10^{-21}$	$2.35\ 10^{-20}$
Cs-134	$5.88\ 10^{-42}$	$3.43\ 10^{-42}$	$2.57\ 10^{-41}$	$1.39\ 10^{-41}$
Cs-137	$5.81\ 10^{-15}$	$3.39\ 10^{-15}$	$2.54\ 10^{-14}$	$1.38\ 10^{-14}$
Pm-147	$2.78\ 10^{-38}$	$7.48\ 10^{-39}$	$1.91\ 10^{-38}$	$1.33\ 10^{-40}$
Eu-152	$4.80\ 10^{-19}$	$2.96\ 10^{-19}$	$1.02\ 10^{-20}$	$5.77\ 10^{-21}$
Eu-154	$1.89\ 10^{-21}$	$1.16\ 10^{-21}$	$4.01\ 10^{-23}$	$2.27\ 10^{-23}$
Eu-155	$6.94\ 10^{-28}$	$4.27\ 10^{-28}$	$1.47\ 10^{-29}$	$8.34\ 10^{-30}$
Pb-210	$1.17\ 10^{-13}$	$7.22\ 10^{-14}$	$9.54\ 10^{-16}$	$3.81\ 10^{-15}$
Ra-226*	$2.54\ 10^{-10}$	$1.57\ 10^{-10}$	$2.84\ 10^{-12}$	$2.18\ 10^{-11}$
Ra-228	$3.61\ 10^{-21}$	$2.22\ 10^{-21}$	$4.02\ 10^{-23}$	$3.09\ 10^{-22}$
Ac-227	$6.64\ 10^{-15}$	$4.08\ 10^{-15}$	$1.23\ 10^{-16}$	$1.64\ 10^{-18}$
Th-229	$9.61\ 10^{-13}$	$2.59\ 10^{-13}$	$1.75\ 10^{-12}$	$5.69\ 10^{-15}$
Th-230	$2.33\ 10^{-11}$	$1.42\ 10^{-11}$	$8.68\ 10^{-13}$	$1.97\ 10^{-12}$
Th-232	$1.69\ 10^{-12}$	$4.56\ 10^{-13}$	$3.09\ 10^{-12}$	$1.00\ 10^{-14}$
Pa-231	$9.42\ 10^{-11}$	$5.80\ 10^{-11}$	$1.27\ 10^{-13}$	$3.08\ 10^{-14}$
U-232	$1.41\ 10^{-13}$	$8.68\ 10^{-14}$	$4.90\ 10^{-15}$	$3.48\ 10^{-14}$
U-233	$1.81\ 10^{-13}$	$1.05\ 10^{-13}$	$3.91\ 10^{-14}$	$4.03\ 10^{-14}$
U-234	$1.57\ 10^{-13}$	$9.65\ 10^{-14}$	$6.54\ 10^{-15}$	$3.87\ 10^{-14}$
U-235	$5.35\ 10^{-13}$	$3.29\ 10^{-13}$	$5.49\ 10^{-15}$	$3.75\ 10^{-14}$
U-236	$1.50\ 10^{-13}$	$9.25\ 10^{-14}$	$5.22\ 10^{-15}$	$3.71\ 10^{-14}$
U-238	$1.55\ 10^{-13}$	$9.53\ 10^{-14}$	$5.38\ 10^{-15}$	$3.82\ 10^{-14}$
Np-237	$3.55\ 10^{-13}$	$2.18\ 10^{-12}$	$5.49\ 10^{-14}$	$1.50\ 10^{-15}$
Pu-238	$1.51\ 10^{-13}$	$9.34\ 10^{-15}$	$1.77\ 10^{-16}$	$1.14\ 10^{-16}$
Pu-239	$7.95\ 10^{-13}$	$4.89\ 10^{-14}$	$9.21\ 10^{-16}$	$5.40\ 10^{-16}$
Pu-240	$7.83\ 10^{-13}$	$4.82\ 10^{-14}$	$9.07\ 10^{-16}$	$5.31\ 10^{-16}$

Radionuclide	Dose per MBq (µSv y^{-1} MBq^{-1})			
	Root vegetables	Green vegetables	Meat	Milk
Pu-241	1.60 10^{-14}	9.84 10^{-15}	9.91 10^{-17}	1.98 10^{-17}
Pu-242	7.67 10^{-13}	4.72 10^{-14}	8.89 10^{-16}	5.21 10^{-16}
Am-241	4.64 10^{-13}	2.86 10^{-13}	2.88 10^{-15}	5.74 10^{-16}
Cm-243	5.04 10^{-15}	3.10 10^{-16}	4.85 10^{-17}	3.17 10^{-18}
Cm-244	2.35 10^{-15}	1.45 10^{-16}	4.61 10^{-18}	1.59 10^{-18}

* Emplaced at any depth with other LLW

873. The doses calculated using illustrative inventories are considered further in Appendix G, Section G.5.

E.5.10. Excavation of particles

874. The dose implications of excavation of waste materials that consist of different sized objects ranging from particles to large contaminated items, such as concrete blocks, are also considered.

875. Radioactive particles are small discrete items that could be as small as a grain of sand and could be incorporated in a radioactive waste stream or package. The approach used draws on the work undertaken for the LLWR ESC (Sumerling, 2013) and considers the possibility that future intrusion events could lead to unintentional recovery of, and exposure to, radioactive particles. Migration of particles in groundwater or uptake from soil into the foodchain is not considered credible. The LLWR ESC considered a set of particles with different radionuclide characteristics and these same particles were considered here.

876. Following the approach in the LLWR ESC (Sumerling, 2013), the two scenarios considered are:

- exposure of waste and subsequent occupancy; and,
- drilling through waste and handling retrieved material.

E.5.10.1. Particle characteristics

877. A number of potential and hypothetical particulate LLW wastes have been identified (Mobbs & Sumerling, 2012) and these were used in LLWR ESC assessments (Sumerling, 2013). These are described below. It needs to be stressed that these calculations are illustrative and they do not imply that any of these particles are intended for disposal at the ENRMF. The calculations are intended to provide guidance on the levels of activity associated with particles that might be acceptable for disposal at the ENRMF where their disposal is the BAT option. The major areas of uncertainty are the time at which exposure occurs (following emplacement of the waste), particle size and (for the ingestion pathway) the dissolution fraction in the gastro-intestinal tract.

878. **Dounreay beach (fuel) particles** - Particles representative of higher activity finds around Dounreay, with a total activity of 15,000 Bq (9,000 Bq Pu-238, 1500 Bq Pu-

239, 1500 Bq Pu-240, 3000 Bq Am-241). It is assumed that disposal of the particle occurs immediately after discovery (i.e. there is no period of radioactive decay prior to disposal).

879. **Dalgety Bay (radium) particles** – Particles representative of items containing radium paint. Two military specifications for paint are known to have existed and both are considered. ZnS paint contains 50 µg per g Ra-226; 'Admiralty specification' paint contains 215 µg per g Ra-226.

880. **Sellafield beach particles** - A particle representative of the highest alpha-rich particle is assumed. This particle has a total activity of 1.03 10^6 Bq (comprising 8.4 10^4 Bq Pu-238, 1.54 10^5 Bq Pu-239, 1.54 10^5 Bq Pu-240 and 6.34 10^5 Bq Am-241). This is considered to be the limiting case. It is assumed that disposal of the particle occurs immediately after discovery (i.e. there is no period of radioactive decay prior to disposal). Other types of particles are known to have been recovered from the environment around Sellafield (dominated by Cs-137 or Sr-90) and a few Co-60 rich (fuel cladding) particles have also been recovered. These are not considered here.

881. **Thorium sands** - Thorium rich monazite and thorianite sands have been disposed of to LLWR. It is assumed here that the Th-232 content ranges from 5% to 70% of the particle mass.

882. **Uranium particles** - A hypothetical waste including uranium particles is assumed. The uranium is represented as either natural uranium (with a U-235 of 0.72%) or enriched uranium (with 3.5% U-235).

883. **Irradiated fuel** - A hypothetical waste including particles of spent fuel is assumed. This case is taken to bound the maximum activities present in particles. Three fuel types are identified:

- Magnox fuel (natural uranium as metal, 6 GWd t^{-1} burn-up);
- AGR fuel (2.4% U-235 enriched as UO_2, 20 GWd t^{-1} burn-up); and,
- PWR fuel (3.4% U-235 enriched as UO_2, 35 GWd t^{-1} burn-up).

E.5.10.2. Assessment calculations for particles

884. Drilling through waste or exposure of waste (through natural processes of erosion or through deliberate human activity) could lead to recovery of particles. In either case, exposure will be through one of three pathways:

- ingestion;
- inhalation; and,
- external irradiation.

885. Following unintentional recovery of a radioactive particle it may be inadvertently ingested. 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. Dose is estimated on a per particle basis. Deliberate but accidental ingestion of larger items is not considered explicitly since the dose, if ingested, depends on the activity on the item rather than the size.

886. Inhalation of particles is also size restricted and in this case an upper limit of 10 µm diameter (0.01 mm) is assumed. The LLWR ESC (Sumerling, 2013) demonstrated that inhalation was not an important pathway and therefore this is not considered further here.

887. External exposure is not limited by size of particle. However, in order to be conservative it is assumed that the particle becomes lodged in direct contact with the skin (for example under a fingernail or toenail) and remains in situ for 8 hours. Consistent with this assumption and with the nominal size of particles identified for ingestion, a 1 mm diameter is assumed.

888. The doses due to each of these pathways are not considered to be additive. A small particle may be lodged on the skin and then unintentionally transferred to the mouth and ingested, and this is considered as a sensitivity assessment.

889. Dose is thus calculated as:

$$Dose_{ext,wb} = G_{wb}^{Rn}.T.A_{Rn}(t)$$

$$Dose_{ext,skin} = G_{skin}^{Rn}.T.A_{Rn}(t)$$

where:

- $Dose_{ext,wb}$ is the external effective (whole body) dose;
- $Dose_{ext,skin}$ is the skin (organ) dose;
- G_{skin}^{Rn} is the point-source effective dose rate for radionuclide *Rn* in contact with the skin (mSv hour^{-1} Bq^{-1});
- G_{wb}^{Rn} is the whole body dose rate for radionuclide *Rn* (mSv hour^{-1} Bq^{-1});
- $A_{Rn}(t)$ is the activity of the contamination (Bq) at the time of exposure (t); and,
- T is the exposure time (hours).

$$Dose_{ing} = D_{ing}^{Rn}.Sol_{Rn}.A_{Rn}(t)$$

$$Dose_{inh} = D_{inh}^{Rn}.A_{Rn}(t)$$

where:

- D_{inh} and D_{ing} are the dose coefficients for inhalation and ingestion of radionuclide *Rn* (Sv Bq^{-1} and Sv Bq^{-1} respectively);
- $A_{Rn}(t)$ is the activity of the contamination (Bq) at the time of exposure (t); and,
- Sol_{Rn} is the solubility of the particle in the gastro-intestinal tract.

890. It is assumed conservatively that such exposure occurs 60 or 300 years from emplacement of the waste, as a result of deliberate excavation of the site.

E.5.10.3. Dose from particles after 60 and 300 years

891. The calculations for exposure to 1 mm particles are presented in this section. The precise dimensions do not determine the dose providing the particle is sufficiently large that it is not respirable but sufficiently small that it remains inadvertently ingestible (e.g. a particle anywhere in the range of 1 to a few mm diameter will deliver the same dose with the same probability). A larger item (fragment) with the same activity will deliver the same ingestion and external dose but will deliver a smaller skin dose due to a shorter contact time with the same area of skin, and self-absorption within the fragment. The calculations implicitly assume that a particle is brought to the surface by the intrusion and then this particle is encountered by a person through inadvertent ingestion, skin exposure or external exposure. No account is taken of the probability of this inadvertent encounter (the radioactive particle will be one of many other similar sized particles of uncontaminated soil and waste). The calculations also cover deliberate identification and encounter with larger sized items with the same activity level.

892. Measurements [(HPA, 2005a) (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 are unrealistically high. Following the LLWR approach, two uptake factors were considered: the standard ICRP uptake factor (conservative case) and the experimentally determined uptake factor (realistic case).

893. The doses due to the different particles types are discussed. It needs to be stressed that these calculations are illustrative and they do not imply that any of these particles are intended for disposal at the ENRMF. The calculations are intended to provide guidance on the levels of activity associated with particles that might be acceptable for disposal at the ENRMF where their disposal is the BAT option.

Dounreay beach (fuel) particles

894. The default fractional gastro-intestinal uptake factor (f_1) of 5 10^{-4} is used for Pu-alpha and Am-241 in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). However, a measured fractional uptake factor of 1 10^{-5} has been derived for actinide uptake from Dounreay particles (HPA, 2005a). This lower value is considered to be the best estimate (realistic) value for use in this ESC since it relates directly to measurements of particles using in-vitro and in-vivo methods.

895. Ingestion doses are presented for both uptake factors in Table 106, based on ingestion at 60 years from disposal or 300 years from disposal.

Table 106 Dose incurred from ingestion of a Dounreay beach (fuel) particle

Radionuclide	Initial activity (MBq)	Activity at 60 y (Bq)*	Activity at 300 y (Bq)*	Ingestion dose (mSv)			
				at 60 y		at 300 y	
				Conservative f_1=5 10^{-4}	Realistic f_1=1 10^{-5}	Conservative f1=5 10^{-4}	Realistic f1=1 10^{-5}
Pu-238	9.00 10^{-3}	5.60 10^{3}	8.40 10^{2}	1.28	4.93 10^{-2}	1.92 10^{-1}	7.40 10^{-3}
Pu-239	1.50 10^{-3}	1.50 10^{3}	1.49 10^{3}	3.41 10^{-1}	1.32 10^{-2}	3.39 10^{-1}	1.31 10^{-2}
Pu-240	1.50 10^{-3}	1.49 10^{3}	1.45 10^{3}	3.40 10^{-1}	1.31 10^{-2}	3.31 10^{-1}	1.28 10^{-2}

Radionuclide	Initial activity (MBq)	Activity at 60 y (Bq)*	Activity at 300 y (Bq)*	Ingestion dose (mSv)			
				at 60 y		at 300 y	
				Conservative f_1=5 10^{-4}	Realistic f_1=1 10^{-5}	Conservative f1=5 10^{-4}	Realistic f1=1 10^{-5}
Am-241	3.00 10^{-3}	2.72 10^{3}	1.85 10^{3}	6.21 10^{-1}	2.40 10^{-2}	4.23 10^{-1}	1.63 10^{-2}
Total	1.50 10^{-2}	1.13 10^{4}	5.64 10^{3}	2.58	9.96 10^{-2}	1.28	4.96 10^{-2}

*Does not take into account ingrowth (from Pu-241 if present)

896. A maximum dose of 2.6 mSv may be incurred if the particle is ingested at 60 years after emplacement, and assuming a conservative uptake factor of 5 10^{-4}. The dose assuming a more realistic uptake factor is 0.1 mSv at 60 years. The doses estimated at 300 years are 1.3 mSv and 0.05 mSv respectively for the conservative and realistic fractional sorption cases.

897. The dose in all cases should be compared with dose guidance levels in the NSGRA (Environment Agency, 2012a) for intrusion scenarios of 3 to 20 mSv, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, 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.

898. The external dose from contact with the skin is also estimated and is presented in Table 107. The effective dose is very low at all times and can be considered to be negligible by comparison to the ingestion dose. The skin dose is also very low (<0.4 mSv) by comparison to an organ dose limit of 50 mSv y^{-1} for members of the public.

Table 107 External dose due to skin contact with a Dounreay beach (fuel) particle

Radionuclide	Initial activity (MBq)	Activity at 60 y (Bq)*	Activity at 300 y (Bq)*	Dose (mSv)			
				at 60 y		at 300 y	
				Effective Dose (mSv)	Skin Dose (mSv)	Effective Dose (mSv)	Skin Dose (mSv)
Pu-238	9.00 10^{-3}	5.60 10^{3}	8.40 10^{2}	9.23 10^{-9}	8.92 10^{-2}	1.39 10^{-9}	1.34 10^{-2}
Pu-239	1.50 10^{-3}	1.50 10^{3}	1.49 10^{3}	1.95 10^{-10}	8.92 10^{-3}	1.94 10^{-10}	8.86 10^{-3}
Pu-240	1.50 10^{-3}	1.49 10^{3}	1.45 10^{3}	2.06 10^{-9}	2.25 10^{-2}	2.01 10^{-9}	2.20 10^{-2}
Am-241	3.00 10^{-3}	2.72 10^{3}	1.85 10^{3}	4.66 10^{-8}	2.62 10^{-1}	3.17 10^{-8}	1.78 10^{-1}
Total	1.50 10^{-2}	1.13 10^{4}	5.64 10^{3}	5.81 10^{-8}	3.82 10^{-1}	3.53 10^{-8}	2.22 10^{-1}

*Does not take into account ingrowth (from Pu-241 if present)

Dalgety Bay (radium) particles

899. It is assumed that the radium present in particles at Dalgety Bay has already aged 50 years (i.e. the paint dates from 1964). In practice, the radium painted dials may have originated in the 1930’s and 1940’s although the relatively long half-life of Ra-226 (1600 years) means that the dose estimates will be insensitive to these assumptions. Calculated activity in the particles is presented in Table 108.

900. A default fractional gastro-intestinal uptake factor (f_1) of 0.2 is assumed for Ra-226 in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). This is

based on a solubility of 100% in the gut. However, particle solubilities are reported by (Tyler, et al., 2013), that range from 0.0% to 36% (mean 5.7% and 95th percentile of 20.2%) based on an updated statistical analysis (COMARE, 2014). A solubility of 20% is used as the basis for a more cautiously realistic dose (COMARE, 2014), resulting in an effective f_1 of 0.04.

Table 108 Radium paint activities for 1 mm particles at 60 y and 300 y

Source	Particle density (g cm^{-3})	Activity (Bq) at 60 y	Activity (Bq) at 300 y
		1 mm	1 mm
ZnS paint (at 50 µg of radium g^{-1})	4.09	3.73 10^3	3.37 10^3
Admiralty specification paint (at 215 µg of radium g^{-1})	4.09	1.61 10^4	1.45 10^4

Table 109 Determination of ingestion doses from radium paint at 60 y and 300 y

Source	Ingestion dose (mSv) 1 mm particle			
	at 60 y		at 300 y	
	Conservative f_1=2 10^{-1}	Realistic f_1=4 10^{-2}	Conservative f_1=2 10^{-1}	Realistic f_1=4 10^{-2}
ZnS paint (50 µg Ra-226 per gram)	8.17	1.63	7.36	1.47
Admiralty specification paint (215 µg Ra-226 per gram)	3.51 10^1	7.02	3.17 10^1	6.33

901. It can be seen that a maximum dose of 35 mSv may be incurred if a 1 mm particle of Admiralty specification paint is ingested at 60 years after emplacement, and assuming a conservative f_1 of 0.2, and drops to 7 mSv when a 95th percentile solubility is applied (the cautiously realistic value). The dose assuming a realistic solubility of 0.057 is 2 mSv at 60 years. The doses estimated at 300 years are 32 mSv and 6.3 mSv respectively for the conservative and cautiously realistic cases, falling to 1.8 mSv when the mean solubility of 0.057 is applied.

902. The dose in all cases should be compared with dose guidance levels in the NSGRA (Environment Agency, 2012a) for intrusion scenarios of 3 to 20 mSv, where the lower end of the range is taken to apply to exposures that may persist over time. In this case, 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.

903. The external dose from contact with the skin is also estimated and is presented below (Table 110). The effective dose is very low at all times and can be considered to be negligible by comparison to the ingestion dose.

Table 110 Determination of external effective dose from radium paint at 60 y and 300 y

Source	Effective dose (mSv) 10 µm particle		Effective dose (mSv) 1 mm particle	
	60 y	300 y	60 y	300 y
ZnS paint (at 50 µg of radium g^{-1})	$7.25\ 10^{-12}$	$6.53\ 10^{-12}$	$7.25\ 10^{-6}$	$6.53\ 10^{-6}$
Admiralty specification paint (at 215 µg of radium g^{-1})	$3.12\ 10^{-11}$	$2.81\ 10^{-11}$	$3.12\ 10^{-5}$	$2.81\ 10^{-5}$

Sellafield beach particles

904. A default fractional gastro-intestinal uptake factor (f_1) of $5\ 10^{-4}$ is assumed for Pu-alpha and Am-241 in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). However, a measured fractional sorption of $3\ 10^{-5}$ has been derived for actinide uptake from Sellafield particles based on a series of in-vivo and in-vitro studies, (HPA, 2011) and (HPA, 2005b).

905. Ingestion doses are presented for both uptake factors in Table 111, based on ingestion at 60 years from disposal or 300 years from disposal. It can be seen that a maximum dose of 200 mSv may be incurred if the particle is ingested at 60 years after emplacement, and assuming a conservative uptake factor of $5\ 10^{-4}$. The dose assuming a more realistic sorption fraction is 17 mSv at 60 years. The doses estimated at 300 years are 160 mSv and 13 mSv respectively for the conservative and realistic fractional sorption cases.

906. The dose in all cases should be compared with dose guidance levels in the NSGRA (Environment Agency, 2012a) for intrusion scenarios of 3 to 20 mSv, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, 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.

907. A potential dose up to 200 mSv (using conservative uptake factors) clearly requires further investigation. Even if a more realistic estimate of 17 mSv is accepted, the potential dose from ingestion of such a particle is near to the upper end of the GRA guidance. In this context, it should be noted that the particle adopted for this assessment is the highest activity particle ever recovered from the vicinity of Sellafield and may be excluded by precluding the disposal of particles known to contain more than 1 MBq activity. The implications are discussed further in para 936 onwards.

Table 111 Dose incurred from ingestion of a Sellafield beach particle

Radionuclide	Initial activity (MBq)	Activity at 60 y (Bq)*	Activity at 300 y (Bq)*	Ingestion dose (mSv)			
				at 60 y		at 300 y	
				Conservative $f_1=5\ 10^{-4}$	Realistic $f_1=3\ 10^{-5}$	Conservative $f_1=5\ 10^{-4}$	Realistic $f_1=3\ 10^{-5}$
Pu-238	$8.40\ 10^{-2}$	$5.23\ 10^{4}$	$7.84\ 10^{3}$	$1.19\ 10^{1}$	$9.24\ 10^{-1}$	1.79	$1.39\ 10^{-1}$
Pu-239	$1.54\ 10^{-1}$	$1.54\ 10^{5}$	$1.53\ 10^{5}$	$3.86\ 10^{1}$	$2.92\ 10^{0}$	$3.83\ 10^{1}$	2.90
Pu-240	$1.54\ 10^{-1}$	$1.53\ 10^{5}$	$1.49\ 10^{5}$	$3.84\ 10^{1}$	$2.91\ 10^{0}$	$3.74\ 10^{1}$	2.83
Am-241	$6.34\ 10^{-1}$	$5.76\ 10^{5}$	$3.92\ 10^{5}$	$1.17\ 10^{2}$	$1.02\ 10^{1}$	$7.99\ 10^{1}$	6.93
Total	1.03	$9.35\ 10^{5}$	$7.02\ 10^{5}$	$2.06\ 10^{2}$	$1.69\ 10^{1}$	$1.57\ 10^{2}$	$1.28\ 10^{1}$

*Does not take into account ingrowth (from Pu-241 if present)

908. The external dose from contact with the skin is also estimated and is presented below (Table 112). The effective dose is very low at all times and can be considered to be negligible by comparison to the ingestion dose. However, the skin dose is up to 60 mSv; comparison to an organ dose limit of 50 mSv y^{-1} for members of the public reinforces the suggestion that WAC may be adopted to preclude particles with levels of activity of the order of 1 MBq. No serious deterministic effects would be expected at this level of dose.

Table 112 External dose due to skin contact with a Sellafield beach particle

Radionuclide	Initial activity (MBq)	Activity at 60 y (Bq)*	Activity at 300 y (Bq)*	External irradiation dose			
				at 60 y		at 300 y	
				Effective Dose (mSv)	Skin Dose (mSv)	Effective Dose (mSv)	Skin Dose (mSv)
Pu-238	$8.40\ 10^{-2}$	$5.23\ 10^{4}$	$7.84\ 10^{3}$	$8.62\ 10^{-8}$	$8.32\ 10^{-1}$	$1.29\ 10^{-8}$	$1.25\ 10^{-1}$
Pu-239	$1.54\ 10^{-1}$	$1.54\ 10^{5}$	$1.53\ 10^{5}$	$2.00\ 10^{-8}$	$9.16\ 10^{-1}$	$1.99\ 10^{-8}$	$9.10\ 10^{-1}$
Pu-240	$1.54\ 10^{-1}$	$1.53\ 10^{5}$	$1.49\ 10^{5}$	$2.12\ 10^{-7}$	2.31	$2.06\ 10^{-7}$	2.26
Am-241	$6.34\ 10^{-1}$	$5.76\ 10^{5}$	$3.92\ 10^{5}$	$9.81\ 10^{-6}$	$5.53\ 10^{1}$	$6.71\ 10^{-6}$	$3.76\ 10^{1}$
Total	1.03	$9.35\ 10^{5}$	$7.02\ 10^{5}$	$1.01\ 10^{-5}$	$5.93\ 10^{1}$	$6.95\ 10^{-6}$	$4.09\ 10^{1}$

*Does not take into account ingrowth (from Pu-241 if present)

Thorium sands

909. Th-232 is an extremely long lived radionuclide, with a half-life in excess of 10^{10} years. Consequently radioactive decay is not considered and dose estimates presented are taken to be applicable at all times post-disposal.

910. Three types of thorium rich sand are identified for this assessment (Table 113).

Table 113 Determination of specific activities for 1 mm particles of thorium rich sands

Source	Th-232 %	Density g cm^{-3}*	Material specific activity ($Bq\ g^{-1}$)	Activity of 1mm particles (Bq)
Typical monazite/thorite	5.0%	5.5	$2.03\ 10^{2}$	$5.84\ 10^{-1}$
High thorium monazite	30.0%	5.5	$1.22\ 10^{3}$	3.51
High thorium thorite	70.0%	5.5	$2.84\ 10^{3}$	8.18

*The density is set as a bounding case. Typical monazite sands have a density around 1.5 g cm-3. Lower density sands will have a correspondingly lower dose.

911. A default fractional gastro-intestinal uptake factor (f_1) of $5\ 10^{-4}$ is assumed for Th-232 in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). However, a measured fractional uptake factor of $2\ 10^{-4}$ is used as the basis for a more cautiously realistic dose (LLWR, 2011a).

912. Ingestion doses are presented for both uptake factors in Table 114. The maximum dose is of the order of 1 μSv and is considered to be very low. The external dose from contact with the skin is also estimated and is very low. The skin dose is also low (<0.2 mSv) in comparison to an organ dose limit of 50 mSv y^{-1} for members of the public.

Table 114 Dose incurred from ingestion of or skin contact with thorium sand particles

Source	Ingestion dose (mSv) 1 mm particle		Dose due to external irradiation (mSv)	
	Conservative f_1=5 10^{-4}	Realistic f_1=2 10^{-4}	Effective Dose (mSv)	Skin Dose (mSv)
Typical monazite/thorite	$1.35\ 10^{-4}$	$5.38\ 10^{-5}$	$8.34\ 10^{-10}$	$2.14\ 10^{-4}$
High thorium monazite	$8.10\ 10^{-4}$	$3.23\ 10^{-4}$	$5.01\ 10^{-9}$	$2.84\ 10^{-4}$
High thorium thorite	$1.89\ 10^{-3}$	$7.53\ 10^{-4}$	$1.17\ 10^{-8}$	$1.66\ 10^{-1}$

Uranium particles

913. All relevant uranium isotopes are extremely long lived radionuclide, with half-lives in excess of 10^{5} years. Consequently radioactive decay is not considered and dose estimates presented are taken to be applicable at all times post-disposal.

Table 115 Determination of activity concentrations for 1 mm particles of natural uranium particles

Isotopic composition		Activity concentration ($Bq\ g^{-1}$)	Activity concentration ($Bq\ g^{-1}\ U$)	Material Activity concentration ($Bq\ g^{-1}$)*	Activity of 1mm particles (Bq)
U-234	0.0054%	$2.30\ 10^{8}$	$1.24\ 10^{4}$	$1.09\ 10^{4}$	$6.31\ 10^{1}$
U-235	0.7204%	$7.99\ 10^{4}$	$5.76\ 10^{2}$	$5.08\ 10^{2}$	2.92
U-238	99.2742%	$1.24\ 10^{4}$	$1.23\ 10^{4}$	$1.09\ 10^{4}$	$6.27\ 10^{1}$
Total	100.0%	$2.30\ 10^{8}$	$2.54\ 10^{4}$	$2.23\ 10^{4}$	$1.29\ 10^{2}$

*based on uranium as 88.15% by mass of UO2 and a material density of 11 g cm^{-3}.

Table 116 Determination of activity concentrations for 1 mm particles of enriched uranium

Isotopic composition		Activity concentration ($Bq\ g^{-1}$)	Activity concentration ($Bq\ g^{-1}\ U$)	Material Activity concentration ($Bq\ g^{-1}$)*	Activity of 1mm particles (Bq)
U-234	0.0288%	$2.30\ 10^{8}$	$6.64\ 10^{4}$	$5.85\ 10^{4}$	$3.37\ 10^{2}$
U-235	3.50%	$7.99\ 10^{4}$	$2.80\ 10^{3}$	$2.47\ 10^{3}$	$1.42\ 10^{1}$
U-238	96.471%	$1.24\ 10^{4}$	$1.20\ 10^{4}$	$1.06\ 10^{4}$	$6.09\ 10^{1}$
Total	100.00%	$2.30\ 10^{8}$	$8.12\ 10^{4}$	$7.15\ 10^{4}$	$4.12\ 10^{2}$

*based on uranium as 88.15% by mass of UO2 and a material density of 11 g cm^{-3}.

914. A default fractional gastro-intestinal uptake factor (f_1) of 0.02 is assumed for uranium in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). However, a more cautiously realistic fractional uptake factor of 0.002 has been derived for uranium uptake (Sumerling, 2013). Ingestion dose estimates are presented in Table 117.

915. It can be seen that a maximum dose of approximately 0.02 mSv may be incurred if a 1 mm particle is ingested, assuming a conservative uptake factor of 0.02. The dose assuming a more realistic sorption fraction of 0.002 falls to around 3.4 μSv.

916. The dose in all cases should be compared with dose guidance levels in the NSGRA (Environment Agency, 2012a) for intrusion scenarios of 3 to 20 mSv, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, 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.

917. The external dose from contact with the skin is also estimated and is presented below. The effective dose is very low at all times and can be considered to be negligible by comparison to the ingestion dose. The skin dose is around 2 mSv and may be compared to an organ dose limit of 50 mSv y^{-1} for members of the public. In this case, although the skin dose remains low, it is the most limiting of the dose pathways assessed.

Table 117 Dose incurred from ingestion of or skin contact with natural or enriched uranium particles

Source	Ingestion dose (mSv) 1 mm particle		Dose due to external irradiation (mSv)	
	Conservative $f_1=2\ 10^{-2}$	Realistic $f_1=2\ 10^{-3}$	Effective Dose	Skin Dose
Natural uranium	$6.26\ 10^{-3}$	$1.02\ 10^{-3}$	$2.26\ 10^{-8}$	1.98
Enriched uranium	$2.03\ 10^{-2}$	$3.38\ 10^{-3}$	$2.45\ 10^{-8}$	2.18

Irradiated fuel

918. Following the approach taken in the LLWR ESC, particles known to derive from irradiated fuel were used to provide a bounding case for the consideration of all particulate disposals. Data are presented for fuel based on the characteristics of Magnox fuel, AGR fuel and PWR fuel, all of which may occur in the UK.

919. The principal radionuclides in each case are Pu-alpha and Am-241. In each case a default fractional gastro-intestinal uptake factor (f_1) of $5\ 10^{-4}$ is assumed in ICRP 72 (ICRP, 1996) and the LLWR Radiological Handbook (LLWR, 2011a). However, a more cautiously realistic fractional sorption of $1\ 10^{-5}$ was used, as determined for Dounreay fuel particles recovered from the environment (HPA, 2005a).

920. Dose estimates are presented only for the ingestion pathway.

Table 118 Activity of Magnox spent fuel particles (0.71% U-235; 6 GWd t^{-1}; uranium metal fuel)

Source	Activity of 1mm particle at 60 y (Bq)	Activity of 1mm particle at 300 y (Bq)
Pu-238	$2.15\ 10^{4}$	$3.94\ 10^{3}$
Pu-239	$2.76\ 10^{4}$	$2.74\ 10^{4}$
Pu-240	$4.24\ 10^{4}$	$4.14\ 10^{4}$
Am-241	$1.75\ 10^{5}$	$1.26\ 10^{5}$
Total	$2.66\ 10^{5}$	$1.99\ 10^{5}$

Table 119 Activity of AGR spent fuel particles (2.4% U-235; 20 GWd t^{-1}; uranium oxide fuel)

Source	Activity of 1mm particle at 60 y (Bq)	Activity of 1mm particle at 300 y (Bq)
Pu-238	$6.26\ 10^{4}$	$1.06\ 10^{4}$
Pu-239	$3.23\ 10^{4}$	$3.21\ 10^{4}$
Pu-240	$7.24\ 10^{4}$	$7.06\ 10^{4}$
Am-241	$2.75\ 10^{5}$	$1.98\ 10^{5}$
Total	$4.42\ 10^{5}$	$3.12\ 10^{5}$

Table 120 Activity of PWR spent fuel particles (3.4% U-235; 35 GWd t^{-1}; uranium oxide fuel)

Source	Activity of 1 mm particle at 60 y (Bq)	Activity of 1mm particle at 300 y (Bq)
Pu-238	$3.10\ 10^{5}$	$4.70\ 10^{4}$
Pu-239	$5.96\ 10^{4}$	$5.92\ 10^{4}$
Pu-240	$1.05\ 10^{5}$	$1.02\ 10^{5}$
Am-241	$7.58\ 10^{5}$	$5.48\ 10^{5}$
Total	$1.23\ 10^{6}$	$7.57\ 10^{5}$

921. The derived ingestion doses are presented below (Table 121). It can be seen that a maximum dose of approximately 280 mSv may be incurred if a 1 mm particle containing 1.2 MBq of spent PWR fuel is ingested, assuming a conservative uptake factor of $5\ 10^{-4}$. The dose assuming a more realistic sorption fraction of $1\ 10^{-5}$ is around 10 mSv. The implications are discussed in para 936 onwards.

922. The dose in all cases should be compared with dose guidance levels in the NSGRA (Environment Agency, 2012a)] for intrusion scenarios of 3 to 20 mSv, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, 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.

Table 121 Dose incurred from ingestion of irradiated fuel particles

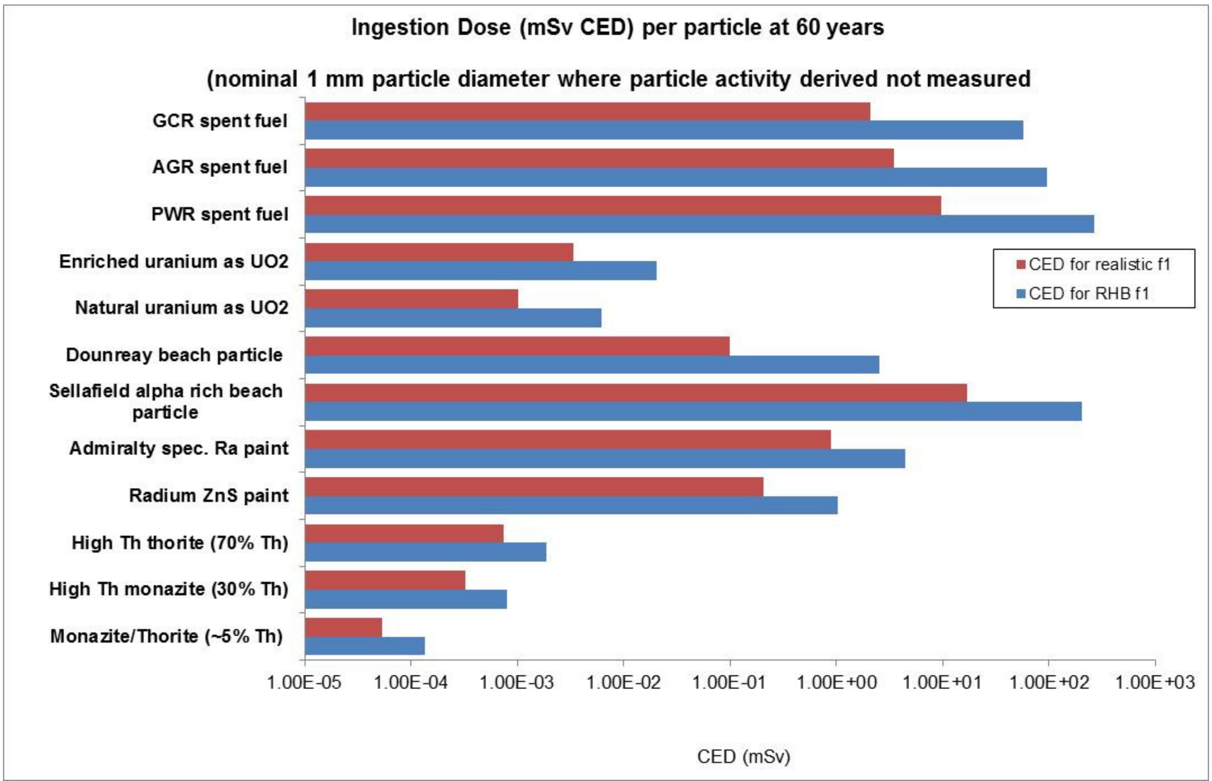
Radionuclide	Ingestion dose (mSv)			
	at 60 y		at 300 y	
	Conservative f_1=5 10^{-4}	Realistic f_1=1 10^{-5}	Conservative f1=5 10^{-4}	Realistic f1=1 10^{-5}
1 mm particle from Magnox fuel				
Pu-238	4.91	1.89 10^{-1}	8.99 10^{-1}	3.47 10^{-2}
Pu-239	6.92	2.48 10^{-1}	6.87	2.46 10^{-1}
Pu-240	1.07 10^{1}	3.82 10^{-1}	1.04 10^{1}	3.72 10^{-1}
Am-241	3.56 10^{1}	1.28	2.57 10^{1}	9.23 10^{-1}
Total	5.81 10^{1}	2.10	4.39 10^{1}	1.58
1 mm particle from AGR fuel				
Pu-238	1.43 10^{1}	5.51 10^{-1}	2.41	9.32 10^{-2}
Pu-239	8.10	2.90 10^{-1}	8.05	2.88 10^{-1}
Pu-240	1.82 10^{1}	6.52 10^{-1}	1.77 10^{1}	6.36 10^{-1}
Am-241	5.60 10^{1}	2.01	4.05 10^{1}	1.45
Total	9.66 10^{1}	3.50	6.86 10^{1}	2.47
1 mm particle from PWR fuel				
Pu-238	7.08 10^{1}	2.73	1.07 10^{1}	4.14 10^{-1}
Pu-239	1.50 10^{1}	5.37 10^{-1}	1.49 10^{1}	5.33 10^{-1}
Pu-240	2.63 10^{1}	9.44 10^{-1}	2.57 10^{1}	9.21 10^{-1}
Am-241	1.55 10^{2}	5.54	1.12 10^{2}	4.01
Total	2.67 10^{2}	9.76	1.63 10^{2}	5.88

Summary of particles results

923. The dose due to exposure to radioactive particles arising from human intrusion is dominated in general by the potential ingestion dose. The dose calculated is very sensitive to assumptions regarding fractional absorption across the gastro-intestinal tract. As discussed earlier, measurements have found that particles are not 100% soluble in the gastro-intestinal tract and therefore the dose results using the ICRP gut uptake factors are unrealistically high.

924. The doses estimated using more realistic values of gut uptake factors range from fractions of a mSv for thorium sands and Dounreay beach particles, to several mSv for radium paint particles, to 10 mSv for a 1 mm spent fuel particle and 17 mSv for the highest alpha rich particle identified at Sellafield. The results are summarised below.

Figure 19. Summary of ingestion doses determined for a range of particulates at 60 y post disposal



925. In practice it is likely that all the calculated doses are conservative following disposal. In addition to radioactive decay, particles are likely to be subject to fragmentation and, in some cases, corrosion. For spent fuel particles the corrosion chemistry of plutonium can be complex. However, for uranium, a corrosion rate of 1.3 10^{-3} mg cm^{-2} h^{-1} in the presence of water vapour can be used to determine that a 1 mm particle will corrode entirely within about 80 years.

926. In many cases the radioactive half-lives of the radionuclides of principal concern are very long and hence the dose estimates are relatively insensitive to the time of exposure. However, the probability of unintentionally recovering a particle and then ingesting the particle is likely to be very low, although this has not been addressed here.

927. The results show that the doses from the particle types and activity levels considered here are below the GRA intrusion dose criteria when using the realistic gut uptake values. They can also be used to determine the activity on a particle or fragment that would meet the GRA intrusion dose criteria and this would form part of the WAC. This is the approach taken at LLWR: the WAC specifies activity limits for high activity particles that are based on the GRA intrusion dose criteria; wastes that do not comply with the WAC are not accepted without specific approval from EA. Demonstration that the disposal route adopted represents BAT is also required.

E.5.11. Excavation of large contaminated items

928. This section considers the implications of disposing of large contaminated items, such as concrete blocks, with a heterogeneous activity distribution profile.

929. 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.

930. 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.

931. 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.

932. A number of different cases 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).

933. 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.
- Exposure of waste and subsequent occupancy.
- Drilling through waste and handling retrieved material.

934. Dissolution in leachate is addressed in Section E.3.4 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.

935. 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.5.9). Hence the use of the waste for cultivation is not considered further here.

936. 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 due to external exposure or inhalation of dust or inadvertent ingestion of dust. These two scenarios are considered further.

937. Following exposure of the waste, occupancy of the area may lead to external exposure and inhalation of dust may occur. Inadvertent ingestion is considered less likely in this scenario but is included for completeness. It is assumed that excavation work will occur only after the end of the period of authorisation. Natural erosion of the landfill surface will depend on processes believed to be credible in the region. In this case, for an inland site in the UK, such erosion is likely to lie many thousands of years in the future.

938. A series of boreholes may 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.

939. These exposure periods are consistent with recent discussions with, and assessments presented to, the Environment Agency concerning disposal of concrete blocks at the ENRMF site (Wilmot, 2014).

E.5.11.1. Waste characteristics

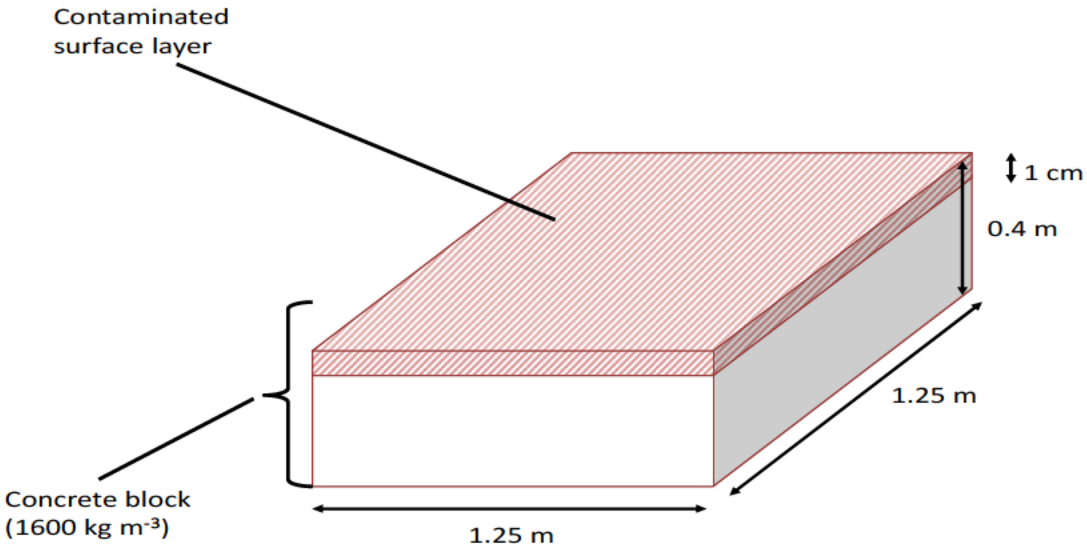
940. Six large item waste streams are characterised and detailed below:

- concrete slabs from decommissioning a Fuel Element Debris (FED) storage silo;
- activated concrete shielding blocks;
- concrete and rubble from building demolition;
- crushed concrete, soil and rubble from building demolition;
- reinforced concrete from dismantling a research facility; and,
- a hypothetical concrete block contaminated with Cs-137.

941. **Concrete demolition slabs** – Contaminated concrete slabs from a FED storage facility (Figure 20). 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 20. Contaminated concrete block



942. **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.

943. **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}.

944. **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}.

945. **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.

946. **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.

947. 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.5.11.2. Assessment calculation for large contaminated items

948. 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.

949. 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 171;
- 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 driller (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*.

950. 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.

951. The whole-body effective dose is determined assuming that the person is, on average, 2 m the waste. Dust in air from the core is assumed to be present at 1 10^{-6} kg m^{-3}. These and other assumptions are tabulated below (Table 122).

Table 122 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}	1.00E-06	dust load of contaminated waste
M_{ing}	kg hr^{-1}	3.42E-06	rate of ingestion of dust
T	hr y^{-1}	52	exposure time
B	m^3 hr^{-1}	1	breathing rate

952. 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 also higher than in previous scenarios presenting the maximum for a single event rather than an average. This is considered to be appropriate since the activity envisaged in this scenario does not involve deliberate handling of the waste. The breathing rate is also somewhat lower, consistent with a more sedentary aspect.

953. The dose to a driller 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.

954. 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}).

955. 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 is assumed to be present at 1 10^{-6} kg m^{-3}. These and other assumptions are tabulated below (Table 123).

Table 123 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}	1.00E-06	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

956. It is assumed conservatively that intrusion occurs 60 years from emplacement of the waste.

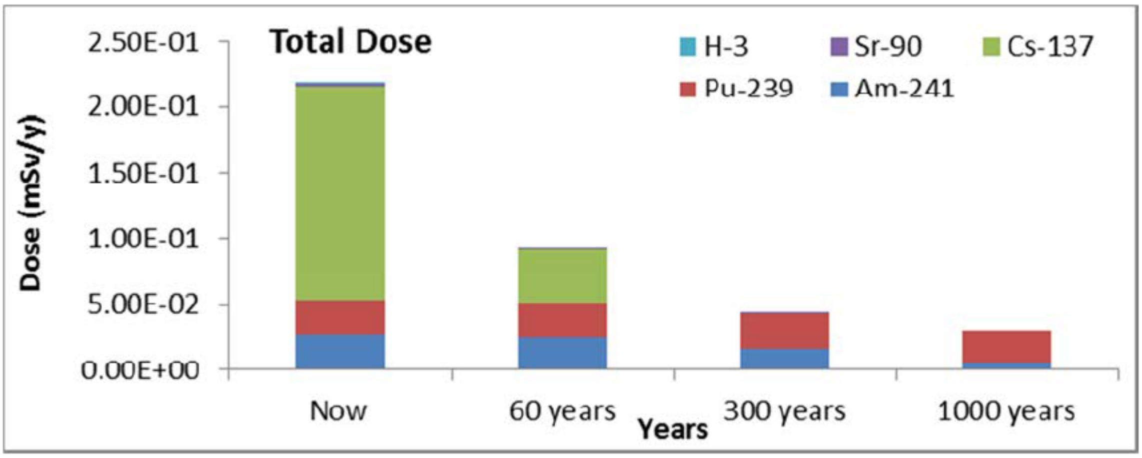
E.5.11.3. Dose from large contaminated waste items

957. The doses to the site occupant and to the site investigator from each of the five characteristic waste types are presented in this section. Consideration of the sensitivity of the doses to certain input parameters is presented in Section E.7.3.

Concrete demolition slabs - dose to site occupant

958. 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 21. 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 0.093 mSv y^{-1} (93 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 21). Exposure before 60 years is not credible as it is within the period of authorisation.

Figure 21. Time-dependant dose to site occupant from contaminated concrete demolition slabs



959. 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 124). The dose at 150 years would be dominated by ingestion and inhalation of Pu-239 and Am-241.

Table 124 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	$4.76\ 10^{-3}$	$1.50\ 10^{-2}$	$5.01\ 10^{-3}$	$2.47\ 10^{-2}$
Pu-239	$2.55\ 10^{-5}$	$1.98\ 10^{-2}$	$6.80\ 10^{-3}$	$2.66\ 10^{-2}$
Cs-137	$4.06\ 10^{-2}$	$4.64\ 10^{-7}$	$9.24\ 10^{-5}$	$4.07\ 10^{-2}$
Sr-90	$3.47\ 10^{-4}$	$3.45\ 10^{-6}$	$1.94\ 10^{-4}$	$5.45\ 10^{-4}$
H-3	0	$6.07\ 10^{-10}$	$3.85\ 10^{-8}$	$3.92\ 10^{-8}$
Total	$4.57\ 10^{-2}$	$3.47\ 10^{-2}$	$1.21\ 10^{-2}$	$9.26\ 10^{-2}$

960. For human intrusion situations, thedose 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 1000 years (a hypothetical earliest date at which 'natural' erosion could expose the waste) gives a dose estimate of 0.03 $mSv\ y^{-1}$, dominated by the ingestion and inhalation of dust containing Pu-239. This dose is equivalent to an annual risk of around $1.5\ 10^{-6}$. Given the grossly conservative nature of the assumption that exposure could occur at 1000 years, and that the waste would be exposed in such a fashion that the contaminated surface 1 cm is uniformly exposed, 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

961. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is 0.0095 mSv (9.5 µSv) per core handled (Figure 22). The dose is dominated by inhalation and unintentional ingestion of Pu-239 and Am-241 (see Table 125).

Figure 22. Time-dependant dose to site investigator from contaminated concrete demolition slabs

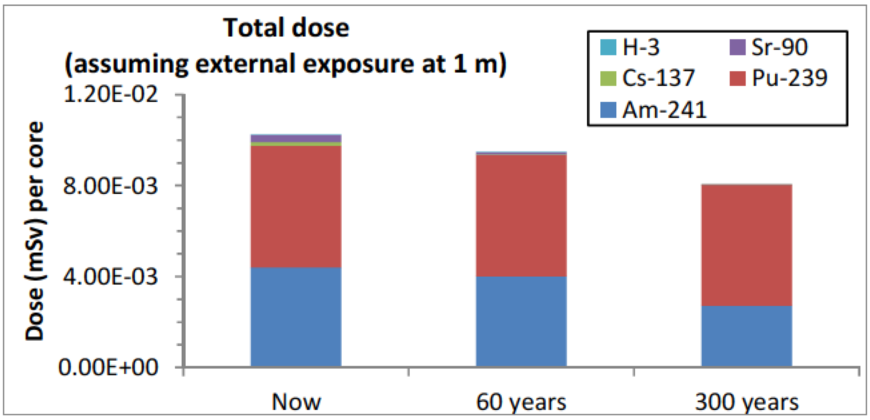


Table 125 Pathway-dependant dose to site investigator from contaminated concrete demolition slabs at 60 years

Radionuclide	Dose ($mSv\ y^{-1}$) at 60 years			
	External	Inhalation	Ingestion	Total
Am-241	$7.42\ 10^{-8}$	$2.07\ 10^{-3}$	$1.93\ 10^{-3}$	$4.00\ 10^{-3}$
Pu-239	$6.21\ 10^{-10}$	$2.74\ 10^{-3}$	$2.62\ 10^{-3}$	$5.35\ 10^{-3}$
Cs-137	$6.37\ 10^{-7}$	$6.42\ 10^{-8}$	$3.55\ 10^{-5}$	$3.62\ 10^{-5}$
Sr-90	$3.44\ 10^{-7}$	$4.78\ 10^{-7}$	$7.46\ 10^{-5}$	$7.54\ 10^{-5}$
H-3	0	$8.40\ 10^{-11}$	$1.48\ 10^{-8}$	$1.49\ 10^{-8}$
Total	$1.06\ 10^{-6}$	$4.81\ 10^{-3}$	$4.65\ 10^{-3}$	$9.46\ 10^{-3}$

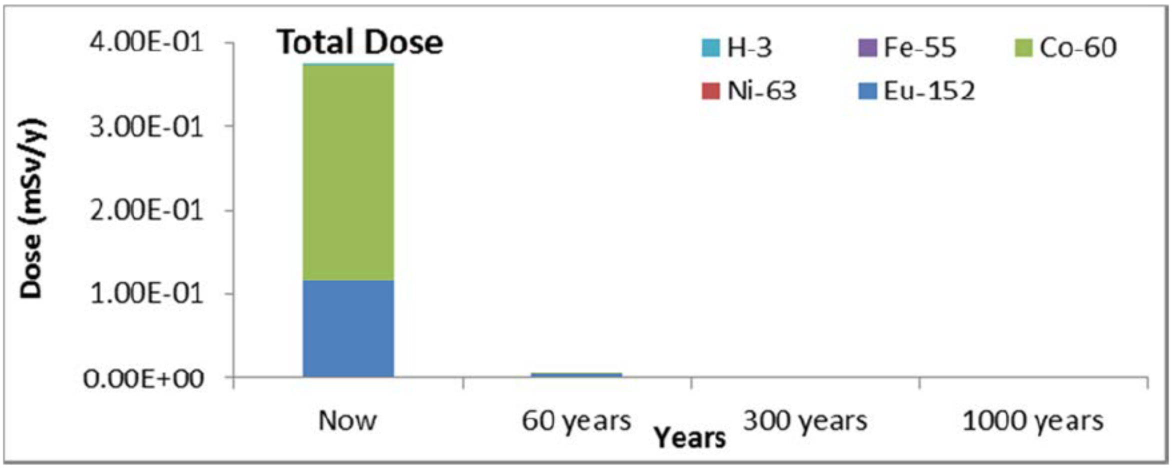
962. The dose incurred at 60 years from emplacement of waste is low. Even if 10 cores were handled, all with similar characteristics, the dose would remain more than a factor of 30 below the lower dose guidance level of 3 mSv for human intrusion scenarios. The potential dose to skin from close handling 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. Comparison may be made with a skin organ dose limit for members of the public of 50 $mSv\ y^{-1}$.

Activated concrete shielding blocks - dose to Site Occupant

963. 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 $Bq\ g^{-1}$.

964. 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 0.005 mSv y^{-1}. The dose is initially dominated by Co-60 and Eu-152 (Figure 23) 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 23. Time-dependant dose to site occupant from activated concrete shielding blocks



965. The dose at 60 years is dominated by the external component from Eu-152 (Table 126). The dose at later times is negligible.

Table 126 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	$5.42\ 10^{-3}$	$2.81\ 10^{-7}$	$6.27\ 10^{-7}$	$5.42\ 10^{-3}$
Ni-63	0	$4.55\ 10^{-8}$	$9.99\ 10^{-10}$	$4.65\ 10^{-8}$
Co-60	$9.55\ 10^{-5}$	$5.56\ 10^{-10}$	$1.28\ 10^{-8}$	$9.55\ 10^{-5}$
Fe-55	0	$1.45\ 10^{-14}$	$8.46\ 10^{-13}$	$8.61\ 10^{-13}$
H-3	0	$2.23\ 10^{-10}$	$1.42\ 10^{-8}$	$1.44\ 10^{-8}$
Total	$5.51\ 10^{-3}$	$3.27\ 10^{-7}$	$6.55\ 10^{-7}$	$5.52\ 10^{-3}$

Activated concrete shielding blocks - dose to Site Investigator

966. 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 24). 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 127).

Figure 24. Time-dependant dose to site investigator from activated shielding blocks

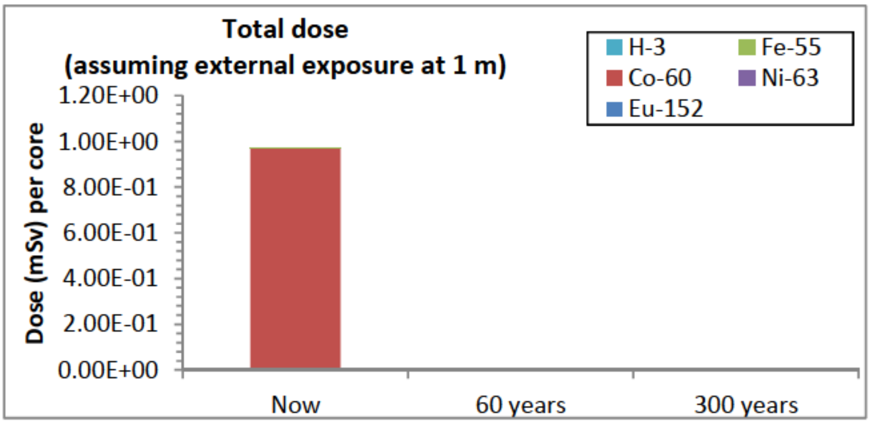


Table 127 Pathway-dependant dose to site investigator from contaminated concrete demolition slabs at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Eu-152	$7.44\ 10^{-8}$	$3.89\ 10^{-8}$	$2.41\ 10^{-7}$	$3.54\ 10^{-7}$
Ni-63	0	$6.29\ 10^{-9}$	$3.84\ 10^{-7}$	$3.91\ 10^{-7}$
Co-60	$3.63\ 10^{-4}$	$7.70\ 10^{-11}$	$4.91\ 10^{-9}$	$3.64\ 10^{-4}$
Fe-55	$5.45\ 10^{-16}$	$2.01\ 10^{-15}$	$3.26\ 10^{-13}$	$3.28\ 10^{-13}$
H-3	0	$3.09\ 10^{-11}$	$5.46\ 10^{-9}$	$5.49\ 10^{-9}$
Total	$3.64\ 10^{-4}$	$4.53\ 10^{-8}$	$2.52\ 10^{-7}$	$3.64\ 10^{-4}$

967. 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.

968. 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

969. 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.

970. 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 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

971. 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

972. 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.

973. 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 25 and Table 128.

Figure 25. Time-dependant dose to site occupant from crushed concrete

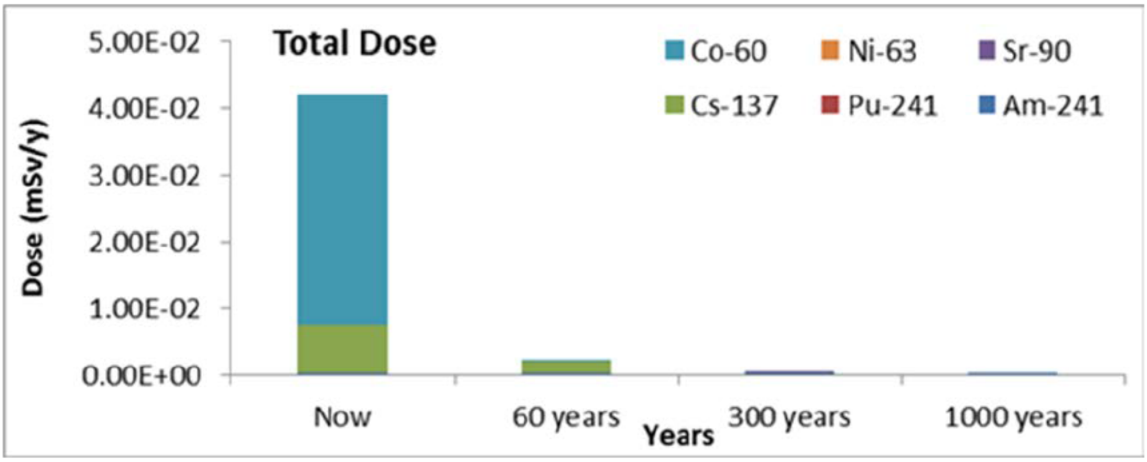


Table 128 Pathway-dependant dose to site occupant from crushed concrete

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Am-241	8.49 10^{-5}	1.31 10^{-4}	4.39 10^{-5}	2.60 10^{-4}

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Pu-241	$7.03\ 10^{-10}$	$1.74\ 10^{-7}$	$6.28\ 10^{-8}$	$2.37\ 10^{-7}$
Cs-137	$1.83\ 10^{-3}$	$4.07\ 10^{-9}$	$8.10\ 10^{-7}$	$1.83\ 10^{-3}$
Sr-90	$1.24\ 10^{-5}$	$3.03\ 10^{-8}$	$1.70\ 10^{-6}$	$1.42\ 10^{-5}$
Ni-63	0	$1.08\ 10^{-9}$	$2.38\ 10^{-11}$	$1.11\ 10^{-9}$
Co-60	$1.30\ 10^{-5}$	$1.32\ 10^{-11}$	$3.04\ 10^{-10}$	$1.30\ 10^{-5}$
Total	$1.94\ 10^{-3}$	$1.32\ 10^{-4}$	$4.65\ 10^{-5}$	$2.12\ 10^{-3}$

Building rubble 2 - dose to Site Investigator

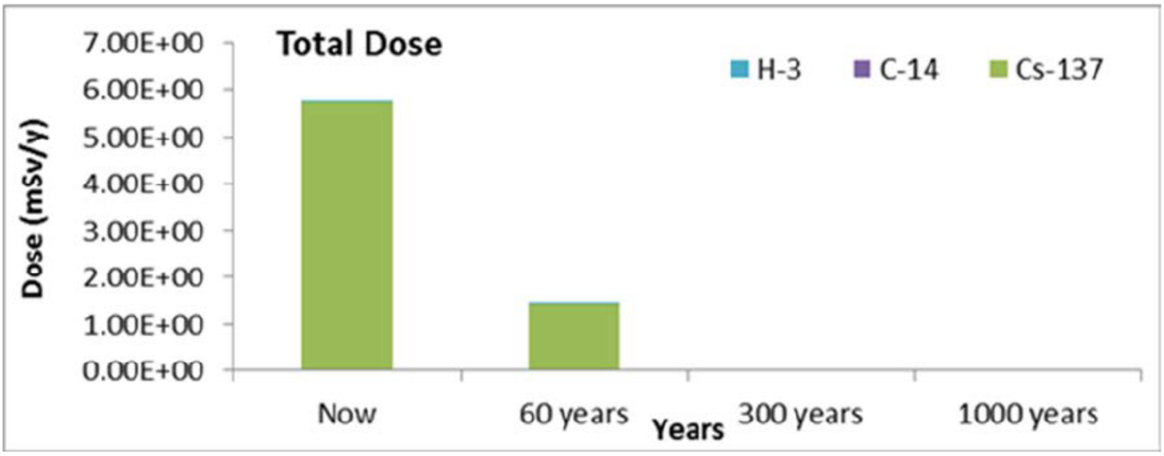
974. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is ca. 1.6 µSv per core handled and is dominated by the presence of Am-241 in the ingestion / inhalation pathways.

Reinforced concrete - dose to Site Occupant

975. 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}.

976. 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 1.4 mSv y^{-1} (Figure 26).

Figure 26. Time-dependant dose to site occupant from contaminated reinforced concrete



977. The dose is dominated by external exposure from Cs-137 (Table 129).

Table 129 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	$1.64\ 10^{-5}$	$3.27\ 10^{-3}$	1.44
C-14	$7.82\ 10^{-7}$	$3.21\ 10^{-7}$	$6.28\ 10^{-6}$	$7.38\ 10^{-6}$
H-3	0	$2.69\ 10^{-9}$	$1.71\ 10^{-7}$	$1.73\ 10^{-7}$
Total	1.44	$1.68\ 10^{-5}$	$3.28\ 10^{-3}$	1.44

Reinforced concrete - dose to Site Investigator

978. The equivalent dose to a site geotechnical worker / investigator taking borehole samples at 60 years is 0.001 mSv (1 µSv) per core handled (Figure 27). The dose is dominated by unintentional ingestion of Cs-137 (Table 130).

Figure 27. Time-dependant dose to site investigator from contaminated reinforced concrete

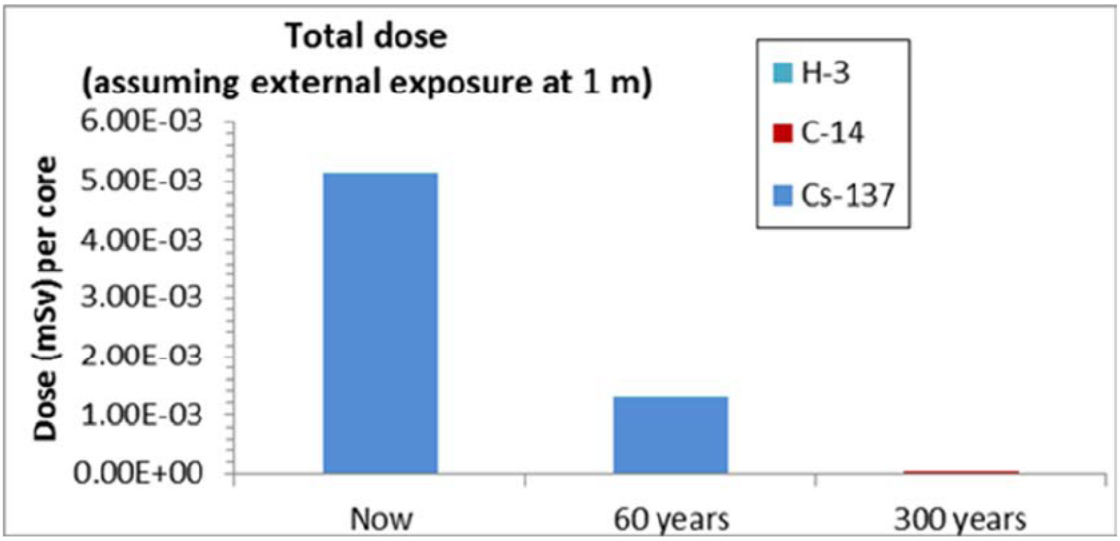


Table 130 Pathway-dependant dose to site investigator from contaminated reinforced concrete at 60 years

Radionuclide	Dose (mSv y^{-1}) at 60 years			
	External	Inhalation	Ingestion	Total
Cs-137	$2.26\ 10^{-5}$	$2.27\ 10^{-6}$	$1.26\ 10^{-3}$	$1.28\ 10^{-3}$
C-14	0	$4.44\ 10^{-8}$	$2.41\ 10^{-6}$	$2.46\ 10^{-6}$
H-3	0	$3.72\ 10^{-10}$	$6.57\ 10^{-8}$	$6.60\ 10^{-8}$
Total	$2.26\ 10^{-5}$	$2.32\ 10^{-6}$	$1.26\ 10^{-3}$	$1.29\ 10^{-3}$

979. 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.

980. 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

981. 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.

982. 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

983. 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.

984. 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. Impact on non-human species {R9}

E.6.1. Exposure to wildlife from all sources

985. A radiological assessment of the potential effects on non-human biota (NHB) from the disposal of LLW at the ENRMF (including the western extension) 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.

986. 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. The most recent update was uploaded on 6 November 2014, and that is the version of the tool used in this assessment. The updated ERICA tool contains revised wildlife concentration factors and associated updated EMCL values (following EMRAS II programme developments). The Tier 1 and Tier 2 approaches within the assessment tool developed as part of the ERICA project have been used to support the ESC.

987. The ERICA toolkit allows consideration of three ecosystems: terrestrial, freshwater and marine. The terrestrial and freshwater ecosystems are applicable to the environment surrounding the ENRMF. Within these ecosystems, the ERICA tool considers a range of organisms and wildlife groups as shown in Table 131.

Table 131 Wildlife groups considered in the ERICA tool

Terrestrial	Freshwater
Amphibian	Amphibian
Annelid	Benthic fish
Anthropod - detritivorous	Bird
Bird	Crustacean
Flying insects	Insect larvae
Grasses and herbs	Mammal
Lichen and bryophytes	Mollusc – bivalve
Mammal large	Mollusc – gastropod
Mammal small – burrowing	Pelagic fish
Mollusc - gastropod	Phytoplankton
Reptile	Reptile
Shrub	Vascular plant
Tree	Zooplankton

988. 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.

989. Input data for the non-human biota (NHB) dose assessment are radioactivity concentrations in soil and air (terrestrial ecosystem assessment) and water or sediment (freshwater ecosystem assessment). The activity concentrations of radionuclides in soil and water are calculated using the same approaches detailed in the ESC and underlying the dose calculations to the public.

990. The impact on burrowing animals that dig into the waste is also considered, based on activity concentrations in the waste.

E.6.2. The ERICA toolkit

991. 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 1.2.0, updated 23.12.2014. The release notes indicate there have been significant amendments to default radionuclides, reference organisms, DCCs, occupancy factors, default transfer parameters, Environmental Media Concentration Levels (EMCLs) and Benchmarks.

992. Tier 1 assessments are media concentration based 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.

993. This assessment has been undertaken using a Tier 1 approach for the ERICA terrestrial and freshwater ecosystems. Tier 2 assessments have been undertaken to add Ag-108m to the list of radionuclides, and to investigate the dose to burrowing animals in the waste cells after the Period of Authorisation. 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.

994. Within the terrestrial and freshwater ecosystems, the ERICA Tool considers a range of wildlife groups considered to be representative (see Table 131). 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).

995. 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'.

996. It is considered that the range of organism types represented within the ERICA assessment tool is sufficiently broad to characterise the reference ecosystems.

997. Within the Tier 1 assessment, the ERICA tool compares environmental concentrations for individual radionuclides with 'limiting concentrations' calculated using generic assumptions about the ecosystem and based on the application of a single dose screening value. These limiting values are based on a screening dose rate of 10 µGy h^{-1}.

E.6.3. ERICA assessments

998. Different approaches are available to derive numerical benchmark values, and a detailed explanation and proposed framework to demonstrate protection of non-human biota 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 non-human biota have gained relatively widespread application.

999. In the UK and Europe, the EC ERICA and PROTECT derived screening value of 10 µGy h^{-1} above background is generally recognised. 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).

1000. The FREDERICA database is available online from www.frederica-online.org. Access to the database requires registration but is free of charge. 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.

1001. 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.

1002. The Environment Agency for England and Wales also 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 the ENRMF.

1003. The 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 considers that adverse impact is unlikely.

E.6.4. ERICA results for a Terrestrial Ecosystem

1004. 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.

1005. 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.

1006. In addition, it is noted that not all radionuclides considered in the ENRMF are available in the ERICA assessment tool. Of the 12 that are 'missing' in Tier 1, the radionuclides Sn-126, Ba-133, Pm-147, Eu-155 and Ac-227 represent a small fraction in the UK Radioactive Waste Inventory; Fe-55 has a short half-life (2.7 years);. U-232, U-233 and U-234 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, Ag-108m, was the only one that was considered further.

1007. A Tier 2 assessment was undertaken to determine a limiting concentration in soil for Ag-108m. A conservative approach was taken to complete missing input parameters. The limiting concentration in a terrestrial ecosystem, using the ERICA screening dose rate of 10 µGy h^{-1},was found to be 1.12 10^4 Bq kg^{-1} in soil.

1008. Table 132 presents the limiting concentrations in soil and air for the radionuclides considered in the wildlife assessment of the terrestrial ecosystem. In each case, the organism type used to determine the limiting concentration is identified.

Table 132 Radionuclide specific limiting environmental activity concentrations in the terrestrial ecosystem derived from ERICA and corresponding to 10 µGy h^{-1}

Radionuclide	Limiting concentrations in terrestrial ecosystem		
	Soil (Bq kg^{-1} d.w.)	Air (Bq m^{-3})	Limiting organism
H-3		2,640	Bird
C-14		84.0	Mammal – small-burrowing
Cl-36	1,200		Grasses and Herbs
Co-60	7,300		Amphibian
Ni-63	8.77 10^5		Lichen & Bryophytes
Sr-90	2,030		Lichen & Bryophytes
Nb-94	1.15 10^4		Annelid
Tc-99	4,610		Grasses and Herbs
Ru-106	2,240		Lichen & Bryophytes
Ag-108m	1.12 10^4		Amphibian
Sb-125	4.41 10^4		Annelid
I-129	1.59 10^5		Mammal – large
Cs-134	1,220		Mammal – large
Cs-137	2,280		Mammal – large
Eu-152	1.72 10^4		Annelid
Eu-154	1.55 10^4		Amphibian
Pb-210	6,250		Lichen & Bryophytes
Ra-226	27.7		Lichen & Bryophytes
Ra-228	1.28 10^4		Lichen & Bryophytes
Th-230	266		Lichen & Bryophytes
Pa-231	96.2		Lichen & Bryophytes
Th-232	311		Lichen & Bryophytes
U-234	117		Lichen & Bryophytes
U-235	126		Lichen & Bryophytes
U-238	133		Lichen & Bryophytes
Np-237	100		Lichen & Bryophytes
Pu-238	752		Lichen & Bryophytes
Pu-239	800		Lichen & Bryophytes
Pu-240	800		Lichen & Bryophytes
Pu-241	2.94 10^6		Lichen & Bryophytes
Am-241	86.2		Lichen & Bryophytes
Cm-243	80.0		Lichen & Bryophytes
Cm-244	80.0		Lichen & Bryophytes

1009. Section 7.4.2.2 explains how the maximum inventory for each radionuclide has been derived from the different assessment scenarios. The result is summarised in paragraph 303: for some radionuclides, the maximum inventory is set by the 89.6 TBq (based on disposal of 448,000 t at a maximum specific activity of 200 Bq g-1); for the other radionuclides, at least one of the assessment scenarios was more limiting, as summarised in Table 25.

1010. Peak radionuclide concentrations in soil for the terrestrial ecosystem assessment were taken from the GoldSim results for the irrigation scenario (see Table 133). The soil and air 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 and the scaled values are also shown.

Table 133 Peak modelled radionuclide activity concentrations in terrestrial environmental media

Radionuclide	Daughter	Activity Concentration in terrestrial ecosystem, scaled to the maximum inventory of each radionuclide in the ENRMF	
		Soil (Bq kg^{-1})	Air (Bq m^{-3})
H-3			$1.37\ 10^{-9}$
C-14			$4.60\ 10^{-7}$
Cl-36		1.29	
Co-60		0	
Ni-63		$2.54\ 10^{-4}$	
Sr-90		$1.90\ 10^{-4}$	
Nb-94		0.235	
Tc-99		0.518	
Ru-106		0	
Ag-108m		0.0812	
Sb-125		0	
I-129		0.0424	
Cs-134		0	
Cs-137		$1.17\ 10^{-6}$	
Eu-152		$6.92\ 10^{-8}$	
Eu-154		0	
Pb-210		$1.46\ 10^{-7}$	
Ra-226		$6.03\ 10^{-3}$	
	Pb-210	$6.14\ 10^{-3}$	
Ra-228		0	
Th-230		0.321	
	Ra-226	0.226	
	Pb-210	0.223	
Pa-231		0.0585	
Th-232		8.94	
	Ra-228	8.91	
U-234		0.125	
	Th-230	3.49	
	Ra-226	2.39	
	Pb-210	2.36	

U-235		0.204	
	Pa-231	1.88	
U-238		1.05	
	U-234	0.924	
	Th-230	29.9	
	Ra-226	20.5	
	Pb-210	20.2	
Np-237		0.174	
Pu-238		$1.01\ 10^{-5}$	
	U-234	$6.24\ 10^{-4}$	
	Th-230	0.0174	
	Ra-226	0.0119	
	Pb-210	0.0118	
Pu-239		0.267	
	U-235	$1.27\ 10^{-4}$	
	Pa-231	$1.17\ 10^{-3}$	
Pu-240		0.0573	
	Th-230	$1.60\ 10^{-6}$	
	Ra-228	$1.60\ 10^{-6}$	
Pu-241		$8.10\ 10^{-9}$	
	Am-241	$1.34\ 10^{-5}$	
	Np-237	$2.31\ 10^{-4}$	
Am-241		$3.82\ 10^{-4}$	
	Np-237	$6.94\ 10^{-3}$	
Cm-243		$3.35\ 10^{-7}$	
	Pu-239	$3.23\ 10^{-4}$	
	U-235	$1.53\ 10^{-7}$	
	Pa-231	$1.41\ 10^{-6}$	
Cm-244		$5.13\ 10^{-8}$	
	Pu-240	$1.58\ 10^{-4}$	
	Th-232	$4.41\ 10^{-9}$	
	Ra-228	$4.40\ 10^{-9}$	

1011. The ERICA assessment tool is 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 screening dose rate of 10 µGy h^{-1}.

1012. Table 134 below summarises the results of the wildlife assessment.

Table 134 Radionuclide specific risk quotients for terrestrial ecosystems for the most limiting organisms based on a generic screening level of 10 µGy h^{-1}

Radionuclide	Daughter	Terrestrial ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
H-3		$5.20\ 10^{-13}$		Bird
C-14		$5.48\ 10^{-9}$		Mammal – small-burrowing
Cl-36		$1.07\ 10^{-3}$		Grasses and herbs
Co-60		0		
Ni-63		$2.89\ 10^{-10}$		Lichen & Bryophytes
Sr-90		$9.35\ 10^{-8}$		Lichen & Bryophytes
Nb-94		$2.04\ 10^{-5}$		Annelid
Tc-99		$1.12\ 10^{-4}$		Grasses and herbs
Ru-106		0		
Ag-108m		$7.25\ 10^{-6}$		Amphibian
Sb-125		0		
I-129		$2.67\ 10^{-7}$		Mammal – large
Cs-134		0		
Cs-137		$5.14\ 10^{-10}$		Mammal – large
Eu-152		$4.02\ 10^{-12}$		Annelid
Eu-154		0		
Pb-210		$2.34\ 10^{-11}$		Lichen & Bryophytes
Ra-226		$2.18\ 10^{-4}$	$2.19\ 10^{-4}$	Lichen & Bryophytes
	Pb-210	$9.82\ 10^{-7}$		Lichen & Bryophytes
Ra-228		0		
Th-230		$1.21\ 10^{-3}$	$9.40\ 10^{-3}$	Lichen & Bryophites
	Ra-226	$8.15\ 10^{-3}$		Lichen & Bryophytes
	Pb-210	$3.57\ 10^{-5}$		Lichen & Bryophytes
Pa-231		$6.08\ 10^{-4}$		Lichen & Bryophytes
Th-232		0.0287	0.0294	Lichen & Bryophytes
	Ra-228	$6.96\ 10^{-4}$		
U-234		$1.07\ 10^{-3}$	0.101	Lichen & Bryophytes
	Th-230	0.0131		Lichen & Bryophytes
	Ra-226	0.0863		Lichen & Bryophytes
	Pb-210	$3.77\ 10^{-4}$		Lichen & Bryophytes
U-235		$1.62\ 10^{-3}$	0.0211	Lichen & Bryophytes
	Pa-231	0.0195		Lichen & Bryophytes
U-238		$7.88\ 10^{-3}$	0.872	Lichen & Bryophytes
	U-234	$7.90\ 10^{-3}$		Lichen & Bryophytes
	Th-230	0.113		Lichen & Bryophytes
	Ra-226	0.740		Lichen & Bryophytes
	Pb-210	$3.24\ 10^{-3}$		Lichen & Bryophytes
Np-237		$1.74\ 10^{-3}$		Lichen & Bryophytes

Radionuclide	Daughter	Terrestrial ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
Pu-238		$1.34\ 10^{-8}$	$5.04\ 10^{-4}$	Lichen & Bryophytes
	U-234	$5.33\ 10^{-6}$		Lichen & Bryophytes
	Th-230	$6.54\ 10^{-5}$		Lichen & Bryophytes
	Ra-226	$4.31\ 10^{-4}$		Lichen & Bryophytes
	Pb-210	$1.89\ 10^{-6}$		Lichen & Bryophytes
Pu-239		$3.34\ 10^{-4}$	$3.47\ 10^{-4}$	Lichen & Bryophytes
	U-235	$1.01\ 10^{-6}$		Lichen & Bryophytes
	Pa-231	$1.22\ 10^{-5}$		Lichen & Bryophytes
Pu-240		$7.16\ 10^{-5}$	$7.16\ 10^{-5}$	Lichen & Bryophytes
	Th-232	$5.15\ 10^{-9}$		Lichen & Bryophytes
	Ra-228	$1.25\ 10^{-10}$		Lichen & Bryophytes
Pu-241		$2.76\ 10^{-15}$	$2.46\ 10^{-6}$	Lichen & Bryophytes
	Am-241	$1.56\ 10^{-7}$		Lichen & Bryophytes
	Np-237	$2.31\ 10^{-6}$		Lichen & Bryophytes
Am-241		$4.44\ 10^{-6}$	$7.39\ 10^{-5}$	Lichen & Bryophytes
	Np-237	$6.94\ 10^{-5}$		Lichen & Bryophytes
Cm-243		$4.19\ 10^{-9}$	$4.23\ 10^{-7}$	Lichen & Bryophytes
	Pu-239	$4.03\ 10^{-7}$		Lichen & Bryophytes
	U-235	$1.22\ 10^{-9}$		Lichen & Bryophytes
	Pa-231	$1.47\ 10^{-8}$		Lichen & Bryophytes
Cm-244		$6.41\ 10^{-10}$	$1.99\ 10^{-7}$	Lichen & Bryophytes
	Pu-240	$1.98\ 10^{-7}$		Lichen & Bryophytes
	Th-232	$1.42\ 10^{-11}$		Lichen & Bryophytes
	Ra-228	$3.44\ 10^{-13}$		Lichen & Bryophytes

1013. We note that for all radionuclides the ERICA risk quotients remain below 1. The maximum dose to Non-Human Biota is therefore below 10 µGy h^{-1}. Hence, terrestrial ecosystems are considered to be sufficiently protected.

E.6.5. ERICA results for a Freshwater Ecosystem

1014. For a freshwater ecosystem, an approach similar to the approach for the terrestrial ecosystem is taken.

1015. 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.

1016. 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.

1017. In addition, it is noted that not all radionuclides considered in the ENRMF are available in the ERICA assessment tool. Of the 12 'missing' radionuclides in the Tier 1 assessment, the radionuclides Sn-126, Ba-133, Pm-147, Eu-155 and Ac-227 represent a small fraction in the UK Radioactive Waste Inventory; Fe-55 has a short half-life (2.7 years); U-232, U-233 and U-234 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, Ag-108m, was the only one that was considered further.

1018. A Tier 2 assessment was undertaken to determine a limiting concentration in soil for Ag-108m. A conservative approach was taken to complete missing input parameters. The limiting concentration in a freshwater ecosystem (corresponding to 10 µGy h^{-1}) was found to be 26.3 Bq l^{-1} for water and 4.83 Bq kg^{-1} d.w. for sediment.

1019. Table 135 presents the corresponding limiting concentrations in water and sediment for the radionuclides considered in the assessment of the freshwater ecosystem.

Table 135 Radionuclide specific limiting environmental activity concentrations in the freshwater ecosystem derived from ERICA based on a generic screening level of 10 µGy h^{-1}

Radionuclide	Limiting concentrations in freshwater ecosystem (based on a generic screening level of 10 µGy h^{-1})			
	Water (Bq l^{-1})	Limiting organism	Sediment (Bq kg^{-1} d.w.)	Limiting organism
H-3	$3.94\ 10^{5}$	Mammal	$6.02\ 10^{4}$	Vascular plant
C-14	0.476	Benthic fish	0.862	Benthic fish
Cl-36	32.7	Benthic fish	2.66	Benthic fish
Co-60	0.0168	Insect larvae	7,040	Insect larvae
Ni-63	227	Insect larvae	$8.13\ 10^{5}$	Insect larvae
Sr-90	0.339	Mammal	265	Mammal
Nb-94	4.88	Mammal	76.9	Mammal
Tc-99	617	Reptile	847	Pelagic fish
Ru-106	0.133	Insect larvae	$1.22\ 10^{4}$	Insect larvae
Ag-108m	26.3	Mammal	4.83	Mollusc – bivalve
Sb-125	0.820	Insect larvae	$1.72\ 10^{4}$	Reptile
I-129	1.91	Insect larvae	$3.94\ 10^{4}$	Reptile
Cs-134	0.0205	Insect larvae	9,900	Insect larvae
Cs-137	0.0510	Insect larvae	$1.93\ 10^{4}$	Benthic fish
Eu-152	2.94	Mammal	1,030	Mammal
Eu-154	1.93	Mammal	680	Mammal
Pb-210	1.12	Insect larvae	5,000	Mollusc – bivalve
Ra-226	$9.09\ 10^{-4}$	Insect larvae	2.80	Insect larvae
Ra-228	0.250	Insect larvae	1,190	Mollusc – bivalve
Th-230	$9.52\ 10^{-4}$	Vascular plant	10.3	Vascular plant
Pa-231	$7.94\ 10^{-3}$	Mollusc – gastropod	48.1	Mollusc – bivalve

Radionuclide	Limiting concentrations in freshwater ecosystem (based on a generic screening level of 10 µGy h^{-1})			
	Water (Bq l^{-1})	Limiting organism	Sediment (Bq kg^{-1} d.w.)	Limiting organism
Th-232	1.12 10^{-3}	Vascular plant	12.0	Vascular plant
U-234	0.253	Vascular plant	12.3	Mollusc – gastropod
U-235	0.272	Vascular plant	13.4	Mollusc – gastropod
U-238	0.296	Vascular plant	14.4	Mollusc – gastropod
Np-237	0.457	Mollusc – bivalve	0.128	Mollusc – bivalve
Pu-238	0.0159	Mollusc – gastropod	606	Phytoplankton
Pu-239	0.0170	Mollusc – gastropod	645	Phytoplankton
Pu-240	0.0169	Mollusc – gastropod	645	Phytoplankton
Pu-241	62.5	Mollusc – gastropod	2.37 10^{6}	Phytoplankton
Am-241	9.62 10^{-3}	Mollusc – gastropod	207	Mollusc – bivalve
Cm-243	9.43 10^{-3}	Mollusc – gastropod	0.361	Mollusc – gastropod
Cm-244	9.52 10^{-3}	Mollusc – gastropod	0.364	Mollusc – bivalve

1020. Section 7.4.2 explains how the maximum inventory for each radionuclide has been derived from the different assessment scenarios. The result is summarised in paragraph 303: for some radionuclides, the maximum inventory is set by 89.6 TBq (based on disposal of 448,000 t at a maximum specific activity of 200 Bq g^{-1}); for the other radionuclides, at least one of the assessment scenarios was more limiting, as summarised in Table 25.

1021. Activity concentrations in surface water were not modelled explicitly as the nearest natural surface water course is Willow Brook, approximately 2.6 km to the south of the site. There are a few ponds and surface drains close to the site but these do not represent sustainable freshwater ecosystems supporting a diverse range of organisms.

1022. In the first instance a simplistic approach to representing the activity concentration in surface water has been adopted. The GoldSim modelled groundwater concentrations in the vicinity of the disposal site have been scaled in a conservative way to represent the concentrations in freshwater. River flows of 10^7 m^3 y^{-1} are modelled in SNIFFER to account for dilution of groundwater (SNIFFER, 2006). The recommended range, suggested by the IAEA for dilution of groundwater in surface water is between 10^6 m^3 y^{-1} for an arid region to 10^7 m^3 y^{-1} for a temperate region (IAEA, 2003). The flow in the aquifer calculated in the GoldSim model is 87,000 m^3 y^{-1}. A dilution factor of 10 is therefore conservative as it effectively assumes that the flow through Willow Brook is no more than 10^5 m^3 y^{-1} (i.e. below the lower flow rate presented by IAEA for generic studies in an arid region) and makes no allowance for retardation of radionuclides as the groundwater moves through 2.6 km to Willow Brook. In practice, it could take hundreds or thousands of years for many of the radionuclides to appear in Willow Brook, with associated reduced activity concentrations due to radioactive decay. The surface water 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 and the scaled values are also shown. The soil concentrations from the irrigation scenario have been scaled in the same way to calculate sediment concentrations (see Table 136). The 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 and the scaled values are also shown.

Table 136 Peak modelled radionuclide activity concentrations in freshwater environmental media

Radionuclide	Daughter	Activity Concentration in freshwater ecosystem scaled to the maximum inventory of each radionuclide in the ENRMF	
		Water (Bq l^{-1})	Sediment (Bq kg^{-1})
H-3		$2.39\ 10^{-2}$	$1.37\ 10^{-3}$
C-14		$8.03\ 10^{-3}$	0.460
Cl-36		0.697	0.129
Co-60		$2.76\ 10^{-10}$	0
Ni-63		$2.33\ 10^{-6}$	$2.54\ 10^{-5}$
Sr-90		$7.01\ 10^{-6}$	$1.90\ 10^{-5}$
Nb-94		$2.46\ 10^{-4}$	0.0235
Tc-99		0.601	0.0518
Ru-106		$2.22\ 10^{-10}$	0
Ag-108m		$3.07\ 10^{-4}$	$8.12\ 10^{-3}$
Sb-125		$3.76\ 10^{-9}$	0
I-129		$6.90\ 10^{-3}$	$4.24\ 10^{-3}$
Cs-134		$4.06\ 10^{-11}$	0
Cs-137		$2.79\ 10^{-8}$	$1.17\ 10^{-7}$
Eu-152		$7.99\ 10^{-10}$	$6.92\ 10^{-9}$
Eu-154		$2.67\ 10^{-10}$	0
Pb-210		$4.33\ 10^{-9}$	$1.46\ 10^{-8}$
Ra-226		$5.17\ 10^{-6}$	$6.03\ 10^{-4}$
	Pb-210	$7.94\ 10^{-6}$	$6.14\ 10^{-4}$
Ra-228		$4.55\ 10^{-11}$	0
Th-230		$2.04\ 10^{-5}$	0.321
	Ra-226	$3.69\ 10^{-5}$	0.0226
	Pb-210	$5.56\ 10^{-5}$	0.0223
Pa-231		$1.88\ 10^{-5}$	$5.85\ 10^{-3}$
Th-232		$5.28\ 10^{-5}$	0.894
	Ra-228	$6.13\ 10^{-5}$	0.891
U-234		$5.49\ 10^{-4}$	0.0125
	Th-230	$9.66\ 10^{-5}$	0.349
	Ra-226	$1.49\ 10^{-4}$	0.239
	Pb-210	$2.23\ 10^{-4}$	0.236
U-235		$5.58\ 10^{-4}$	0.0204
	Pa-231	$2.18\ 10^{-4}$	0.188
U-238		$2.87\ 10^{-3}$	0.105
	U-234	$7.07\ 10^{-4}$	0.0924
	Th-230	$8.17\ 10^{-5}$	2.99

Radionuclide	Daughter	Activity Concentration in freshwater ecosystem scaled to the maximum inventory of each radionuclide in the ENRMF	
		Water (Bq l^{-1})	Sediment (Bq kg^{-1})
	Ra-226	1.22 10^{-4}	2.05
	Pb-210	1.84 10^{-4}	2.02
Np-237		6.89 10^{-3}	0.0174
Pu-238		1.04 10^{-7}	1.01 10^{-6}
	U-234	2.74 10^{-6}	6.24 10^{-5}
	Th-230	4.82 10^{-7}	1.74 10^{-3}
	Ra-226	7.42 10^{-7}	1.19 10^{-3}
	Pb-210	1.11 10^{-6}	1.18 10^{-3}
Pu-239		8.79 10^{-5}	0.0267
	U-235	2.57 10^{-7}	1.27 10^{-5}
	Pa-231	8.61 10^{-8}	1.17 10^{-4}
Pu-240		2.38 10^{-5}	5.73 10^{-3}
	Th-232	2.84 10^{-12}	1.60 10^{-7}
	Ra-228	3.30 10^{-12}	1.60 10^{-7}
Pu-241		4.23 10^{-10}	8.10 10^{-10}
	Am-241	2.85 10^{-8}	1.34 10^{-6}
	Np-237	9.14 10^{-6}	2.31 10^{-5}
Am-241		8.13 10^{-7}	3.82 10^{-5}
	Np-237	2.75 10^{-4}	6.94 10^{-4}
Cm-243		8.35 10^{-9}	3.35 10^{-8}
	Pu-239	1.06 10^{-7}	3.23 10^{-5}
	U-235	3.10 10^{-10}	1.53 10^{-8}
	Pa-231	1.04 10^{-10}	1.41 10^{-7}
Cm-244		2.00 10^{-9}	5.13 10^{-9}
	Pu-240	6.59 10^{-8}	1.58 10^{-5}
	Th-232	7.84 10^{-15}	4.41 10^{-10}
	Ra-228	9.09 10^{-15}	4.40 10^{-10}

1023. The ERICA assessment tool is 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 screening dose rate of 10 µGy h^{-1}.

1024. Table 137 below summarises the results of the wildlife assessment.

Table 137 Radionuclide specific risk quotients for freshwater ecosystems for the most limiting organisms based on a generic screening level of 10 μGy h^{-1}

Radionuclide	Daughter	Freshwater ecosystem (based on a generic screening level of 10 μGy h^{-1})		
		Risk quotient radionuclide	Limiting organism	Risk quotient chain
H-3		$6.06\ 10^{-8}$	Mammal	
C-14		0.534	Benthic fish	
Cl-36		0.0483	Pelagic fish	
Co-60		$1.64\ 10^{-8}$	Insect larvae	
Ni-63		$1.03\ 10^{-8}$	Insect larvae	
Sr-90		$2.07\ 10^{-5}$	Mammal	
Nb-94		$3.05\ 10^{-5}$	Mammal	
Tc-99		$9.75\ 10^{-4}$	Reptile	
Ru-106		$1.67\ 10^{-9}$	Insect larvae	
Ag-108m		$1.68\ 10^{-3}$	Mollusc – bivalve	
Sb-125		$4.58\ 10^{-9}$	Insect larvae	
I-129		$3.61\ 10^{-3}$	Insect larvae	
Cs-134		$1.98\ 10^{-9}$	Insect larvae	
Cs-137		$5.48\ 10^{-7}$	Insect larvae	
Eu-152		$2.72\ 10^{-10}$	Mammal	
Eu-154		$1.38\ 10^{-10}$	Mammal	
Pb-210		$3.86\ 10^{-9}$	Insect larvae	
Ra-226		$5.69\ 10^{-3}$	Insect larvae	$5.69\ 10^{-3}$
	Pb-210	$7.09\ 10^{-6}$	Insect larvae	
Ra-228		$1.82\ 10^{-10}$	Insect larvae	
Th-230		0.0214	Vascular plant	0.0621
	Ra-226	0.0406	Insect larvae	
	Pb-210	$4.97\ 10^{-5}$	Insect larvae	
Pa-231		$2.37\ 10^{-3}$	Mollusc – gastropod	
Th-232		0.0745	Vascular plant	0.0752
	Ra-228	$7.49\ 10^{-4}$	Insect larvae	
U-234		$2.17\ 10^{-3}$	Vascular plant	0.267
	Th-230	0.101	Vascular plant	
	Ra-226	0.163	Insect larvae	
	Pb-210	$1.99\ 10^{-4}$	Insect larvae	
U-235		$2.05\ 10^{-3}$	Vascular plant	0.0295
	Pa-231	0.0274	Mollusc – gastropod	
U-238		$9.71\ 10^{-3}$	Vascular plant	1.04
	U-234	$7.52\ 10^{-3}$	Vascular plant	
	Th-230	0.291	Vascular plant	
	Ra-226	0.732	Insect larvae	
	Pb-210	$4.05\ 10^{-4}$	Insect larvae	
Np-237		0.136	Mollusc – bivalve	

Radionuclide	Daughter	Freshwater ecosystem (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Limiting organism	Risk quotient chain
Pu-238		$6.54\ 10^{-6}$	Mollusc – gastropod	$1.34\ 10^{-3}$
	U-234	$1.08\ 10^{-5}$	Vascular plant	
	Th-230	$5.06\ 10^{-4}$	Vascular plant	
	Ra-226	$8.16\ 10^{-4}$	Insect larvae	
	Pb-210	$9.94\ 10^{-7}$	Insect larvae	
Pu-239		$5.17\ 10^{-3}$	Mollusc – gastropod	$5.18\ 10^{-3}$
	U-235	$9.47\ 10^{-7}$	Vascular plant	
	Pa-231	$1.08\ 10^{-5}$	Mollusc – gastropod	
Pu-240		$1.41\ 10^{-3}$	Mollusc – gastropod	$1.41\ 10^{-3}$
	Th-232	$1.33\ 10^{-8}$	Vascular plant	
	Ra-228	$1.34\ 10^{-10}$	Insect larvae	
Pu-241		$6.76\ 10^{-12}$	Mollusc – gastropod	$1.83\ 10^{-4}$
	Am-241	$2.97\ 10^{-6}$	Mollusc – gastropod	
	Np-237	$1.80\ 10^{-4}$	Mollusc – bivalve	
Am-241		$8.46\ 10^{-5}$	Mollusc – gastropod	$5.51\ 10^{-3}$
	Np-237	$5.42\ 10^{-3}$	Mollusc – bivalve	
Cm-243		$8.86\ 10^{-7}$	Mollusc – gastropod	$7.15\ 10^{-6}$
	Pu-239	$6.25\ 10^{-6}$	Mollusc – gastropod	
	U-235	$1.14\ 10^{-9}$	Vascular plant	
	Pa-231	$1.31\ 10^{-8}$	Mollusc – gastropod	
Cm-244		$2.11\ 10^{-7}$	Mollusc – gastropod	$4.11\ 10^{-6}$
	Pu-240	$3.90\ 10^{-6}$	Mollusc – gastropod	
	Th-232	$3.68\ 10^{-11}$	Vascular plant	
	Ra-228	$3.70\ 10^{-13}$	Insect larvae	

1025. We note that the ERICA risk quotients exceed 1 only for U-238 (where the risk quotient is 1.04). The dose is mainly due to ingrowth of Th-230 and Ra-226. The maximum dose to Non-Human Biota would still remain below 40 µGy h^{-1}. Hence, freshwater ecosystems are considered to be sufficiently protected.

E.6.6. Tier 2 assessment for Burrowing Animals

1026. A minimum of 1 m soil will be put on top of the facility as a restoration layer. Below this soil, there are membranes, 30 cm of regulating material 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 2.6 m. The granular layer deters burrowing animals at least for a few hundred years, until it is naturally broken down and mixed with soil. 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. A Tier 2 ERICA assessment was therefore undertaken to estimate the potential dose to animals burrowing into the waste cells after closure. 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.

1027. While this assessment was focused on burrowing animals (rabbits), all the parameters (concentration factors, dose factors, etc.) were set to default ERICA values.

1028. Table 138 presents the limiting concentrations in soil and air for the radionuclides considered in the wildlife assessment of the terrestrial ecosystem focused on burrowing animals.These were calculated using the Tier 2 approach and are based on a screening level of 10 µGy h^{-1}.

Table 138 Radionuclide specific limiting environmental activity concentrations in the terrestrial ecosystem focused on burrowing animals derived from ERICA

Radionuclide	Limiting concentrations in terrestrial ecosystem focused on burrowing animals, based on the ERICA 10 µGy h^{-1} screening value		
	Soil (Bq kg^{-1} d.w.)	Air (Bq m^{-3})	Limiting organism
H-3		8,080	Mammal – small-burrowing
C-14		252	Mammal – small-burrowing
Cl-36	8,930		Mammal – small-burrowing
Co-60	8,110		Mammal – small-burrowing
Ni-63	$7.57\ 10^{6}$		Mammal – small-burrowing
Sr-90	9,680		Mammal – small-burrowing
Nb-94	$1.25\ 10^{4}$		Mammal – small-burrowing
Tc-99	$4.38\ 10^{5}$		Mammal – small-burrowing
Ru-106	$5.23\ 10^{4}$		Mammal – small-burrowing
Ag-108m	$1.14\ 10^{4}$		Mammal – small-burrowing
Sb-125	$4.75\ 10^{4}$		Mammal – small-burrowing
I-129	$4.20\ 10^{5}$		Mammal – small-burrowing
Cs-134	7,350		Mammal – small-burrowing
Cs-137	$1.16\ 10^{4}$		Mammal – small-burrowing
Eu-152	$1.80\ 10^{4}$		Mammal – small-burrowing
Eu-154	$1.64\ 10^{4}$		Mammal – small-burrowing
Pb-210	$1.02\ 10^{6}$		Mammal – small-burrowing
Ra-226	1,460		Mammal – small-burrowing
Ra-228	$2.06\ 10^{4}$		Mammal – small-burrowing
Th-230	$2.61\ 10^{6}$		Mammal – small-burrowing
Pa-231	643		Mammal – small-burrowing
Th-232	$3.09\ 10^{6}$		Mammal – small-burrowing
U-234	$6.52\ 10^{4}$		Mammal – small-burrowing
U-235	$4.90\ 10^{4}$		Mammal – small-burrowing
U-238	$7.61\ 10^{4}$		Mammal – small-burrowing
Np-237	673		Mammal – small-burrowing
Pu-238	$2.23\ 10^{4}$		Mammal – small-burrowing

Radionuclide	Limiting concentrations in terrestrial ecosystem focused on burrowing animals, based on the ERICA 10 µGy h^{-1} screening value		
	Soil (Bq kg^{-1} d.w.)	Air (Bq m^{-3})	Limiting organism
Pu-239	2.38 10^{4}		Mammal – small-burrowing
Pu-240	2.38 10^{4}		Mammal – small-burrowing
Pu-241	8.82 10^{7}		Mammal – small-burrowing
Am-241	1.30 10^{4}		Mammal – small-burrowing
Cm-243	558		Mammal – small-burrowing
Cm-244	560		Mammal – small-burrowing

1029. 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 maximum inventory, see Section 7.4.2.2 and Table 25. Activity concentrations of radionuclides that were ingrown through radioactive decay were calculated separately and are also shown. The results are given in Table 139.

Table 139 Peak modelled radionuclide activity concentrations in terrestrial environmental media

Radionuclide	Daughter	Activity Concentration in the waste cells, scaled to the radiological capacity of the ENRMF	
		Soil (Bq kg^{-1})	Air (Bq m^{-3})
H-3			1.67 10^{-4}
C-14			4.82 10^{-3}
Cl-36		7950	
Co-60		20.6	
Ni-63		3.20 10^{4}	
Sr-90		1.15 10^{4}	
Nb-94		4.84 10^{4}	
Tc-99		4.79 10^{4}	
Ru-106		3.47 10^{-7}	
Ag-108m		4.39 10^{4}	
Sb-125		0.0217	
I-129		22.5	
Cs-134		1.94 10^{-4}	
Cs-137		1.23 10^{4}	
Eu-152		2,290	
Eu-154		402	
Pb-210		7,500	
Ra-226*		1,000	
	Pb-210	853	
Ra-228		39.0	
Th-230		3.75 10^{4}	
	Ra-226	962	

	Pb-210	530	
Pa-231		$1.01\ 10^{4}$	
Th-232		$3.88\ 10^{4}$	
	Ra-228	$3.87\ 10^{4}$	
U-234		3,470	
	Th-230	1.91	
	Ra-226	0.0248	
	Pb-210	0.0103	
U-235		2,660	
	Pa-231	3.38	
U-238		$1.37\ 10^{4}$	
	U-234	2.32	
	Th-230	$6.43\ 10^{-4}$	
	Ra-226	$5.58\ 10^{-6}$	
	Pb-210	$1.89\ 10^{-6}$	
Np-237		244	
Pu-238		$3.02\ 10^{4}$	
	U-234	6.54	
	Th-230	$1.95\ 10^{-3}$	
	Ra-226	$1.76\ 10^{-5}$	
	Pb-210	$6.07\ 10^{-6}$	
Pu-239		$4.84\ 10^{4}$	
	U-235	$2.86\ 10^{-3}$	
	Pa-231	$1.82\ 10^{-6}$	
Pu-240		$4.82\ 10^{4}$	
	Th-232	$1.28\ 10^{-10}$	
	Ra-228	$9.74\ 10^{-11}$	
Pu-241		2,720	
	Am-241	1,420	
	Np-237	0.0203	
Am-241		$4.41\ 10^{4}$	
	Np-237	0.896	
Cm-243		$1.17\ 10^{4}$	
	Pu-239	44.4	
	U-235	$1.62\ 10^{-6}$	
	Pa-231	$7.64\ 10^{-10}$	
Cm-244		4,930	
	Pu-240	120	
	Th-232	$1.66\ 10^{-13}$	
	Ra-228	$1.19\ 10^{-13}$	

* Ra-226 is allowed in the top 5 m with an activity of up to 5 Bq g^{-1}. This level leads to an activity concentration in the soil of approximately 1000 Bq kg^{-1}.

1030. The ERICA assessment tool is 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 screening dose rate of 10 µGy h^{-1}.

1031. Table 140 below summarises the results of the wildlife assessment for burrowing animals in the waste cells.

Table 140 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	Daughter	Terrestrial ecosystem, burrowing animals in the waste cells (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
H-3		$2.07\ 10^{-8}$		Mammal – small-burrowing
C-14		$1.91\ 10^{-5}$		Mammal – small-burrowing
Cl-36		0.0890		Mammal – small-burrowing
Co-60		$2.54\ 10^{-3}$		Mammal – small-burrowing
Ni-63		$4.23\ 10^{-3}$		Mammal – small-burrowing
Sr-90		1.19		Mammal – small-burrowing
Nb-94		3.86		Mammal – small-burrowing
Tc-99		0.109		Mammal – small-burrowing
Ru-106		$6.64\ 10^{-12}$		Mammal – small-burrowing
Ag-108m		3.86		Mammal – small-burrowing
Sb-125		$4.57\ 10^{-7}$		Mammal – small-burrowing
I-129		$5.35\ 10^{-5}$		Mammal – small-burrowing
Cs-134		$2.63\ 10^{-8}$		Mammal – small-burrowing
Cs-137		1.06		Mammal – small-burrowing
Eu-152		0.127		Mammal – small-burrowing
Eu-154		0.0245		Mammal – small-burrowing
Pb-210		$7.39\ 10^{-3}$		Mammal – small-burrowing
Ra-226		0.683	0.684	Mammal – small-burrowing
	Pb-210	$8.40\ 10^{-4}$		Mammal – small-burrowing
Ra-228		$1.89\ 10^{-3}$		Mammal – small-burrowing
Th-230		0.0144	0.672	Mammal – small-burrowing
	Ra-226	0.657		Mammal – small-burrowing
	Pb-210	$5.21\ 10^{-4}$		Mammal – small-burrowing
Pa-231		15.6		Mammal – small-burrowing
Th-232		0.0125	1.89	Mammal – small-burrowing
	Ra-228	1.88		Mammal – small-burrowing
U-234		0.0532	0.0532	Mammal – small-burrowing
	Th-230	$7.34\ 10^{-7}$		Mammal – small-burrowing
	Ra-226	$1.69\ 10^{-5}$		Mammal – small-burrowing
	Pb-210	$1.02\ 10^{-8}$		Mammal – small-burrowing
U-235		0.0543	0.0596	Mammal – small-burrowing
	Pa-231	$5.26\ 10^{-3}$		Mammal – small-burrowing

Radionuclide	Daughter	Terrestrial ecosystem, burrowing animals in the waste cells (based on a generic screening level of 10 µGy h^{-1})		
		Risk quotient radionuclide	Risk quotient chain	Limiting organism
U-238		0.180	0.180	Mammal – small-burrowing
	U-234	3.56 10^{-5}		Mammal – small-burrowing
	Th-230	2.47 10^{-10}		Mammal – small-burrowing
	Ra-226	3.81 10^{-9}		Mammal – small-burrowing
	Pb-210	1.86 10^{-12}		Mammal – small-burrowing
Np-237		0.363		Mammal – small-burrowing
Pu-238		1.35	1.35	Mammal – small-burrowing
	U-234	1.00 10^{-4}		Mammal – small-burrowing
	Th-230	7.49 10^{-10}		Mammal – small-burrowing
	Ra-226	1.20 10^{-8}		Mammal – small-burrowing
	Pb-210	5.97 10^{-12}		Mammal – small-burrowing
Pu-239		2.03	2.03	Mammal – small-burrowing
	U-235	5.84 10^{-8}		Mammal – small-burrowing
	Pa-231	2.84 10^{-9}		Mammal – small-burrowing
Pu-240		2.02	2.02	Mammal – small-burrowing
	Th-232	4.14 10^{-17}		Mammal – small-burrowing
	Ra-228	4.73 10^{-15}		Mammal – small-burrowing
Pu-241		3.08 10^{-5}	0.109	Mammal – small-burrowing
	Am-241	0.109		Mammal – small-burrowing
	Np-237	3.01 10^{-5}		Mammal – small-burrowing
Am-241		3.38	3.38	Mammal – small-burrowing
	Np-237	1.33 10^{-3}		Mammal – small-burrowing
Cm-243		20.9	20.9	Mammal – small-burrowing
	Pu-239	1.87 10^{-3}		Mammal – small-burrowing
	U-235	3.30 10^{-11}		Mammal – small-burrowing
	Pa-231	1.19 10^{-12}		Mammal – small-burrowing
Cm-244		8.80	8.80	Mammal – small-burrowing
	Pu-240	5.03 10^{-3}		Mammal – small-burrowing
	Th-232	5.39 10^{-20}		Mammal – small-burrowing
	Ra-228	5.76 10^{-18}		Mammal – small-burrowing

1032. We note that for Sr-90, Nb-94, Ag-108m, Cs-137, Th-232, Pu-238, Pu-239, Pu-240 and Am-241 the ERICA risk quotients exceed 1 but are less than 4. In these cases the maximum dose to Non-Human Biota would still remain below 40 µGy h^{-1}. Hence, terrestrial ecosystems are considered to be sufficiently protected.

1033. In addition, for Pa-231, Cm-243 and Cm-244 we observe an ERICA risk quotient in excess of 4, which means that burrowing animals in the waste cells potentially receive a dose above 40 µGy h^{-1}. The maximum dose rate to the burrowing animal, whilst it is in the waste material, is estimated to be 210 µGy h^{-1}. The dose rate in the cap material and layer of non-radioactive waste would be zero. The only burrowing animals that could burrow into the waste itself are rabbits burrowing at the maximum warren depth as other burrowing animals have shallower burrows that would not reach the waste.

E.6.7. Discussion

1034. In many cases the risk quotient indicates that the modelled environmental activity concentration is some orders of magnitude below the limiting activity concentration.

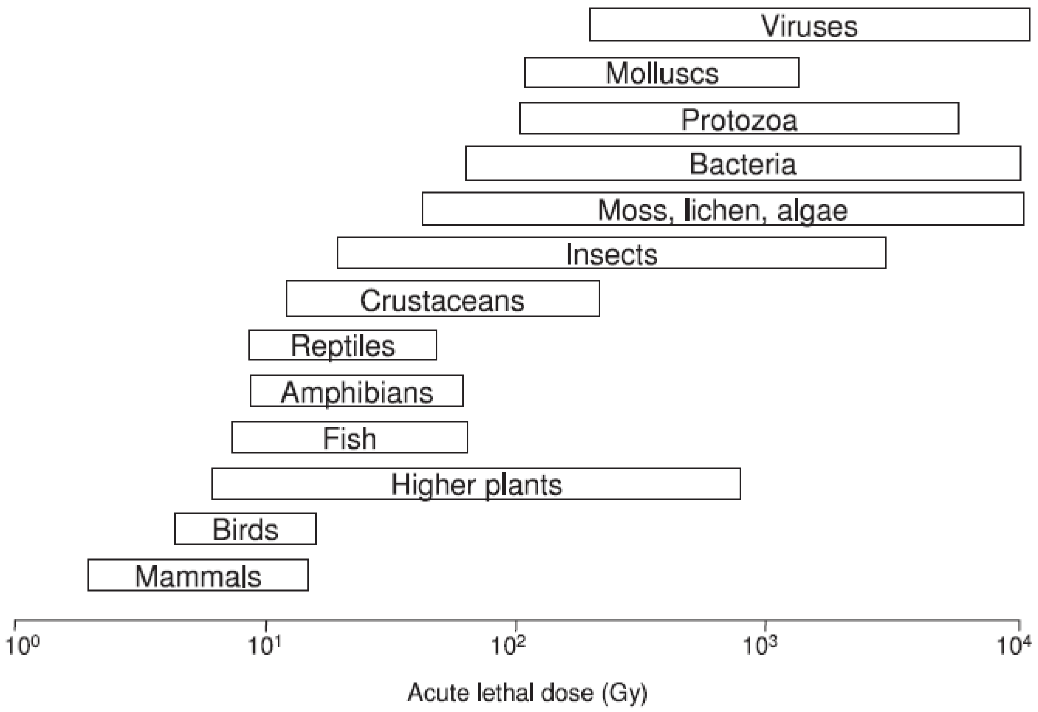
1035. For U-238 the derived risk quotient is exceeded by a factor between 1 and 4 and the most limiting organism type is identified as "Vascular plant". The Tier 1 risk quotient includes an uncertainty factor of 3 which applied to the estimated dose rate. Hence the expected value of the dose rate lies between 3.3 µGy h^{-1} and 13.3 µGy h^{-1}.

1036. Given the extreme conservatism of the derivation of the activity concentrations it is considered that vascular plants remain adequately protected and certainly the implied dose rates remain below the Environment Agency regulatory action level of 40 µGy h^{-1}.

1037. Within the ERICA assessment tool, it is possible to progress to Tier 2 or Tier 3 assessments enabling the use of site-specific data across a range of parameters and allowing a more refined interpretation of results. A Tier 2 assessment was performed for burrowing animals that enter the waste. Where dose rates to non-human biota exceed 10 µGy h^{-1} based on a more refined Tier 2 or Tier 3 assessment, it does not automatically follow that site measures are failing to protect the environment.

1038. Different organism types have different levels of radiosensitivity. This general statement is broadly supported by Figure 28, which is taken from (UNSCEAR, 1996) and relates to acute lethal doses. The general conclusion is that vertebrates are more radiosensitive than invertebrates, considered as a whole, and similarly higher plants are more radiosensitive than other plant species, again considered as a whole.

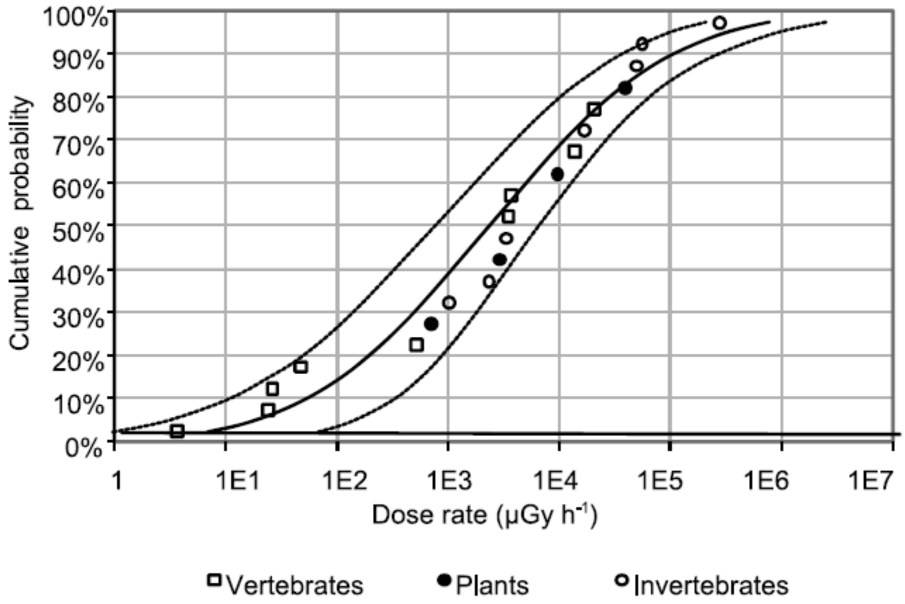
Figure 28. Approximate acute lethal dose ranges for various taxonomic groups



Reproduced from (UNSCEAR, 1996)*; based on* (Sparrow, et al., 1967)*. and* (Whicker & Schultz, 1982)

1039. Andersson et al. undertook a more refined assessment of the FREDERICA database to suggest potential screening levels linked to very broad organism groupings (Andersson, et al., 2009), but these have not been widely adopted and have not been applied in this study. However, further analysis of the data (Garnier-Laplace, et al., 2010) within the FREDERICA database demonstrated broad evidence that radiosensitivity is linked to organism type, as represented in the Figure 29.

Figure 29. Species sensitivity distribution based on all available relevant EDR_{10} values within the FREDERICA database



EDR_{10} is the dose rate giving rise to a 10% effect in the exposed group in comparison to the control group and HDR_5 (as noted previously) is the dose rate at which 95% of species will not experience more than a 10% effect. It will be noted that whereas values of HDR_5 for vertebrates (primarily mammals) range down to less than 10 µGy h^{-1}, the lowest value for invertebrates is 1000 µGy h^{-1}. Thus, invertebrates (including insects) are much less radiosensitive than vertebrates. For plants, values of HDR_5 lie in the range around 1000 to 15,000 µGy h^{-1}. The presentation does not allow a clear distinction between 'lower' and 'higher' plants, but it is likely that vascular angiosperms lie toward the lower end of the HDR_5 values and algae and phytoplankton lie towards the upper end.

1040. Given the design of the landfill facility and the design of the cap, it seems very unlikely that burrowing animals (rabbits) will build their warren in the disposed waste. A typical warren would not extend deep enough to penetrate the 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).

1041. We also note that within the regulatory framework the site operator has the obligation to protect a species rather than individual animals.

1042. 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 HDR_{50} (or even the EDR_{50}). Other burrowing animals such as mice, voles and moles have burrows up to 70 cm deep (molecatchers, n.d.) and will therefore not enter the waste. Hence they are protected.

1043. The dose rates to the rabbits burrowing into a deep warren that intercepts the waste cells could be reduced to below 40 µGy h^{-1} by applying a reduction factor to the radiological capacity for Pa-231, Cm-243 and Cm-244, if required. These radiological capacity reduction factors are summarised in Table 141.

Table 141 Suggested Radiological Capacity Reduction factor to reduce the potential dose to burrowing rabbits to a value below 40 µGy h^{-1}

Radionuclide	Radiological Capacity Reduction factor
Pa-231	4
Cm-243	6
Cm-244	3

E.7. Management of uncertainty

1044. 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.

1045. 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. Comparisons are undertaken for two scenarios using the current and previous approaches. These consider the borehole excavation and groundwater pathway using the SNIFFER models. A comparison is also presented between doses associated with contaminated water calculated by the HPA and the boundary well groundwater activity concentrations calculated using the ESC model.

1046. 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.

1047. Parameter value uncertainties have been considered in terms of the sensitivity of the limiting dose assessments to parameter selection. This is summarised in Section E.7.2. The uncertainty due to waste heterogeneity and age dependent doses to members of the public are discussed in Section E.7.3. Key findings are summarised in Section E.7.4.

E.7.1. Conceptual model uncertainty

Borehole Excavation

1048. The dose to workers was calculated using the same input parameters from the previous ESC (Augean, 2009a). The output from the previous ESC and the current model agreed except in the case of four radionuclides that are shown in Table 142. It is assumed that the differences in the treatment of daughter ingrowth accounts for the differences for these radionuclides. The current ESC calculates ingrowth explicitly.

Table 142 Comparison between 2009 ESC dose rates and those produced in the current model using equivalent input parameters

Radionuclide	Dose rate (µSv MBq-1)		
	Current model, 2009 parameters	2009 ESC model and parameters	Ratio Current/previous*
Pa-231	$1.73\ 10^{-4}$	$6.82\ 10^{-5}$	2.53
Pu-241	$9.27\ 10^{-7}$	$1.47\ 10^{-5}$	$6.30\ 10^{-2}$
Cm-243	$2.92\ 10^{-6}$	$3.03\ 10^{-6}$	$9.63\ 10^{-1}$
Cm-244	$7.29\ 10^{-7}$	$1.57\ 10^{-6}$	$4.65\ 10^{-1}$

* The ratio for all other radionuclides considered was 1.0

Groundwater

1049. A comparison of output from the SNIFFER groundwater model (Augean, 2009a) and the Goldsim model (used in this ESC) is presented in Table 143. This shows all radionuclides where the groundwater pathway, from either model, limits the radiological capacity of the ENRMF. All doses relate to the groundwater pathways. The capacity constraint listed for the SNIFFER groundwater model (Column 3) always relates to that pathway. This is the case for Sn-126 where the existing ENRMF disposal limit is derived from intrusion at 60 years. The capacity constraint shown for the revised ESC (Column 6) is the radiological capacity for the ENRMF based on the scenario shown in the table.

1050. The dose (µSv y^{-1} MBq^{-1}) from groundwater at the boundary in the 2015 ESC is similar or greater than that calculated using SNIFFER in the case of Cl-36, Tc-99, I-129, U-233, U-234, U-235 and U-238. This leads to a lower radiological capacity in the 2015 ESC for Cl-36, I-129, U-233, U-234, U-235 and U-238. The radiological capacity of Tc-99 in the 2015 ESC is further constrained by the small holding scenario, and the limit shown (90.7 TBq) relates to that scenario.

1051. The dose (µSv y-1 MBq-1) from groundwater at the boundary in the 2015 ESC is lower than that calculated using SNIFFER for all other radionuclides. Those where the difference is about a factor of 10 or less include Sn-126, Th-230, U-236 and Np-237. In each case the radiological capacity in the 2015 ESC is greater; noting that in the case of Th-230 the smallholder scenario limits the radiological capacity. For the remaining radionuclides the 2015 ESC calculates a lower dose for the groundwater pathway by a factor of >20 for Cm-244; by a factor of 30 to 40 for C-14, Th-229, Th-232, Pa-231, and Pu-242; by a factor of 208 for Pu-239; and, by a factor of 617 for Pu-240. In all these cases the radiological capacity is greater in the 2015 ESC, and with the exception of Pu-242, the radiological capacity is limited by other scenarios.

1052. The differences between these models are not therefore straight forward. Most differences are for radionuclides where ingrowth of daughter radionuclides needs to be calculated and it is noted that this is not dealt with in SNIFFER and requires side calculations to adjust the dose coefficients applied based on the transport of the parent radionuclide. Goldsim deals with the ingrowth and transport of the daughter products explicitly. For the other 5 radionuclides (that do not have daughter products), the groundwater dose rate is either greater in the 2015 ESC and/or the radiological capacity is further constrained by another pathway. The exception is Sn-126 which is now constrained by the groundwater pathway and has a higher radiological capacity.

Table 143 Comparison of Groundwater model outputs

Radionuclide	SNIFFER Groundwater model			Revised Environmental Safety Case		
	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Capacity (TBq)	Limiting scenario	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Radiological Capacity (TBq)	Limiting scenario
C-14	$1.39\ 10^{-7}$	$1.44\ 10^{2}$	Drinking water 100m from site	$3.49\ 10^{-9}$	$1.20\ 10^{2}$	Recreational (0 years)
Cl-36	$1.69\ 10^{-6}$	$1.19\ 10^{1}$	Drinking water 100m from site	$1.35\ 10^{-5}$	1.48	Well at boundary (All pathways)
Tc-99	$1.52\ 10^{-7}$	$1.31\ 10^{2}$	Drinking water 100m from site	$1.26\ 10^{-7}$	$9.07\ 10^{1}$	Small holding 200 years
Sn-126	$4.87\ 10^{-7}$	$4.10\ 10^{1}$*	Intruder	$9.10\ 10^{-8}$	$2.20\ 10^{2}$	Well at boundary (All pathways)
I-129	$3.02\ 10^{-4}$	$6.63\ 10^{-2}$	Drinking water 100m from site	$4.80\ 10^{-4}$	$4.17\ 10^{-2}$	Well at boundary (All pathways)
Th-229	$1.29\ 10^{-6}$	$1.55\ 10^{1}$	Drinking water 100m from site	$3.68\ 10^{-8}$	$2.98\ 10^{2}$	Excavator (Borehole) 60 years
Th-230	$7.35\ 10^{-7}$	$2.72\ 10^{1}$	Drinking water 100m from site	$6.94\ 10^{-8}$	$6.93\ 10^{1}$	Small holding 200 years
Th-232	$3.59\ 10^{-6}$	5.57	Drinking water 100m from site	$1.23\ 10^{-7}$	$7.16\ 10^{1}$	Small holding 200 years
Pa-231	$3.08\ 10^{-6}$	6.49	Drinking water 100m from site	$9.02\ 10^{-8}$	$1.86\ 10^{1}$	Small holding 200 years
U-233	$4.13\ 10^{-7}$	$4.84\ 10^{1}$	Drinking water 100m from site	$6.38\ 10^{-7}$	$3.13\ 10^{1}$	Well at boundary (All pathways)
U-234	$3.89\ 10^{-7}$	$5.14\ 10^{1}$	Drinking water 100m from site	$3.12\ 10^{-6}$	6.41	Well at boundary (All pathways)
U-235	$3.87\ 10^{-7}$	$5.17\ 10^{1}$	Drinking water 100m from site	$4.07\ 10^{-6}$	4.92	Well at boundary (All pathways)
U-236	$3.78\ 10^{-7}$	$5.30\ 10^{1}$	Drinking water 100m from site	$1.39\ 10^{-7}$	$1.44\ 10^{2}$	Well at boundary (All pathways)
U-238	$3.89\ 10^{-7}$	$5.14\ 10^{1}$	Drinking water 100m from site	$7.89\ 10^{-7}$	$2.53\ 10^{1}$	Well at boundary (All pathways)
Np-237	$9.52\ 10^{-5}$	$2.10\ 10^{-1}$	Drinking water 100m from site	$4.43\ 10^{-5}$	$4.52\ 10^{-1}$	Well at boundary (All pathways)
Pu-239	$1.37\ 10^{-6}$	$1.46\ 10^{1}$	Drinking water 100m from site	$6.62\ 10^{-9}$	$8.01\ 10^{2}$	Excavator (Borehole) 60 years
Pu-240	$9.33\ 10^{-7}$	$2.14\ 10^{1}$	Drinking water 100m from site	$1.51\ 10^{-9}$	$8.05\ 10^{2}$	Excavator (Borehole) 60 years
Pu-242	$1.51\ 10^{-6}$	$1.33\ 10^{1}$	Drinking water 100m from site	$4.06\ 10^{-8}$	$4.93\ 10^{2}$	Well at boundary (All pathways)
Cm-244	$1.15\ 10^{-7}$	$1.74\ 10^{2}$	Drinking water 100m from site	$<5.6\ 10^{-9}$	$1.59\ 10^{4}$	Excavator (Borehole) 60 years

*In this case the capacity value relates to the groundwater pathway.

HPA dose calculations

1053. A study by HPA (Ewers & Mobbs, 2010) considered the doses associated with aqueous liquid activity concentrations and gives activity concentrations that correspond to a dose of 20 µSv y^{-1}. These are presented in Table 144 alongside the boundary well groundwater activity concentrations expected when the maximum inventory is disposed. The radionuclides where the groundwater pathway is limiting are marked (*) and these are the radionuclides where the associated dose would be 20 µSv y^{-1} according to the model used for the ESC.

1054. The HPA activity concentration that gives rise to 20 µSv y^{-1} is always greater than the activity concentration calculated for groundwater when radionuclide disposal is at the maximum inventory. Thus, applying the HPA model to the boundary well groundwater activity concentrations would also give rise to doses below 20 µSv y^{-1}. This provides confidence that the ESC dose model is not underestimating the dose and that these groundwater concentrations will give rise to doses substantially lower than 20 µSv y^{-1}.

1055. Whilst the predicted activity concentration for many of these radionuclides is below the limit of detection the comparison provides reassurance for the following reasons:

- The limit of detection for H-3 is 4 Bq l^{-1} and this mobile radionuclide will be one of the first to appear in the environment and indicate at an early stage if the barriers are not behaving as expected. The dose from 4 Bq l^{-1} would be about 0.08 µSv y^{-1} and is a very small fraction of the dose from background radiation.
- The approach to calculating the maximum inventory and application of the sum of fractions ensures that the potential activity concentrations in groundwater are in most cases many orders of magnitude lower than the activity concentration that could result in a dose of 20 µSv y^{-1}.
- The groundwater activity concentrations are based on peak values, occurring over a timescale of 5 to 100,000 years. The resulting doses are treated as occurring to the same individual whereas only 15 of the shorter lived radionuclides peak within a period of 100 years.

Table 144 Activity concentration in aqueous liquid giving 20 µSv y^{-1} dose and arising in groundwater from disposal of the maximum inventory

Radionuclide	Activity concentration leading to 20 µSv y^{-1} (Bq l^{-1})	Activity concentration from disposal at maximum inventory (Bq l^{-1})
H-3	$1.00\ 10^{3}$	$2.39\ 10^{-1}$
C-14	$2.00\ 10^{-1}$	$8.03\ 10^{-2}$
Cl-36*	$1.04\ 10^{1}$	6.95
Fe-55	4.60	$2.27\ 10^{-9}$
Co-60	$1.00\ 10^{-2}$	$2.76\ 10^{-9}$
Ni-63	$7.60\ 10^{1}$	$2.33\ 10^{-5}$
Sr-90	$4.00\ 10^{-1}$	$7.01\ 10^{-5}$
Nb-94	$3.40\ 10^{-1}$	$2.46\ 10^{-3}$

Radionuclide	Activity concentration leading to 20 µSv y^{-1} (Bq l^{-1})	Activity concentration from disposal at maximum inventory (Bq l^{-1})
Tc-99	$1.48\ 10^{1}$	$6.01\ 10^{0}$
Ru-106	$2.00\ 10^{-1}$	$2.22\ 10^{-9}$
Ag-108m	$2.00\ 10^{-1}$	$3.07\ 10^{-3}$
Sb-125	1.20	$3.76\ 10^{-8}$
Sn-126*	nd	$2.03\ 10^{-1}$
I-129*	$1.00\ 10^{-1}$	$6.91\ 10^{-2}$
Ba-133	nd	$4.75\ 10^{-5}$
Cs-134	$2.00\ 10^{-2}$	$4.06\ 10^{-10}$
Cs-137	$4.00\ 10^{-2}$	$2.79\ 10^{-7}$
Pm-147	$4.60\ 10^{1}$	$1.17\ 10^{-9}$
Eu-152	$8.60\ 10^{-3}$	$7.99\ 10^{-9}$
Eu-154	$8.00\ 10^{-3}$	$2.67\ 10^{-9}$
Eu-155	$1.72\ 10^{-1}$	$7.08\ 10^{-10}$
Pb-210*	$6.00\ 10^{-3}$	$4.33\ 10^{-8}$
Ra-226	$4.00\ 10^{-2}$	$5.17\ 10^{-5}$
Ra-228	$8.80\ 10^{-3}$	$4.55\ 10^{-10}$
Ac-227	$1.20\ 10^{-1}$	$2.40\ 10^{-8}$
Th-229	$3.40\ 10^{-2}$	$2.56\ 10^{-5}$
Th-230	1.74	$2.04\ 10^{-4}$
Th-232	1.94	$5.28\ 10^{-4}$
Pa-231	$4.00\ 10^{-2}$	$1.88\ 10^{-4}$
U-232	$7.60\ 10^{-2}$	$1.24\ 10^{-5}$
U-233*	$5.00\ 10^{-1}$	$2.30\ 10^{-2}$
U-234*	$4.00\ 10^{-1}$	$5.49\ 10^{-3}$
U-235*	$4.00\ 10^{-1}$	$5.58\ 10^{-3}$
U-236*	$5.40\ 10^{-1}$	$1.01\ 10^{-1}$
U-238*	$4.00\ 10^{-1}$	$2.88\ 10^{-2}$
Np-237*	$1.50\ 10^{-1}$	$6.88\ 10^{-2}$
Pu-238	$2.00\ 10^{-1}$	$1.04\ 10^{-6}$
Pu-239	$2.00\ 10^{-1}$	$8.79\ 10^{-4}$
Pu-240	$2.00\ 10^{-1}$	$2.38\ 10^{-4}$
Pu-241	8.00	$4.23\ 10^{-9}$
Pu-242*	$2.00\ 10^{-1}$	$5.62\ 10^{-3}$
Am-241	$2.00\ 10^{-1}$	$8.13\ 10^{-6}$
Cm-243	$1.00\ 10^{-1}$	$8.35\ 10^{-8}$
Cm-244	$1.20\ 10^{-1}$	$2.00\ 10^{-8}$

* Groundwater pathway limits radiological capacity

E.7.2. Parameter sensitivity

1056. 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.

1057. 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).

1058. 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.

Gas models - C-14 and H-3 gas release rates

1059. The hazardous waste acceptance criteria at the ENRMF include a restriction on the amount of organic carbon that is disposed (6% of total carbon). 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 CFA permits LLW to contain a greater amount of organic carbon subject to the overall site limit and this is considered below.

1060. The release rate is expected to vary with time and the rate of gas production within the landfill has been simulated using the GasSim model (Augean, 2010). This used a medium and slow rate for carbon-based gas generation, applied to 2% and 4%, respectively. The gas generation curve shows a rapid build-up in the rate of release after capping followed by an exponential decline.

1061. The peak annual gas yield for carbon is less than 10% of the total quantity of gas. The average timescale of gas generation was set at 10 years during the period of operation. 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. The waste cells are capped sequentially so a series of peaks during the operational period would be expected.

1062. The release rate of gases from a landfill varies over time. A longer timescale for gas generation (20 years) has been applied to the period after closure using the value recommended by IAEA (IAEA, 2003).

1063. The GasSim model (Augean, 2010) shows 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. The average timescale for gas generation has therefore been set to 600 for the Residential occupant scenario (90/0.15), and 900 (approximately 140/0.15) for the Smallholding scenario.

1064. The recreational scenario limits the H-3 and C-14 disposals at the site. The gas release models are simple and very cautious. Any gas generated in the waste cells will be collected and discharged to air with greater mixing than modelled in the ESC. None of the waste disposed at the ENRMF will be putrescible and most organic carbon is expected to be plastic, materials or laboratory papers. These are expected to degrade very slowly and the peak release rate used for the ESC is not expected to occur.

1065. The dose varies in proportion to the average timescale for gas production and is illustrated in Table 145. Increasing the proportion of organic carbon in LLW would increase the amount of carbon available for degradation and result in a proportional increase in dose rate. The dose from C-14 using an average timescale of 10 years at the maximum inventory would be 30 µSv y^{-1}.

Table 145 Dose from releases to atmosphere: C-14 sensitivity to average timescale for gas production.

Average timescale for gas production (y)	Release rate (Bq y^{-1} MBq^{-1})	Air activity (Bq m^{-3} MBq^{-1})	Dose rate (µSv MBq^{-1})
1	$6.00\ 10^{4}$	$7.67\ 10^{-7}$	$3.33\ 10^{-6}$
5	$1.20\ 10^{4}$	$1.53\ 10^{-7}$	$6.67\ 10^{-7}$
10	$6.00\ 10^{3}$	$7.67\ 10^{-8}$	$3.33\ 10^{-7}$
20	$3.00\ 10^{3}$	$3.83\ 10^{-8}$	$1.67\ 10^{-7}$
50	$1.20\ 10^{3}$	$1.53\ 10^{-8}$	$6.67\ 10^{-8}$
100	$6.00\ 10^{2}$	$7.67\ 10^{-9}$	$3.33\ 10^{-8}$
200	$3.00\ 10^{2}$	$3.83\ 10^{-9}$	$1.67\ 10^{-8}$
600	$1.00\ 10^{2}$	$1.28\ 10^{-9}$	$5.56\ 10^{-9}$

Goldsim model

1066. The sensitivity of groundwater model outputs to changes in parameter assumptions is illustrated using the dose rate from drinking water. The impact of the total timeperiod assumed for the assessment was considered. The impact of changes to parameter values also considered barrier hydraulic conductivity with depth of clay, leachate head and cap efficiency. Reducing the cap efficiency by 5% had no impact on the groundwater drinking water pathway dose rate and is not discussed further.

1067. The dose rates from groundwater vary over time, generally rising to a peak and then reducing again (see Figure 33 in Appendix F for an example). Hence, the peak dose rate will depend on the total time period considered in the assessment, or on the specific subdivision of the time period that is being considered. The time of the peak dose is given in Table 78. If the peak has not occurred by the end of the assessment period (100,000 years) then the table records the time of the peak dose as 100,000 years. This is the case for ten radionuclides including Sn-126, U-238 and Th-232. Increasing the assessment period to 1,000,000 years means that the peak dose rates are realised for all radionuclides except U-238 and Th-232. However, the dose rate for these two radionuclides is increasing slowly at 1,000,000 years, as shown in Figure 30. The time of the peak dose rate and the ratio between the peak dose rate and the peak dose rate up to 100,000 years is given in Table 146.

Figure 30. Drinking water pathway dose rate increase for U-238 and Th-232 over time.

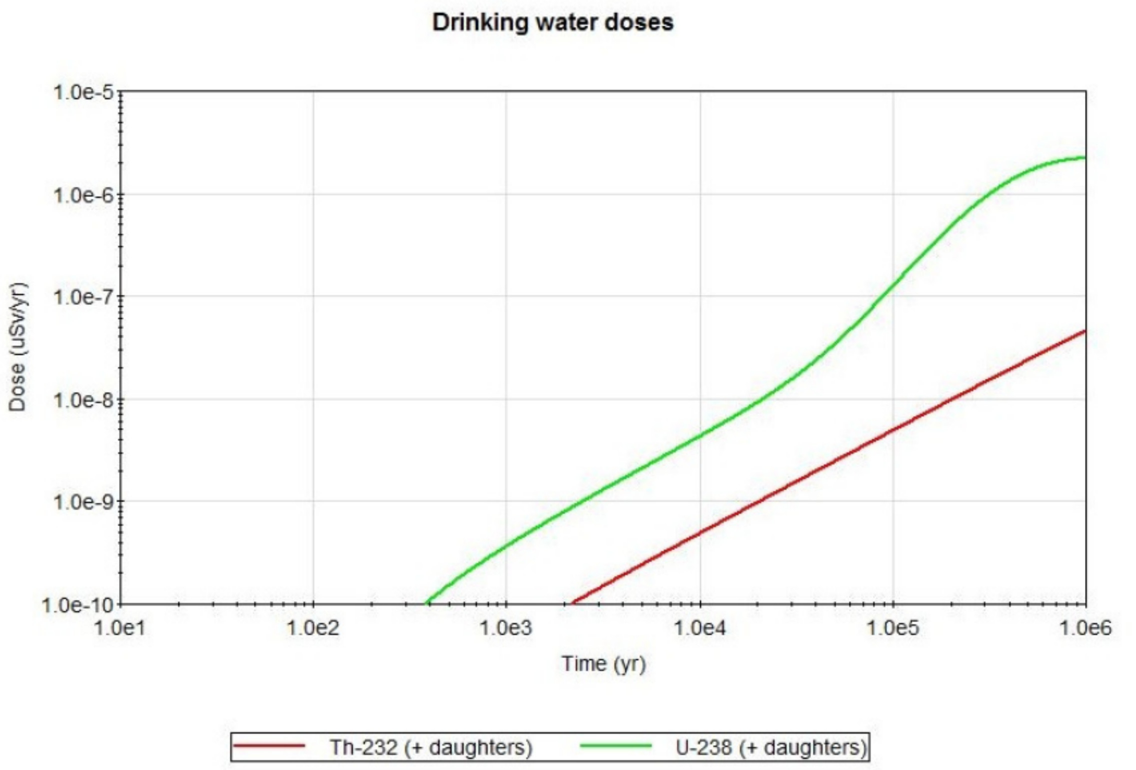


Table 146 Peak dose rate for radionuclides where the peak dose rate is after 100,000 years

Radionuclide	Time of peak dose rate (y)	Ratio peak dose rate/peak dose rate up to 100,000y
Sn-126	200,000	1.25
Th-229	10,000	2.81
Th-232	>1,000,000	10.76
U-233	150,000	1.07
U-234	300,000	2.41
U-235	600,000	2.73
U-236	500,000	1.78
Pu-238	300,000	3.24
U-238	>1,000,000	21.05
Pu-242	500,000	2.14

1068. Hence, using a time cut-off of 100,000 years captures the peak dose rate for the majority of radionuclides, underestimates the peak dose rate by less than a factor of about 3 for 8 long-lived radionuclides and underestimates the peak dose rate over the next 1,000,000 years by more than a factor of 10 for Th-232 and U-238.

1069. The effect of the different times of the peak dose for the different radionuclides is illustrated in Table 147 which shows the peak dose rates from disposal of an example inventory using the proportion of each radionuclide in the national inventory of LLW (Nuclear Decommissioning Authority, 2013) and the maximum quantity of this type of waste that can be disposed of in the site i.e. the sum of fractions for these radionuclides is <1. This example considers water extracted from a well at the boundary of the ENRMF and used for drinking and crop irrigation. The table shows that the overall peak dose rate increases as the time period increases, i.e. the peak dose rate is delivered at very long times in the future. For the first 20,000 years the peak dose is less than half the peak dose up to 100,000 years. After about 100,000 years the dose is dominated by the ingrowth of daughter products of Th-232, U-238 and to a lesser extent U-234. The dose is still below 20 $\mu Sv\ y^{-1}$.

Table 147 Peak dose rate at boundary well within different time periods after disposal for an example inventory

Radionuclide	Peak dose rate at boundary well for different time periods ($\mu Sv\ y^{-1}$)				
	60 y – 300 y	300 y – 20 ky	20 ky – 100 ky	100 ky – 300 ky	300 ky – 1 My
C-14	<1.0 10^{-8}	7.70 10^{-3}	4.07 10^{-3}	<1.0 10^{-8}	<1.0 10^{-8}
Cl-36	3.88 10^{-3}	6.81 10^{-3}	9.01 10^{-8}	<1.0 10^{-8}	<1.0 10^{-8}
Ra-226	1.45 10^{-2}	9.43 10^{-2}	<1.0 10^{-8}	<1.0 10^{-8}	<1.0 10^{-8}
Tc-99	2.97 10^{-4}	2.12 10^{-3}	1.87 10^{-3}	8.91 10^{-4}	1.40 10^{-4}
Th-230	<1.0 10^{-8}	3.60 10^{-3}	7.50 10^{-3}	7.52 10^{-3}	3.49 10^{-3}
Th-232	<1.0 10^{-8}	4.46 10^{-2}	3.84 10^{-1}	1.26	4.13
U-234	2.34 10^{-6}	2.12 10^{-3}	3.51 10^{-2}	8.47 10^{-2}	8.47 10^{-2}
U-238	1.81 10^{-5}	3.97 10^{-3}	6.99 10^{-2}	5.87 10^{-1}	1.47
Am-241	<1.0 10^{-8}	4.89 10^{-3}	5.09 10^{-3}	1.43 10^{-3}	<1.0 10^{-8}
Other	8.60 10^{-5}	3.84 10^{-3}	5.03 10^{-3}	4.92 10^{-3}	4.37 10^{-3}
Total	1.88 10^{-2}	1.74 10^{-1}	5.13 10^{-1}	1.95	5.69

1070. This analysis shows that using the peak dose rates from the 60 to 100,000 years period offers adequate protection given the size of the LLW waste inventory and the uncertainty about how much will be disposed at the site.

1071. The impact of a higher leachate head is to increase the migration rate through the barriers at the base of the waste cells. This is illustrated in Table 148 (showing the 10 radionuclides resulting in the greatest dose rate through the drinking water pathway) and indicates that for those radionuclides most likely to be limiting, the Goldsim model is not sensitive to the leachate head. This sensitivity is similar to that shown by SNIFFER and reported in the previous assessment (Augean, 2009a).

Table 148 Sensitivity of dose to leachate head: drinking water pathway

Radionuclide	Dose rate with a leachate head of 1m* ($\mu Sv\ MBq^{-1}$)	Dose rate with a leachate head of 5 m ($\mu Sv\ MBq^{-1}$)	Dose rate with a leachate head of 10 m ($\mu Sv\ MBq^{-1}$)
I-129	$1.01\ 10^{-4}$	$1.01\ 10^{-4}$	$1.01\ 10^{-4}$
Np-237	$9.35\ 10^{-6}$	$9.35\ 10^{-6}$	$9.35\ 10^{-6}$
Cl-36	$2.42\ 10^{-6}$	$2.42\ 10^{-6}$	$2.42\ 10^{-6}$
U-235	$6.73\ 10^{-7}$	$6.74\ 10^{-7}$	$6.74\ 10^{-7}$
U-234	$4.39\ 10^{-7}$	$4.40\ 10^{-7}$	$4.40\ 10^{-7}$
U-238	$1.25\ 10^{-7}$	$1.25\ 10^{-7}$	$1.25\ 10^{-7}$
U-233	$1.21\ 10^{-7}$	$1.22\ 10^{-7}$	$1.22\ 10^{-7}$
U-236	$2.94\ 10^{-8}$	$2.94\ 10^{-8}$	$2.94\ 10^{-8}$
Tc-99	$2.37\ 10^{-8}$	$2.37\ 10^{-8}$	$2.37\ 10^{-8}$
Pa-231	$1.47\ 10^{-8}$	$1.47\ 10^{-8}$	$1.47\ 10^{-8}$

* ESC value

1072. The greatest impact on dose rate is observed for radionuclides that produce lower dose rates for the drinking pathway. The greatest changes in dose rate are shown in Table 149 and occur when the head increases from 1 m to 5 m; a further increase to 10 m has less impact. However, since other scenarios dominate, these changes do not affect the radiological capacity for these radionuclides.

Table 149 Greatest sensitivity of dose to leachate head: drinking water pathway

Radionuclide	Dose rate with a leachate head of 1m ($\mu Sv\ MBq^{-1}$)	Dose rate with a leachate head of 5 m ($\mu Sv\ MBq^{-1}$)	Dose rate with a leachate head of 10 m ($\mu Sv\ MBq^{-1}$)
H-3	$1.00\ 10^{-10}$	$7.32\ 10^{-10}$	$7.81\ 10^{-10}$
Sr-90	$1.00\ 10^{-10}$	$2.55\ 10^{-10}$	$2.73\ 10^{-10}$

1073. The impact of hydraulic conductivity and the depth of the assumed clay barrier depth beneath the waste cells were examined in detail when deciding on the combination that would be used for the ESC. 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.

1074. The barrier thicknesses for the currently permitted site and the western extension have been taken from the HRA (Augean, 2014). In addition to the clay barrier a further 2 m of Rutland Formation clay will be left in situ beneath the clay barrier in the western extension. Tests indicate the hydraulic conductivity of the underlying Rutland Formation is $8.8\ 10^{-11}$ $m\ s^{-1}$ (geometric mean). This is close to the geometric mean for cells constructed using the same clay in other parts of the ENRMF site and at the nearby Thornhaugh Landfill ($6.9\ 10^{-11}\ m\ s^{-1}$).

1075. The clay barrier depth in the model was varied between 1 m and 3 m and the hydraulic conductivity between 1.15 10^{-10} m s^{-1} and 8.8 10^{-11} m s^{-1}. The results are shown below illustrating the impact of changes to these two parameters on the drinking water pathway dose rates (Table 150).

1076. On the basis of the sensitivity analysis, observed hydraulic conductivity and the depth of clay in the different parts of the site a clay barrier thickness of 1.5 m was selected for the ESC along with a hydraulic conductivity of 8.8 10^{-11} m s^{-1}. This combination is conservative, providing a depth of clay (clay barrier plus underlying in situ clay layer) that is less than the 3 m expected and a hydraulic conductivity that is not a low as that expected from the clay barrier but similar to the in situ clay.

Table 150 Sensitivity of drinking water dose rate to clay depth and hydraulic conductivity

Radionuclide	ESC	Sensitivity analysis			
	Dose per unit disposal: 1.5 m at 8.8 10^{-11} m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 1.5 m at 3.0 10^{-10} m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 3.0 m at 1.15 10^{-10} m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 2 m at 7.2 10^{-11} m s^{-1} (µSv MBq^{-1})	No liner (µSv MBq^{-1})
H3	<1.0 10^{-10}	3.80 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	7.81 10^{-10}
C14	2.87 10^{-10}	4.11 10^{-9}	4.76 10^{-10}	1.91 10^{-10}	2.91 10^{-10}
Cl36	2.42 10^{-6}	8.92 10^{-6}	3.14 10^{-6}	1.96 10^{-6}	2.42 10^{-6}
Sr90	<1.0 10^{-10}	1.78 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	2.73 10^{-10}
Nb94	<1.0 10^{-10}	4.18 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}
Tc99	2.37 10^{-8}	1.21 10^{-7}	3.09 10^{-8}	1.92 10^{-8}	2.37 10^{-8}
Ag108m	<1.0 10^{-10}	7.12 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}
Sn126	6.32 10^{-9}	6.21 10^{-8}	1.01 10^{-8}	4.31 10^{-9}	6.32 10^{-9}
I129	1.01 10^{-4}	3.75 10^{-4}	1.31 10^{-4}	8.18 10^{-5}	1.01 10^{-4}
Ra226	1.01 10^{-9}	1.93 10^{-8}	1.79 10^{-9}	6.55 10^{-10}	1.07 10^{-9}
Th229	<1.0 10^{-10}	1.65 10^{-9}	1.63 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}
Th230	9.53 10^{-9}	1.79 10^{-7}	1.68 10^{-8}	6.16 10^{-9}	9.54 10^{-9}
Pa231	1.47 10^{-8}	2.71 10^{-7}	2.58 10^{-8}	9.54 10^{-9}	1.47 10^{-8}
Th232	4.88 10^{-9}	9.13 10^{-8}	8.49 10^{-9}	3.18 10^{-9}	4.88 10^{-9}
U232	<1.0 10^{-10}	3.33 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}
U233	1.21 10^{-7}	1.11 10^{-6}	1.87 10^{-7}	8.50 10^{-8}	1.22 10^{-7}
U234	4.39 10^{-7}	5.06 10^{-6}	7.21 10^{-7}	2.97 10^{-7}	4.40 10^{-7}
U235	6.73 10^{-7}	7.29 10^{-6}	1.09 10^{-6}	4.57 10^{-7}	6.74 10^{-7}
U236	2.94 10^{-8}	2.52 10^{-7}	4.50 10^{-8}	2.06 10^{-8}	2.94 10^{-8}
Np237	9.35 10^{-6}	4.06 10^{-5}	1.22 10^{-5}	7.59 10^{-6}	9.35 10^{-6}
Pu238	1.57 10^{-10}	1.81 10^{-9}	2.58 10^{-10}	<1.0 10^{-10}	1.57 10^{-10}
U238	1.25 10^{-7}	1.29 10^{-6}	2.00 10^{-7}	8.55 10^{-8}	1.25 10^{-7}
Pu239	1.36 10^{-9}	2.26 10^{-8}	2.28 10^{-9}	8.97 10^{-10}	1.36 10^{-9}
Pu240	3.68 10^{-10}	6.26 10^{-9}	6.19 10^{-10}	2.43 10^{-10}	3.72 10^{-10}
Pu241	<1.0 10^{-10}	2.71 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}	<1.0 10^{-10}

Radionuclide	ESC	Sensitivity analysis			
	Dose per unit disposal: 1.5 m at $8.8\ 10^{-11}$ m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 1.5 m at $3.0\ 10^{-10}$ m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 3.0 m at $1.15\ 10^{-10}$ m s^{-1} (µSv MBq^{-1})	Dose per unit disposal: 2 m at $7.2\ 10^{-11}$ m s^{-1} (µSv MBq^{-1})	No liner (µSv MBq^{-1})
Am241	$1.88\ 10^{-9}$	$8.15\ 10^{-9}$	$2.45\ 10^{-9}$	$1.53\ 10^{-9}$	$1.88\ 10^{-9}$
Pu242	$8.31\ 10^{-9}$	$1.32\ 10^{-7}$	$1.39\ 10^{-8}$	$5.52\ 10^{-9}$	$8.32\ 10^{-9}$

1077. The first column shows the results of the parameter combination used in the ESC. Various combinations are then shown moving right to the last column which indicates the dose rate when it is assumed that there is no liner.

1078. The impact of diffusion through the liner, i.e. a diffusive flux between the waste cell and the clay barrier on the activity concentration in the groundwater was investigated for H-3. Three cases were compared:

- No diffusion (as modelled in the ESC);
- Diffusion through holes in the basal liner only; and
- Diffusion through the whole surface of the basal liner.

1079. The assumed diffusive length in all cases was the thickness of the barrier (1.5 m). The area of the holes in the liner ($A_{Diff,h}$) was calculated as:

$$A_{Diff,h} = A_{Basal} \cdot \left(n_p \cdot a_p + n_h \cdot a_h + n_t \cdot a_t\right)$$

1080. The results of these calculations are summarised in the table below.

Table 151 Diffusion of H-3 through the liner

Case	H-3 activity concentration in groundwater (Bq l^{-1} MBq^{-1})
No diffusion (ESC)	<9.3 10^{-9}
Diffusion through the holes in the basal liner only	<9.3 10^{-9}
Diffusion through the whole surface of the basal liner	$1.24\ 10^{-8}$

1081. It is expected that diffusion would only occur through the holes in the basal liner. In that case diffusion would have no impact on the H-3 activity concentration in groundwater. In the bounding case assuming diffusion through the whole area of the basal liner, the H-3 activity concentration in groundwater would be just above the dose rate cut-off applied in Goldsim. The radiological capacity for H-3 is limited by gas release from the landfill and the dose from diffusion through the whole surface of the basal liner, 1.34 10^{-10} µSv y^{-1} MBq^{-1} does not limit the radiological capacity of H-3.

Intrusion – Borehole driller

1082. The Borehole driller scenario (see Section E.5.2) limits the radiological capacity of the ENRMF for 22 radionuclides. The impact of changes to parameter values has considered waste bulk density, exposure duration and timing of the event. The impact of bulk density and duration is to produce a linear change in the radiological capacity.

1083. The impact of waste bulk density is to produce a decrease in the dose rate with increasing waste bulk density (Table 152). The parameters values chosen range from the value used in the previous assessment through to the upper value used in the HRA. The adopted ESC value is 1530 kg m^{-3}. Dose rate changes in direct proportion to the change in waste bulk density. The radionuclides listed are those where the Borehole driller scenario constrains radiological capacity.

Table 152 Sensitivity of dose to waste bulk density: Borehole driller

Radionuclide	Dose per unit disposal at 700 kg m^{-3} (µSv MBq^{-1})	Dose per unit disposal at 1490 kg m^{-3} (µSv MBq^{-1})	ESC value Dose per unit disposal at 1530 kg m^{-3} (µSv MBq^{-1})	Dose per unit disposal at 1780 kg m^{-3} (µSv MBq^{-1})
Fe-55	1.13 10^{-16}	5.29 10^{-17}	5.15 10^{-17}	4.43 10^{-17}
Co-60	1.71 10^{-8}	8.05 10^{-9}	7.84 10^{-9}	6.74 10^{-9}
Sb-125	1.97 10^{-12}	9.24 10^{-13}	9.00 10^{-13}	7.73 10^{-13}
Ba-133	1.07 10^{-7}	5.04 10^{-8}	4.91 10^{-8}	4.22 10^{-8}
Cs-134	4.80 10^{-14}	2.25 10^{-14}	2.19 10^{-14}	1.89 10^{-14}
Cs-137	2.43 10^{-6}	1.14 10^{-6}	1.11 10^{-6}	9.54 10^{-7}
Pm-147	1.38 10^{-16}	6.49 10^{-17}	6.32 10^{-17}	5.44 10^{-17}
Eu-152	9.15 10^{-7}	4.30 10^{-7}	4.19 10^{-7}	3.60 10^{-7}
Eu-154	1.72 10^{-7}	8.08 10^{-8}	7.86 10^{-8}	6.76 10^{-8}
Eu-155	8.29 10^{-11}	3.89 10^{-11}	3.79 10^{-11}	3.26 10^{-11}
Ra-226	4.22 10^{-5}	1.98 10^{-5}	1.93 10^{-5}	1.66 10^{-5}
Ra-228	3.65 10^{-8}	1.72 10^{-8}	1.67 10^{-8}	1.44 10^{-8}
Ac-227	6.45 10^{-6}	3.03 10^{-6}	2.95 10^{-6}	2.53 10^{-6}
Th-229	2.20 10^{-5}	1.03 10^{-5}	1.01 10^{-5}	8.65 10^{-6}
U-232	1.54 10^{-6}	7.25 10^{-7}	7.06 10^{-7}	6.07 10^{-7}
Pu-238	4.68 10^{-6}	2.20 10^{-6}	2.14 10^{-6}	1.84 10^{-6}
Pu-239	8.19 10^{-6}	3.85 10^{-6}	3.74 10^{-6}	3.22 10^{-6}
Pu-240	8.15 10^{-6}	3.83 10^{-6}	3.73 10^{-6}	3.20 10^{-6}
Pu-241	2.05 10^{-7}	9.64 10^{-8}	9.38 10^{-8}	8.07 10^{-8}
Am-241	6.07 10^{-6}	2.85 10^{-6}	2.78 10^{-6}	2.39 10^{-6}
Cm-243	1.53 10^{-6}	7.21 10^{-7}	7.02 10^{-7}	6.03 10^{-7}
Cm-244	4.12 10^{-7}	1.93 10^{-7}	1.88 10^{-7}	1.62 10^{-7}

1084. The range of waste bulk density has a small impact on the calculated dose rate and the dose rate would increase by about 3% if the lower bulk density from the HRA (give value) was used.

1085. The impact on radiological capacity is illustrated in Table 153 where radiological capacity increases with increasing bulk density. The radiological capacities of all radionuclides shown in Table 152 increase in this way. Those listed in Table 153 would become constrained by the Borehole driller scenario due to a change in the waste bulk density: in this instance only the radiological capacity at 700 kg m^{-3} are slightly lower than the ENRMF value.

Table 153 Sensitivity of radiological capacity to waste bulk density: Borehole driller

Radionuclide	Scenario radiological capacity at 700 kg m^{-3} (MBq)	Scenario radiological capacity at 1490 kg m^{-3} (MBq)	Scenario radiological capacity at 1530 kg m^{-3} (MBq)	Scenario radiological capacity at 1780 kg m^{-3} (MBq)
Nb-94	$1.10\ 10^{8}$	$2.34\ 10^{8}$	$2.40\ 10^{8}$	$2.79\ 10^{8}$
Ag-108m	$1.22\ 10^{8}$	$2.59\ 10^{8}$	$2.66\ 10^{8}$	$3.09\ 10^{8}$
Th-232	$5.17\ 10^{7}$	$1.10\ 10^{8}$	$1.13\ 10^{8}$	$1.31\ 10^{8}$
Pu-242	$3.99\ 10^{8}$	$8.48\ 10^{8}$	$8.71\ 10^{8}$	$1.01\ 10^{9}$

1086. The sensitivity of the assessment to exposure time assumptions has also been considered. The impact of exposure time is to produce an increase in dose rate with increasing exposure time (Table 154). The parameter values used range from the value for a single borehole (the ESC assumed 5 boreholes) to an upper value selected to illustrate the exposure time increase needed for this scenario to limit the radiological capacity of the four radionuclides in Table 153.

Table 154 Sensitivity of dose to exposure time: Borehole driller

Radionuclide	Dose per unit disposal; 16 h exposure (µSv MBq^{-1})	ESC Dose per unit disposal; 80 h exposure (µSv MBq^{-1})	Dose per unit disposal; 88 h exposure (µSv MBq^{-1})	Dose per unit disposal; 160 h exposure (µSv MBq^{-1})
Fe-55	$1.03\ 10^{-17}$	$5.15\ 10^{-17}$	$5.66\ 10^{-17}$	$1.03\ 10^{-16}$
Co-60	$1.57\ 10^{-9}$	$7.84\ 10^{-9}$	$8.62\ 10^{-9}$	$1.57\ 10^{-8}$
Sb-125	$1.80\ 10^{-13}$	$9.00\ 10^{-13}$	$9.90\ 10^{-13}$	$1.80\ 10^{-12}$
Ba-133	$9.82\ 10^{-9}$	$4.91\ 10^{-8}$	$5.40\ 10^{-8}$	$9.82\ 10^{-8}$
Cs-134	$4.39\ 10^{-15}$	$2.19\ 10^{-14}$	$2.41\ 10^{-14}$	$4.39\ 10^{-14}$
Cs-137	$2.22\ 10^{-7}$	$1.11\ 10^{-6}$	$1.22\ 10^{-6}$	$2.22\ 10^{-6}$
Pm-147	$1.26\ 10^{-17}$	$6.32\ 10^{-17}$	$6.96\ 10^{-17}$	$1.26\ 10^{-16}$
Eu-152	$8.38\ 10^{-8}$	$4.19\ 10^{-7}$	$4.61\ 10^{-7}$	$8.38\ 10^{-7}$
Eu-154	$1.57\ 10^{-8}$	$7.86\ 10^{-8}$	$8.65\ 10^{-8}$	$1.57\ 10^{-7}$
Eu-155	$7.58\ 10^{-12}$	$3.79\ 10^{-11}$	$4.17\ 10^{-11}$	$7.58\ 10^{-11}$

Radionuclide	Dose per unit disposal; 16 h exposure (µSv MBq^{-1})	ESC Dose per unit disposal; 80 h exposure (µSv MBq^{-1})	Dose per unit disposal; 88 h exposure (µSv MBq^{-1})	Dose per unit disposal; 160 h exposure (µSv MBq^{-1})
Ra-226	$3.86\ 10^{-6}$	$1.93\ 10^{-5}$	$2.12\ 10^{-5}$	$3.86\ 10^{-5}$
Ra-228	$3.34\ 10^{-9}$	$1.67\ 10^{-8}$	$1.84\ 10^{-8}$	$3.34\ 10^{-8}$
Ac-227	$5.90\ 10^{-7}$	$2.95\ 10^{-6}$	$3.24\ 10^{-6}$	$5.90\ 10^{-6}$
Th-229	$2.01\ 10^{-6}$	$1.01\ 10^{-5}$	$1.11\ 10^{-5}$	$2.01\ 10^{-5}$
U-232	$1.41\ 10^{-7}$	$7.06\ 10^{-7}$	$7.76\ 10^{-7}$	$1.41\ 10^{-6}$
Pu-238	$4.28\ 10^{-7}$	$2.14\ 10^{-6}$	$2.35\ 10^{-6}$	$4.28\ 10^{-6}$
Pu-239	$7.49\ 10^{-7}$	$3.74\ 10^{-6}$	$4.12\ 10^{-6}$	$7.49\ 10^{-6}$
Pu-240	$7.46\ 10^{-7}$	$3.73\ 10^{-6}$	$4.10\ 10^{-6}$	$7.46\ 10^{-6}$
Pu-241	$1.88\ 10^{-8}$	$9.38\ 10^{-8}$	$1.03\ 10^{-7}$	$1.88\ 10^{-7}$
Am-241	$5.55\ 10^{-7}$	$2.78\ 10^{-6}$	$3.05\ 10^{-6}$	$5.55\ 10^{-6}$
Cm-243	$1.40\ 10^{-7}$	$7.02\ 10^{-7}$	$7.72\ 10^{-7}$	$1.40\ 10^{-6}$
Cm-244	$3.77\ 10^{-8}$	$1.88\ 10^{-7}$	$2.07\ 10^{-7}$	$3.77\ 10^{-7}$

1087. The change in exposure time from 88 h to 80 h has a small impact, decreasing dose rate by about 10%. The impact on radiological capacity is illustrated in Table 155 where radiological capacity decreases with increasing exposure time. Those listed in Table 155 would become constrained by the Borehole driller scenario due to a change in the exposure time from 80 h to 160 h.

Table 155 Sensitivity of radiological capacity to exposure time: Borehole driller

Radionuclide	Scenario radiological capacity; 16 h exposure (MBq)	ESC Scenario radiological capacity; 80 h exposure (MBq)	Scenario radiological capacity; 88 h exposure (MBq)	Scenario radiological capacity; 160 h exposure (MBq)
Nb-94	$1.20\ 10^{9}$	$2.40\ 10^{8}$	$2.18\ 10^{8}$	$1.20\ 10^{8}$
Ag-108m	$1.33\ 10^{9}$	$2.66\ 10^{8}$	$2.41\ 10^{8}$	$1.33\ 10^{8}$
Th-232	$5.65\ 10^{8}$	$1.13\ 10^{8}$	$1.03\ 10^{8}$	$5.65\ 10^{7}$
Pu-242	$4.36\ 10^{9}$	$8.71\ 10^{8}$	$7.92\ 10^{8}$	$4.36\ 10^{8}$

1088. The impact of changing the timing of the intrusion event is shown using two example inventories. The first inventory uses the proportion of each radionuclide in the national inventory of LLW (Nuclear Decommissioning Authority, 2013) and the maximum quantity of this type of waste that can be disposed of in the site i.e. the sum of fractions for these radionuclides is <1. The second uses the proportions in the waste that is already disposed at the ENRMF, and the maximum quantity of this waste that ensures that the sum of fractions is <1. The dose is then calculated for up to 20,000 years after closure.

1089. Table 156 shows how the estimated dose decreases as the elapsed time period to intrusion increases. The radiological capacity of individual radionuclides is never lower than 89.6 TBq and it is unreasonable to assume intrusion will occur earlier than the period of managed controls.

Table 156 Sensitivity of projected dose to timing of exposure: Borehole driller

Years after closure	Dose from inventory based on (Nuclear Decommissioning Authority, 2013) (mSv)	Dose from inventory based on ENRMF disposals (mSv)
0	2.96 10^{-1}	6.43 10^{-1}
20	1.28 10^{-1}	5.70 10^{-1}
60*	6.91 10^{-2}	5.24 10^{-1}
150	3.70 10^{-2}	4.89 10^{-1}
200	3.20 10^{-2}	4.78 10^{-1}
500	2.09 10^{-2}	4.26 10^{-1}
2,000	6.14 10^{-3}	2.55 10^{-1}
10,000	1.87 10^{-3}	7.34 10^{-2}
20,000	1.07 10^{-3}	5.43 10^{-2}

*Period used in the ESC to determine radiological capacity

1090. The dose to the borehole driller who was exposed to excavated spoil containing waste that was at the maximum activity concentration (200 Bq g^{-1}) was also assessed explicitly. Table 157 shows the 5 radionuclides giving the largest doses after the period of authorisation.

Table 157 Dose to borehole driller exposed to waste containing 200 Bq g^{-1}

Radionuclide	Dose (mSv)
Nb-94	5
Ra-226	6
Th-229	4
Th-232	10
Pa-231	8

1091. The doses are between 4 and 10 mSv. The dose guidance levels in the GRA for intrusion scenarios are 3 to 20 mSv, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, 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 (20 mSv). The estimated doses are all well below the upper dose guidance level. In the case of Nb-94 and Ra-226 the main contributor to dose is external irradiation. For the other three radionuclides, the inhalation dose is the main contributor. Hence the 200 Bq g^{-1} limit on activity concentration provides adequate protection for the borehole driller.

Intrusion - Smallholder

1092. The Smallholder scenario (see Section E.5.9) limits the radiological capacity of the ENRMF for 8 radionuclides. The impact of changes to parameter values has considered waste bulk density, the dilution factor and timing of the event. The impact is to produce a linear change in the radiological capacity.

1093. The effect of changing the waste bulk density is to produce a decrease in dose rate with increasing waste bulk density (Table 158). The parameter values used range from the value used in the previous assessment through to the upper value used in the HRA. The adopted ESC value is 1530 kg m^{-3}. The dose changes in direct proportion to the change in waste bulk density. The radionuclides listed are those where the Smallholder scenario constrains the radiological capacity.

Table 158 Sensitivity of dose to waste bulk density: Smallholder

Radionuclide	Dose per unit disposal at 700 kg m^{-3} (µSv MBq^{-1})	Dose per unit disposal at 1490 kg m^{-3} (µSv MBq^{-1})	ESC Dose per unit disposal at 1530 kg m^{-3} (µSv MBq^{-1})	Dose per unit disposal at 1780 kg m^{-3} (µSv MBq^{-1})
Ni-63	$2.36\ 10^{-8}$	$1.11\ 10^{-8}$	$1.08\ 10^{-8}$	$9.29\ 10^{-9}$
Sr-90	$5.29\ 10^{-6}$	$2.48\ 10^{-6}$	$2.42\ 10^{-6}$	$2.08\ 10^{-6}$
Nb-94	$4.56\ 10^{-5}$	$2.14\ 10^{-5}$	$2.09\ 10^{-5}$	$1.80\ 10^{-5}$
Tc-99	$7.23\ 10^{-5}$	$3.40\ 10^{-5}$	$3.31\ 10^{-5}$	$2.84\ 10^{-5}$
Ag-108m	$3.28\ 10^{-5}$	$1.54\ 10^{-5}$	$1.50\ 10^{-5}$	$1.29\ 10^{-5}$
Th-230	$9.46\ 10^{-5}$	$4.44\ 10^{-5}$	$4.33\ 10^{-5}$	$3.72\ 10^{-5}$
Th-232	$9.15\ 10^{-5}$	$4.30\ 10^{-5}$	$4.19\ 10^{-5}$	$3.60\ 10^{-5}$
Pa-231	$3.53\ 10^{-4}$	$1.66\ 10^{-4}$	$1.61\ 10^{-4}$	$1.39\ 10^{-4}$

1094. The range of waste bulk density used in the HRA has a small impact on the calculated dose and would increase the dose by about 3% if the lower bulk density from the HRA was used.

1095. The impact on radiological capacity is illustrated in Table 159 where radiological capacity increases with increasing bulk density. The radiological capacities of all radionuclides shown in Table 158 increase in this way. Those listed in Table 159 would become constrained by the Smallholder scenario due to a change in the waste bulk density, for example the radiological capacity of Th-229 at 700 kg m^{-3} is lower than the ENRMF value.

Table 159 Sensitivity of radiological capacity to waste bulk density: Smallholder

Radionuclide	Scenario radiological capacity at 700 kg m^{-3} (MBq)	Scenario radiological capacity at 1490 kg m^{-3} (MBq)	Scenario radiological capacity at 1530 kg m^{-3} (MBq)	Scenario radiological capacity at 1780 kg m^{-3} (MBq)
Sn-126	$1.65\ 10^{8}$	$3.51\ 10^{8}$	$3.60\ 10^{8}$	$4.19\ 10^{8}$
Th-229	$1.68\ 10^{8}$	$3.57\ 10^{8}$	$3.67\ 10^{8}$	$4.27\ 10^{8}$
U-232	$4.20\ 10^{9}$	$8.93\ 10^{9}$	$9.17\ 10^{9}$	$1.07\ 10^{10}$

Radionuclide	Scenario radiological capacity at 700 kg m^{-3} (MBq)	Scenario radiological capacity at 1490 kg m^{-3} (MBq)	Scenario radiological capacity at 1530 kg m^{-3} (MBq)	Scenario radiological capacity at 1780 kg m^{-3} (MBq)
Pu-241	$3.05\ 10^{10}$	$6.50\ 10^{10}$	$6.67\ 10^{10}$	$7.76\ 10^{10}$
Am-241	$1.05\ 10^{9}$	$2.24\ 10^{9}$	$2.30\ 10^{9}$	$2.68\ 10^{9}$

1096. The sensitivity of the Smallholder assessment to the dilution factor assumptions has also been considered. The impact of a numerically smaller dilution factor is to reduce the dose (Table 160) since it is applied as a multiplying factor. The dilution factors used range from that used in the previous assessment (1/10,000) to 0.48 which accounts only for dilution by uncontaminated material (the cap of 1.6 m and the hazardous waste layer of 1 m when a 5 m excavation occurs). The other dilution factors are obtained by applying further dilution to the 0.48 factor, to account for the average LLW content of the ENRMF (0.096) and the dilution with clean soil (10% spoil giving 0.0096). The sensitivity of the Smallholder dose is shown in Table 160.

Table 160 Sensitivity of dose to dilution factor: Smallholder

Radionuclide	Dose per unit disposal; *DIL*=0.0001 (µSv MBq^{-1})	ESC soil Dose per unit disposal; *DIL*=0.0096 (µSv MBq^{-1})	Dose per unit disposal; *DIL*=0.096 (µSv MBq^{-1})	Dose per unit disposal; *DIL*=0.48 (µSv MBq^{-1})
Ni-63	$1.13\ 10^{-16}$	$1.08\ 10^{-14}$	$1.08\ 10^{-13}$	$5.40\ 10^{-13}$
Sr-90	$2.52\ 10^{-14}$	$2.42\ 10^{-12}$	$2.42\ 10^{-11}$	$1.21\ 10^{-10}$
Nb-94	$2.18\ 10^{-13}$	$2.09\ 10^{-11}$	$2.09\ 10^{-10}$	$1.04\ 10^{-9}$
Tc-99	$3.45\ 10^{-13}$	$3.31\ 10^{-11}$	$3.31\ 10^{-10}$	$1.65\ 10^{-9}$
Ag-108m	$1.56\ 10^{-13}$	$1.50\ 10^{-11}$	$1.50\ 10^{-10}$	$7.49\ 10^{-10}$
Th-230	$4.51\ 10^{-13}$	$4.33\ 10^{-11}$	$4.33\ 10^{-10}$	$2.16\ 10^{-9}$
Th-232	$4.36\ 10^{-13}$	$4.19\ 10^{-11}$	$4.19\ 10^{-10}$	$2.09\ 10^{-9}$
Pa-231	$1.68\ 10^{-12}$	$1.61\ 10^{-10}$	$1.61\ 10^{-9}$	$8.07\ 10^{-9}$

1097. 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 161 where radiological capacity decreases with higher values of the dilution factor. Those listed in Table 161 would become constrained by the Smallholder scenario due to a higher value of the dilution factor.

Table 161 Sensitivity of radiological capacity to dilution factors: Smallholder

Radionuclide	Scenario radiological capacity; *DIL*=0.0001 (MBq)	Scenario radiological capacity; *DIL*=0.0096 (MBq)	Scenario radiological capacity; *DIL*=0.096 (MBq)	Scenario radiological capacity; *DIL*=0.48 (MBq)
C-14*	$7.76\ 10^{10}$	$8.08\ 10^{8}$	$8.08\ 10^{7}$	$1.62\ 10^{7}$
Cl-36	$5.19\ 10^{9}$	$5.40\ 10^{7}$	$5.40\ 10^{6}$	$1.08\ 10^{6}$

Radionuclide	Scenario radiological capacity; *DIL*=0.0001 (MBq)	Scenario radiological capacity; *DIL*=0.0096 (MBq)	Scenario radiological capacity; *DIL*=0.096 (MBq)	Scenario radiological capacity; *DIL*=0.48 (MBq)
Sn-126*	$3.46\ 10^{10}$	$3.60\ 10^{8}$	$3.60\ 10^{7}$	$7.20\ 10^{6}$
Cs-137*	$2.35\ 10^{12}$	$2.44\ 10^{10}$	$2.44\ 10^{9}$	$4.89\ 10^{8}$
Pb-210*	$1.46\ 10^{12}$	$1.52\ 10^{10}$	$1.52\ 10^{9}$	$3.05\ 10^{8}$
Th-229*	$3.52\ 10^{10}$	$3.67\ 10^{8}$	$3.67\ 10^{7}$	$7.34\ 10^{6}$
U-232*	$8.81\ 10^{11}$	$9.17\ 10^{9}$	$9.17\ 10^{8}$	$1.83\ 10^{8}$
Pu-238*	$9.06\ 10^{11}$	$9.43\ 10^{9}$	$9.43\ 10^{8}$	$1.89\ 10^{8}$
Pu-239*	$1.72\ 10^{11}$	$1.79\ 10^{9}$	$1.79\ 10^{8}$	$3.59\ 10^{7}$
Pu-240*	$1.75\ 10^{11}$	$1.82\ 10^{9}$	$1.82\ 10^{8}$	$3.64\ 10^{7}$
Pu-241*	$6.41\ 10^{12}$	$6.67\ 10^{10}$	$6.67\ 10^{9}$	$1.33\ 10^{9}$
Pu-242*	$1.82\ 10^{11}$	$1.89\ 10^{9}$	$1.89\ 10^{8}$	$3.78\ 10^{7}$
Am-241*	$2.21\ 10^{11}$	$2.30\ 10^{9}$	$2.30\ 10^{8}$	$4.60\ 10^{7}$
Cm-243	$1.35\ 10^{13}$	$1.41\ 10^{11}$	$1.41\ 10^{10}$	$2.82\ 10^{9}$
Cm-244	$5.82\ 10^{13}$	$6.06\ 10^{11}$	$6.06\ 10^{10}$	$1.21\ 10^{10}$

1098. The move from a dilution factor of 0.0096 to 0.096 constrains 12 radionuclides as marked with an asterisk.

1099. The impact of changing the timing of the intrusion event is shown using two example inventories. The first inventory uses the proportion of each radionuclide in the national inventory of LLW (Nuclear Decommissioning Authority, 2013) and the maximum quantity of this type of waste that can be disposed of at the site i.e. the sum of fractions for these radionuclides is <1. The second uses the proportions in the waste already disposed at the ENRMF and the maximum quantity of this type of waste that can be disposed of at the site. The dose is then calculated for up to 20,000 years after closure.

1100. Table 162 shows how the estimated dose decreases with increases delay before the time of exposure for both inventories. This overall trend masks the effects of ingrowth for some radionuclides: the Th-230 dose increases with time due to the ingrowth of Ra-226. The doses are all below the 3 mSv y^{-1} dose criterion, even for times of exposure below 200 years.

1101. As the time of exposure falls below about 90 years post-closure, the radiological capacity of Sr-90 becomes less than 89.6 TBq. However, it is considered unlikely that a smallholder will excavate the site within such a short time after closure (the period of authorisation is assumed to be 60 years, records of the presence of a hazardous and radioactive waste landfill site would persist beyond this time, and the EA would not end the Permit if they considered there to still to be risks above their Permit revocation criteria).

Table 162 Sensitivity of projected dose to timing of exposure: Smallholder

Years after closure	Dose from inventory based on (Nuclear Decommissioning Authority, 2013) (mSv)	Dose from inventory based on ENRMF disposals (mSv)
20	2.83	1.60
60	1.10	$6.18\ 10^{-1}$
150	$1.61\ 10^{-1}$	$1.88\ 10^{-1}$
200*	$7.58\ 10^{-2}$	$1.58\ 10^{-1}$
500	$3.33\ 10^{-2}$	$1.47\ 10^{-1}$
2,000	$2.19\ 10^{-2}$	$1.39\ 10^{-1}$
10,000	$8.06\ 10^{-3}$	$5.98\ 10^{-2}$
20,000	$2.38\ 10^{-3}$	$1.80\ 10^{-2}$

*Period used in the ESC to determine radiological capacity

1102. The dose to the smallholder who was exposed to excavated spoil containing waste that was at the maximum activity concentration (200 Bq g^{-1}) was also assessed explicitly. The estimated annual dose for the majority of the radionuclides considered was equal to or below 3 mSv, the lower guidance level for human intrusion scenarios. Two radionuclides (Th-230 and Th-232) gave doses of 3 mSv and Table 163 shows the expected doses for the three radionuclides for which the annual dose was greater than 3 mSv. The maximum recorded activity concentration in disposed waste is also shown.

Table 163 Dose to the smallholder exposed to waste containing 200 Bq g^{-1}

Radionuclide	Dose from waste disposed at 200 Bq g^{-1} (mSv)	Maximum recorded activity concentration in disposed waste (Bq g^{-1})
Cl-36	4	0.9
I-129	8	0.001
Pa-231	12	0.1

1103. The doses are between 4 and 12 mSv. The dose guidance levels specified in the GRA are 3 to 20 mSv y^{-1}, where the lower end of the guidance is taken to apply to exposures that may persist over time. In this case, the exposure is regarded as a long term event and hence the appropriate dose guidance value would lie towards the lower end of the range cited.

1104. In the case of Cl-36 and I-129 the main contributor to dose is from livestock consumption (meat and milk), for Pa-231 it is vegetable crops. The radiological capacity of Cl-36 and I-129 are determined by the groundwater pathway and Pa-231 by the smallholder scenario. Disposals to date of these radionuclides at the ENRMF have all had a substantially lower specific activity than 200 Bq g^{-1} and the maximum activity concentrations disposed of would lead to doses well below the 3 mSv y^{-1} lower guidance level. Consideration of the national inventory also suggests that these radionuclides are unlikely to appear in any great amount in the LLW offered for disposal at the ENRMF, comprising less than 0.2% of available waste. Hence, it is extremely unlikely that waste containing activity concentrations greater than a few Bq g^{-1} of these radionuclides would be received and hence the dose to the smallholder from these radionuclides in the waste would be well below 3 mSv. The expected mixtures of radionuclides in LLW and the overall activity concentration limit of 200 Bq g^{-1} for the total activity concentration in the waste (summed over all radionuclides present) will therefore limit the maximum activity concentration of these three radionuclides to levels that give rise to doses well below 3 mSv y^{-1} anyway. Hence, the smallholder will be sufficiently protected. On this basis it does not seem necessary to specify individual limits on the specific activity for these three radionuclides.

E.7.3. Waste heterogeneity

Exposure to higher than average package

1105. The potential impact from exposure to a higher activity concentration within a waste package that forms part of a 10 t consignment was assessed using cautious assumptions. It is assumed that the specific activity of the consignment is 200 Bq g^{-1} but it may contain packages with a disproportionate amount of activity.

1106. The dose to a trial pit excavator who uncovers just LLW i.e. a single consignment of 10 t, corresponding to 10 packages, each with a specific activity of 200 Bq g^{-1} is given in (Table 164). 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}.

Table 164 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	$3.28\ 10^{-1}$	$4.74\ 10^{-1}$

Exposure to particles

1107. The analysis of a range of particle types was undertaken (see Section E.5.10.1). These were bounding cases and it is expected that doses from other particle types will be lower. The ingestion dose, if encountered, scales linearly with the activity on the particle. The size of the particle or fragment will determine the appropriate exposure pathways and larger items (fragments) would give lower skin doses than smaller particles containing the same activity due to self-absorption within the item and shorter contact times. Therefore, no further sensitivity studies have been undertaken for this waste type.

Exposure to large contaminated waste items: sensitivity

1108. The assessment of large contaminated waste items was presented in Section E.5.11. The primary parameters that may be subject to uncertainty are the duration of the exposure (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. All sensitivity calculations are based on the hypothetical concrete block containing Cs-137.

Duration of exposure

1109. Varying the exposure duration simply scales the total dose estimated in a direct and linear fashion. That is, doubling the exposure period doubles the dose received. The assumption for the geotechnical worker of handling time per core may be varied (for example, a handling time of 4 hours rather than the assumed 2 hour exposure may be used) or the exposure time may be used as a surrogate for the number of contaminated cores handled. Similarly, for the site occupant, the exposure time reflects both the frequency and duration of time spent in the vicinity of the contaminated waste.

Time of exposure

1110. Increasing the time at which exposure occurs decreases the dose estimate as a consequence of radioactive decay, and is thus dependant on the radionuclide fingerprint assumed for the waste. In the case of the hypothetical concrete block, contamination with Cs-137 has been assumed (200 Bq g^{-1}). Cs-137 has a half-life of about 30.2 years. If a site investigator undertook drilling immediately after emplacement of the block, for an exposure time of 2 hours at an average distance of 1 m (all other assumptions remaining constant), the whole-body dose would be approximately 7.3 µSv (i.e. about 4 times the dose incurred assuming exposure 60 years after emplacement of the waste, the reference time). At 300 years, the dose would reduce to 0.0075 µSv, reflecting a 1000 fold decrease in line with 10 half-lives.

1111. For the site occupier the dose rises to 2.3 mSv if exposure occurs immediately after waste emplacement, but reduces to 9 µSv if intrusion and exposure of the waste is delayed to occur 300 years after emplacement of the waste. The reference time of exposure is 60 years after closure. In the case of natural erosion of the site, which is assumed for illustration purposes to occur on a timescale of thousands of years, the dose effectively drops to zero.

1112. For very long-lived radionuclides (such as Pu-239 or C-14 identified in some waste streams) this reduction of dose with increased delay before exposure occurs would be much reduced.

Distance from the waste

1113. The geotechnical worker is assumed to be, on average, 1 m distant from the drill core for the period of exposure. If it is assumed that the average distance is 2 m, the dose from the Cs-137 in the waste at 60 years after emplacement of the waste remains almost constant (reducing from 1.9 µSv to 1.8 µSv per 2 hour exposure). In this case, the dominant exposure pathway is via unintentional ingestion of contaminated surface dust from the core.

1114. The dose to the site occupier is not dependent on the distance from the waste.

Ingestion and inhalation rate

1115. The fractional dose contribution from inhalation or ingestion of contaminated dust is dependent on the radionuclides present in the waste, and on the assumed distance of the exposed person, on average, from the waste. In the case of the hypothetical waste concrete containing Cs-137, the inhalation dose is negligible by comparison to the drill core handling dose at a distance of 0.05 m or the site occupant dose from a semi-infinite slab at 2 m. At 60 years from emplacement of waste, inhalation contributes around 0.1% of the dose arising from external exposure. However, for a point source at 1 m distance, the external dose reduces by some orders of magnitude and the dominant dose contribution becomes the unintentional ingestion of dust, with both inhalation and external exposure roughly equal and some two orders of magnitude lower than the ingestion contribution (Table 165). This reflects the relatively high rate of ingestion assumed (34 mg per hour). The US EPA (US EPA, 2005) recommends an unintentional soil intake of 100 mg per day for adults, with an upper estimate of 200 mg per day associated with activities such as mountain biking. Unfortunately the length of time over which such intake may occur within a day is not stated. However, if an activity duration of 4 to 5 hours is assumed the rate of ingestion would be around 20 to 25 mg per hour, rather lower than that assumed for the site investigator.

Table 165 Comparison of dose pathway contributions for a site investigator and site occupant for a Cs-137 contaminated waste item

	Dose (mSv/y) at 60 years			
	External	Inhalation	Ingestion	Total
Drill core handler				
0.05 m distance	$1.35\ 10^{-2}$	$2.23\ 10^{-5}$	$1.80\ 10^{-3}$	$1.53\ 10^{-2}$
1 m distance	$3.37\ 10^{-5}$			$1.85\ 10^{-3}$
Site occupant				
2 m distance	2.25	$4.63\ 10^{-3}$	$4.79\ 10^{-4}$	2.25

1116. For the site occupant, the Cs-137 ingestion dose is generally trivial (less than 1% of the dose arising from external irradiation).

Depth of contamination

1117. The greater the depth of penetration of the contamination in the waste item, the lower the activity concentration in the relevant layer since the average activity concentration over the block remains at 200 Bq g^{-1}. The dose to the site investigator and site occupant varies as summarised below (Table 166).

Table 166 Dose to site investigator and site occupant at 60 years from emplacement of Cs-137 contaminated waste item

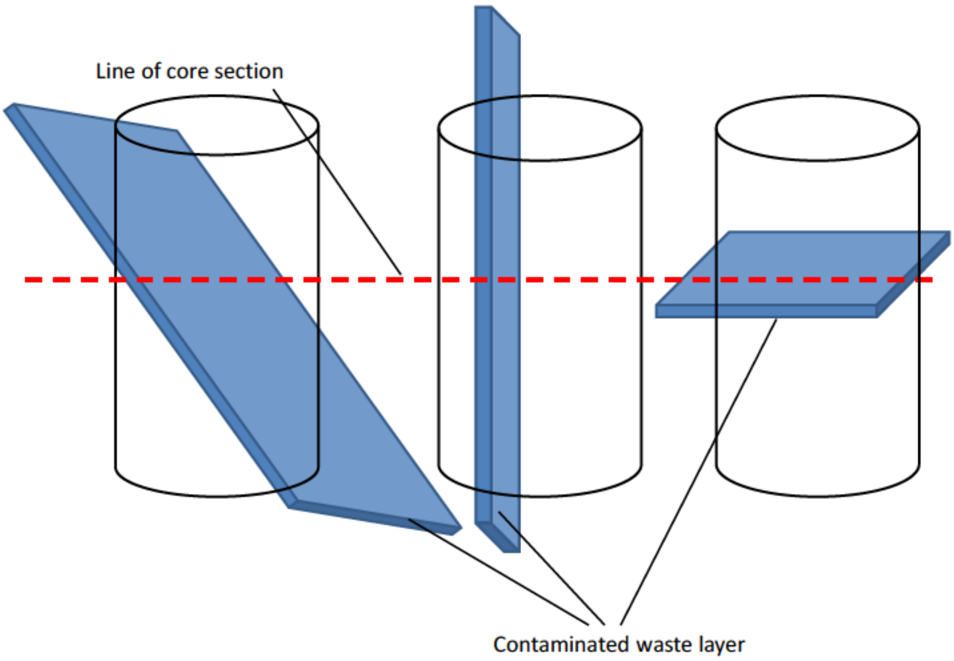
Exposed group	Dose (mSv/y) for different contamination penetration depths			
	1 cm	5 cm	15 cm	Uniform
Site investigator	$1.85\ 10^{-3}$	$3.70\ 10^{-4}$	$1.23\ 10^{-4}$	$4.63\ 10^{-5}$
Site occupant	2.25	1.31	$6.84\ 10^{-1}$	$2.89\ 10^{-1}$

Angle of exposure

1118. It is assumed that a site occupant is exposed to a semi-infinite plane source term, based on all contaminated waste being exposed with the contaminated surface layer uppermost. In practice, this represents the most conservative assumption. Assuming a point source exposure, the external dose falls by about an order of magnitude. Further sensitivity analysis is not undertaken.

1119. A drill core may penetrate waste lying in different configurations, potentially extracting a greater volume of the contaminated waste surface layer. However, fracturing the core to reveal the contaminated layer is unlikely to present a larger surface area, as illustrated below (Figure 31). Further consideration of the angle of the waste relative to the drill core is not undertaken here.

Figure 31. Angle of waste relative to drill core



Density of waste

1120. The external dose coefficients used in this assessment are from US Federal Guidance 12 (US EPA, 1993). This guidance document presents values in terms of radionuclide specific dose coefficients per unit volume (i.e. Sv s^{-1} per Bq m^{-3}). These have been converted to units of dose per unit mass (adjusted to give mSv h^{-1} per Bq kg^{-1}), based on a waste density of 1600 kg m^{-3} since this is the value used by US EPA for the calculated dose coefficients and US-EPA do not recommend scaling the data for other densities. This density relates to the density of soil and can be regarded as typical for a range of concrete masonry types (see Table 167). A typical value for concrete slabs is 2,400 kg m^{-3} (Dorf, 1996).

Table 167 Density of concrete

	Density (kg m^{-3})		Comments
	Low	High	
Dense aggregate block	1800	2100	
Lightweight aggregate blocks	650	1500	
Aircrete blocks	400	900	
Concrete bricks	1900	2100	Commons, Facing and Engineering Quality
		1400	Lightweight bricks

1121. Adjusting for other densities is not straight forward. Applying a simple linear scaling, the dose coefficient for dense concrete masonry (e.g. 2100 kg m^{-3}) or concrete slabs (2,400 kg m^{-3}) will be elevated (x1.31 and x1.5, respectively) and the dose coefficient for lightweight blocks (e.g. 1000 kg m^{-3}; x0.625) will be depressed relative to the base case. However, the coefficients for a source with finite thickness expressed in mean free paths are inversely proportional to density and the coefficients at lower photon energies are sensitive to elemental composition (US EPA, 1993). Simple scaling therefore results in an overestimate of dose at higher density. For these reasons, the recommendation is made in US Federal Guidance 12 not to scale the DCF's in relation to density.

E.7.4. Summary of key uncertainties

1122. The key uncertainties are discussed in relation to the limiting scenarios as these have a direct impact on the radiological capacity.

Recreational use

1123. The release of H-3 and C-14 gas from the capped site limits the radiological capacity of these radionuclides. This is based on a release rate to atmosphere of 5% of the inventory as gas per annum for both radionuclides and sensitivity to this parameter was illustrated in Table 145. Release of the total inventory to atmosphere in a single year would give rise to a peak dose of 300 µSv for C-14, a scenario which is not credible.

1124. External exposure to Ru-106 also limits the radiological capacity but the dose from the maximum capacity is insignificant (<1 10^{-13} µSv).

Borehole driller

1125. The duration and timing of exposure are the key uncertainties determining the dose to the borehole driller. Doubling the exposure period halves the radiological capacity and this would reduce the maximum inventory of only two radionuclides, Ra-226 and Th-232.

1126. As the time until intrusion increases the dose declines for the two illustrative inventories presented (Table 156), however the rate of decline varies with the assumed inventory. The dose falls by an order of magnitude over the first 200 years if the proportions in the national inventory are considered, but only by 30% if the inventory is based on ENRMF disposals to date. The impact of exposure timing is therefore dependent on the mix of radionuclides disposed and the proportion of the inventory with long half-lives and those supporting daughter ingrowth.

1127. Doses to the borehole driller are relatively insensitive to waste bulk density.

Smallholder

1128. The dose to the smallholder is sensitive to the dilution that is assumed for excavated waste. The mixture of hazardous and LLW is unlikely to support plant growth without substantial dilution (see paragraph 831). In the event that some of the excavated material is incorporated in soil surrounding and under a smallholders property a range of dilution factors is considered (from 0.0001 to 0.48). The radionuclides where a change to the dilution factor has an impact on radiological capacity are presented in Table 161. This shows that complete removal of the clean soil contribution to the dilution factor would reduce the radiological capacity of C-14, Sn-126 and Th-229 below 89.6 TBq. For C-14 the radiological capacity would then be about 81 TBq and for the others about 36 TBq.

1129. As the time until intrusion increases the dose declines for the two illustrative inventories presented (Table 162), however the rate of decline varies with the assumed inventory. The dose falls by an order of magnitude over the first 150 years for both inventories but over the next 2000 years the dose based on the national inventory declines by another order of magnitude, whereas it only decreases 25% if the inventory is based on ENRMF disposals to date. The impact of exposure timing is therefore dependent on the mix of radionuclides disposed and the proportion of the inventory with long half-lives and those supporting daughter ingrowth.

Groundwater

1130. 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 the ENRMF 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 limestone within the aquifer. The rate of infiltration to the landfill through the cap is also conservative (see paragraph 693).

1131. The application of peak dose output from the model to calculate radiological capacity is also conservative, The time to peak dose varies from 53 to 100,000 years and daughter ingrowth is calculated at the time of peak dose but is assumed to affect the same individual in dose calculations (see paragraph 1067). The impact of using a longer period, up to a million years was considered and although the dose from Th-232 and U-238 continues to rise due to ingrowth of daughter radionuclides, the projected dose using an illustrative inventory based on the national LLW inventory shows a dose at a million years of less 20 µSv y^{-1}.

1132. The depth of clay beneath the landfill used in the assessments (1.5 m) is less than that agreed with the EA for the western extension area. The dose is sensitive to the hydraulic conductivity of clay but not to the presence of the HDPE liner which is assumed to deteriorate over time (Table 150). The performance of the engineered clay barrier is also expected to be better than that assumed in the model which used a value similar to the clay underlying the landfill. The additional 1.5 m of clay and the lower hydraulic conductivity of the engineered clay will therefore apply to the majority of the radiological capacity at the site (99.7% currently unused).

1133. We have also shown that the radiological capacity of H-3 is not sensitive to assumptions concerning diffusion through the base of the landfill.

E.8. Tables of universal model parameters

1134. The following Tables list model parameters that are common to all models used in the assessments.

E.8.1. Radionuclide half-lives and decay constants

Table 168 Half-life 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	$3.01\ 10^{5}$	$2.30\ 10^{-6}$
Fe-55	2.74	$2.53\ 10^{-1}$
Co-60	5.27	$1.32\ 10^{-1}$
Ni-63	100.1	$6.92\ 10^{-3}$
Sr-90	28.8	$2.41\ 10^{-2}$
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	$1.66\ 10^{-3}$
Sb-125	2.8	$2.51\ 10^{-1}$
Sn-126	$2.30\ 10^{5}$	$3.01\ 10^{-6}$
I-129	$1.57\ 10^{7}$	$4.41\ 10^{-8}$
Ba-133	10.5	$6.59\ 10^{-2}$
Cs-134	2.1	$3.36\ 10^{-1}$
Cs-137	30.2	$2.30\ 10^{-2}$
Pm-147	2.6	$2.64\ 10^{-1}$
Eu-152	13.5	$5.12\ 10^{-2}$
Eu-154	8.6	$8.07\ 10^{-2}$
Eu-155	4.76	$1.46\ 10^{-1}$
Pb-210	22.2	$3.12\ 10^{-2}$
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-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	$2.34\ 10^{7}$	$2.96\ 10^{-8}$

Radionuclide	Half-life (y)	Decay constant (y^{-1})
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	$3.75\ 10^{5}$	$1.85\ 10^{-6}$
Am-241	432	$1.60\ 10^{-3}$
Cm-243	29.1	$2.38\ 10^{-2}$
Cm-244	18.1	$3.83\ 10^{-2}$

Note: Half-lives are taken from the LLWR radiological handbook (LLWR, 2011a) or from ICRP where radionuclides are not included in the LLWR assessment (ICRP, 1996).

E.8.2. Sorption Distribution coefficients

Table 169 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	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$
Cl-36	$3.00\ 10^{-4}$	$2.00\ 10^{-4}$
Fe-55	$2.20\ 10^{-1}$	$8.00\ 10^{-1}$
Co-60	$6.00\ 10^{-2}$	$1.00\ 10^{1}$
Ni-63	$4.00\ 10^{-1}$	$6.00\ 10^{-1}$
Sr-90	$1.30\ 10^{-2}$	$1.40\ 10^{-1}$
Nb-94	$1.60\ 10^{-1}$	7.60
Tc-99	$1.40\ 10^{-4}$	$1.90\ 10^{-1}$
Ru-106	$5.50\ 10^{-2}$	$4.00\ 10^{-1}$
Ag-108m	$9.00\ 10^{-2}$	$1.80\ 10^{-1}$
Sb-125	$4.50\ 10^{-2}$	$2.40\ 10^{-1}$
Sn-126	$1.30\ 10^{-1}$	$6.70\ 10^{-1}$
I-129	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$
Ba-133	$4.10\ 10^{-3}$	$4.00\ 10^{-2}$
Cs-134	$2.70\ 10^{-1}$	2.00
Cs-137	$2.70\ 10^{-1}$	2.00
Pm-147	$2.40\ 10^{-1}$	1.30
Eu-152	$2.40\ 10^{-1}$	7.80
Eu-154	$2.40\ 10^{-1}$	7.80
Eu-155	$2.40\ 10^{-1}$	7.80
Pb-210	$2.70\ 10^{-1}$	4.90
Ra-226	$4.90\ 10^{-1}$	9.00

Radionuclide	K_d soil ($m^3\ kg^{-1}$)	K_d clay ($m^3\ kg^{-1}$)
Ra-228	$4.90\ 10^{-1}$	9.00
Ac-227	$4.50\ 10^{-1}$	5.00
Th-229	3.00	$1.43\ 10^{1}$
Th-230	3.00	$1.43\ 10^{1}$
Th-232	3.00	$1.43\ 10^{1}$
Pa-231	$5.40\ 10^{-1}$	$1.00\ 10^{1}$
U-232	$3.30\ 10^{-2}$	6.00
U-233	$3.30\ 10^{-2}$	6.00
U-234	$3.30\ 10^{-2}$	6.00
U-235	$3.30\ 10^{-2}$	6.00
U-236	$3.30\ 10^{-2}$	6.00
U-238	$3.30\ 10^{-2}$	6.00
Np-237	$4.10\ 10^{-3}$	$4.60\ 10^{-2}$
Pu-238	$5.40\ 10^{-1}$	7.60
Pu-239	$5.40\ 10^{-1}$	7.60
Pu-240	$5.40\ 10^{-1}$	7.60
Pu-241	$5.40\ 10^{-1}$	7.60
Pu-242	$5.40\ 10^{-1}$	7.60
Am-241	2.00	3.20
Cm-243	$4.00\ 10^{-1}$	4.00
Cm-244	$4.00\ 10^{-1}$	4.00

Note:

Values from (Augean, 2009a) except for Cl-36 which was modified; the default value in SNIFFER is high and results in unrealistically low groundwater activity concentrations). The revised values are from the review presented in TecDoc 1616 (IAEA, 2009) which supports the IAEA handbook of parameter values (IAEA, 2010).

E.8.3. Dose Coefficients

Table 170 Radionuclide dose and attenuation coefficients

Radionuclide	Ingestion (Sv Bq^{-1})	Inhalation (Sv Bq^{-1})	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	$1.80\ 10^{-11}$	$2.60\ 10^{-10}$	0	0	0	0	0
C-14	$5.80\ 10^{-10}$	$5.80\ 10^{-9}$	$3.64\ 10^{-12}$	0	$9.02\ 10^{-7}$	0	$5.59\ 10^{1}$
Cl-36	$9.30\ 10^{-10}$	$7.30\ 10^{-9}$	$6.46\ 10^{-10}$	$1.10\ 10^{-11}$	$2.51\ 10^{-6}$	$5.37\ 10^{-7}$	$2.04\ 10^{1}$
Fe-55	$3.30\ 10^{-10}$	$7.70\ 10^{-10}$	0	$1.60\ 10^{-8}$	0	0	0
Co-60	$3.40\ 10^{-9}$	$3.10\ 10^{-8}$	$4.38\ 10^{-6}$	$1.30\ 10^{-7}$	$1.83\ 10^{-6}$	$2.85\ 10^{-8}$	$1.20\ 10^{1}$
Ni-63	$1.50\ 10^{-10}$	$1.30\ 10^{-9}$	0	0	$1.83\ 10^{-8}$	0	0
Sr-90	$3.07\ 10^{-8}$	$1.62\ 10^{-7}$	$6.65\ 10^{-9}$	$2.40\ 10^{-12}$	$5.14\ 10^{-6}$	$1.76\ 10^{-6}$	$1.85\ 10^{1}$
Nb-94	$1.70\ 10^{-9}$	$4.90\ 10^{-8}$	$2.62\ 10^{-6}$	$1.00\ 10^{-7}$	$2.17\ 10^{-6}$	$1.83\ 10^{-7}$	$1.38\ 10^{1}$
Tc-99	$6.40\ 10^{-10}$	$1.30\ 10^{-8}$	$3.39\ 10^{-11}$	$3.49\ 10^{-14}$	$1.60\ 10^{-6}$	$1.37\ 10^{-8}$	$3.85\ 10^{1}$
Ru-106	$7.00\ 10^{-9}$	$6.60\ 10^{-8}$	$3.49\ 10^{-7}$	$1.20\ 10^{-8}$	$2.85\ 10^{-6}$	$1.60\ 10^{-6}$	$1.47\ 10^{1}$
Ag-108m	$2.30\ 10^{-9}$	$3.70\ 10^{-8}$	$2.61\ 10^{-6}$	$1.28\ 10^{-7}$	$2.76\ 10^{-7}$	$1.15\ 10^{-7}$	$1.49\ 10^{1}$
Sb-125	$1.30\ 10^{-9}$	$1.30\ 10^{-8}$	$6.62\ 10^{-7}$	$3.51\ 10^{-8}$	$1.73\ 10^{-6}$	$6.61\ 10^{-8}$	$1.54\ 10^{1}$
Sn-126	$5.04\ 10^{-9}$	$2.84\ 10^{-8}$	$6.87\ 10^{-7}$	$1.33\ 10^{-7}$	$4.54\ 10^{-6}$	$1.43\ 10^{-6}$	$1.46\ 10^{1}$
I-129	$1.10\ 10^{-7}$	$3.60\ 10^{-8}$	$3.50\ 10^{-9}$	$9.70\ 10^{-9}$	$6.51\ 10^{-7}$	0	$1.31\ 10^{2}$
Ba-133	$1.50\ 10^{-9}$	$1.00\ 10^{-8}$	$5.35\ 10^{-7}$	0	0	0	$1.79\ 10^{1}$
Cs-134	$1.90\ 10^{-8}$	$2.00\ 10^{-8}$	$2.56\ 10^{-6}$	$8.80\ 10^{-8}$	$1.83\ 10^{-6}$	$3.08\ 10^{-7}$	$1.42\ 10^{1}$
Cs-137	$1.30\ 10^{-8}$	$3.90\ 10^{-8}$	$9.20\ 10^{-7}$	$3.31\ 10^{-8}$	$2.54\ 10^{-6}$	$3.92\ 10^{-7}$	$1.45\ 10^{1}$
Pm-147	$2.60\ 10^{-10}$	$5.00\ 10^{-9}$	$1.35\ 10^{-11}$	$4.90\ 10^{-13}$	$1.26\ 10^{-6}$	$4.11\ 10^{-10}$	$3.73\ 10^{1}$
Eu-152	$1.40\ 10^{-9}$	$4.20\ 10^{-8}$	$1.89\ 10^{-6}$	$1.18\ 10^{-7}$	$1.60\ 10^{-6}$	$1.71\ 10^{-7}$	$1.30\ 10^{1}$

Radionuclide	Ingestion ($Sv\ Bq^{-1}$)	Inhalation ($Sv\ Bq^{-1}$)	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})*
Eu-154	$2.00\ 10^{-9}$	$5.30\ 10^{-8}$	$2.08\ 10^{-6}$	$9.02\ 10^{-8}$	$3.42\ 10^{-6}$	$3.77\ 10^{-7}$	$1.29\ 10^{1}$
Eu-155	$3.20\ 10^{-10}$	$6.90\ 10^{-9}$	$4.92\ 10^{-8}$	$1.77\ 10^{-8}$	$8.68\ 10^{-7}$	$3.20\ 10^{-10}$	$3.37\ 10^{1}$
Pb-210	$1.89\ 10^{-6}$	$9.99\ 10^{-6}$	$1.65\ 10^{-9}$	$8.30\ 10^{-9}$	$2.63\ 10^{-6}$	$8.45\ 10^{-7}$	$1.60\ 10^{1}$
Ra-226	$2.17\ 10^{-6}$	$1.95\ 10^{-5}$	$3.03\ 10^{-6}$	$1.64\ 10^{-7}$	$5.89\ 10^{-6}$	$1.66\ 10^{-6}$	$1.43\ 10^{1}$
Ra-228	$8.34\ 10^{-7}$	$5.96\ 10^{-5}$	$4.37\ 10^{-6}$	$5.26\ 10^{-8}$	$3.08\ 10^{-6}$	$7.19\ 10^{-7}$	$1.03\ 10^{1}$
Ac-227	$1.21\ 10^{-6}$	$5.69\ 10^{-4}$	$4.44\ 10^{-7}$	$3.81\ 10^{-8}$	$6.59\ 10^{-6}$	$2.00\ 10^{-6}$	$1.69\ 10^{1}$
Th-229	$6.13\ 10^{-7}$	$2.56\ 10^{-4}$	$4.33\ 10^{-7}$	$7.31\ 10^{-8}$	$8.56\ 10^{-6}$	$1.36\ 10^{-6}$	$1.46\ 10^{1}$
Th-230	$2.10\ 10^{-7}$	$1.00\ 10^{-4}$	$3.27\ 10^{-10}$	$3.83\ 10^{-9}$	$1.04\ 10^{-7}$	0	$2.93\ 10^{1}$
Th-232	$1.06\ 10^{-6}$	$1.70\ 10^{-4}$	$4.37\ 10^{-6}$	$2.20\ 10^{-9}$	$3.08\ 10^{-8}$	0	$1.27\ 10^{1}$
Pa-231	$7.10\ 10^{-7}$	$1.40\ 10^{-4}$	$5.15\ 10^{-8}$	$6.27\ 10^{-8}$	$1.48\ 10^{-7}$	$5.14\ 10^{-9}$	$1.91\ 10^{1}$
U-232	$3.30\ 10^{-7}$	$3.70\ 10^{-5}$	$2.44\ 10^{-10}$	$9.36\ 10^{-8}$	$6.38\ 10^{-8}$	$1.22\ 10^{-6}$	$2.93\ 10^{1}$
U-233	$5.10\ 10^{-8}$	$9.60\ 10^{-6}$	$3.78\ 10^{-10}$	$1.70\ 10^{-9}$	$5.25\ 10^{-7}$	0	$2.29\ 10^{1}$
U-234	$4.90\ 10^{-8}$	$9.40\ 10^{-6}$	$1.09\ 10^{-10}$	$2.70\ 10^{-9}$	$7.42\ 10^{-9}$	0	$3.58\ 10^{1}$
U-235	$4.73\ 10^{-8}$	$8.50\ 10^{-6}$	$2.05\ 10^{-7}$	$5.31\ 10^{-8}$	$2.52\ 10^{-6}$	$1.09\ 10^{-8}$	$2.32\ 10^{1}$
U-236	$4.70\ 10^{-8}$	$8.70\ 10^{-6}$	$5.81\ 10^{-11}$	$3.55\ 10^{-9}$	$4.57\ 10^{-9}$	0	$3.16\ 10^{1}$
U-238	$4.84\ 10^{-8}$	$8.01\ 10^{-6}$	$3.58\ 10^{-8}$	$9.23\ 10^{-9}$	$3.82\ 10^{-6}$	$1.26\ 10^{-6}$	$1.60\ 10^{1}$
Np-237	$1.11\ 10^{-7}$	$5.00\ 10^{-5}$	$2.97\ 10^{-7}$	$3.20\ 10^{-8}$	$3.46\ 10^{-6}$	$9.93\ 10^{-8}$	$3.16\ 10^{1}$
Pu-238	$2.30\ 10^{-7}$	$1.10\ 10^{-4}$	$4.09\ 10^{-11}$	$2.70\ 10^{-9}$	0	0	$3.73\ 10^{1}$
Pu-239	$2.50\ 10^{-7}$	$1.20\ 10^{-4}$	$7.98\ 10^{-11}$	$1.00\ 10^{-9}$	$4.34\ 10^{-10}$	0	$2.18\ 10^{1}$
Pu-240	$2.50\ 10^{-7}$	$1.20\ 10^{-4}$	$3.96\ 10^{-11}$	$2.60\ 10^{-9}$	0	0	$4.44\ 10^{1}$
Pu-241	$4.80\ 10^{-9}$	$2.30\ 10^{-6}$	$1.60\ 10^{-12}$	$3.30\ 10^{-12}$	0	0	$2.96\ 10^{1}$
Pu-242	$2.40\ 10^{-7}$	$1.10\ 10^{-4}$	$3.46\ 10^{-11}$	$3.07\ 10^{-9}$	0	0	$5.58\ 10^{1}$
Am-241	$2.00\ 10^{-7}$	$9.60\ 10^{-5}$	$1.18\ 10^{-8}$	$1.70\ 10^{-8}$	$5.48\ 10^{-8}$	0	$5.37\ 10^{1}$
Cm-243	$1.50\ 10^{-7}$	$6.90\ 10^{-5}$	$1.58\ 10^{-7}$	$2.75\ 10^{-8}$	$1.94\ 10^{-6}$	$3.42\ 10^{-8}$	$2.29\ 10^{1}$

Radionuclide	Ingestion ($Sv\ Bq^{-1}$)	Inhalation ($Sv\ Bq^{-1}$)	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})*
Cm-244	$1.20\ 10^{-7}$	$5.70\ 10^{-5}$	$3.40\ 10^{-11}$	$2.20\ 10^{-9}$	0	0	$3.52\ 10^{2}$

*Attenuation coefficient for soil from (SNIFFER, 2006) taken from Hung (2000). . The mass attenuation coefficient would be this divided by the soil density.

Table 171 Dose coefficients for depth of contamination of 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.8.4. Crop and animal transfer parameters

Table 172 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	5.00	5.00
C	$1.60\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$
Cl	$8.80\ 10^{-2}$	5.00	5.00	5.00
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	$5.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$	$2.00\ 10^{-2}$
Sr	$1.20\ 10^{-1}$	3.00	$9.00\ 10^{-2}$	3.00
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	$8.80\ 10^{-2}$	$2.70\ 10^{-4}$	$1.30\ 10^{-3}$	$1.50\ 10^{-1}$
Sb	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$
Sn	$2.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$2.00\ 10^{-1}$
I	$2.80\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$	$1.00\ 10^{-1}$
Ba	$1.00\ 10^{-1}$	$4.00\ 10^{-3}$	$1.00\ 10^{-2}$	$4.00\ 10^{-2}$
Cs	$2.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$	$3.00\ 10^{-2}$
Pm	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$
Eu	$4.80\ 10^{-2}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$	$3.00\ 10^{-3}$
Pb	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$	$1.00\ 10^{-2}$
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	$3.00\ 10^{-5}$	$1.00\ 10^{-4}$	$1.00\ 10^{-3}$	$1.00\ 10^{-3}$

Values from (Augean, 2009a)

Table 173 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
Cl	4.30 10^{-2}	1.70 10^{-2}	5.00 10^{-2}
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}
Sr	8.00 10^{-3}	3.00 10^{-3}	6.00 10^{-2}
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}
Sb	4.00 10^{-5}	2.50 10^{-5}	1.00 10^{-1}
Sn	1.90 10^{-3}	1.00 10^{-3}	1.00
I	4.00 10^{-2}	1.00 10^{-2}	3.00 10^{-2}
Ba	5.00 10^{-4}	5.00 10^{-4}	4.00 10^{-3}
Cs	5.00 10^{-2}	7.90 10^{-3}	2.00
Pm	5.00 10^{-3}	2.00 10^{-5}	3.00 10^{-2}
Eu	4.70 10^{-4}	5.00 10^{-5}	3.00 10^{-2}
Pb	4.00 10^{-4}	3.00 10^{-4}	3.00 10^{-1}
Ra	9.00 10^{-4}	1.30 10^{-3}	5.00 10^{-2}
Ac	1.60 10^{-4}	4.00 10^{-7}	2.40 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^{-5}	1.50 10^{-6}	3.00 10^{-2}
Cm	1.00 10^{-4}	1.00 10^{-6}	3.00 10^{-2}

* Ac fish value from IAEA 472 (IAEA, 2010)

Other values from (Augean, 2009a)

Appendix F. Confidence building in the Goldsim groundwater model

F.1. Description of the GNU Octave model

1135. GNU Octave (an open source multi-purpose maths analysis programme) comes with a numerical solver for a set of coupled linear ordinary differential equations (lsode).

$$\frac{d\bar{x}}{dt} = f(\bar{x}, t)$$

Where: $\bar{x}$ is a vector with elements x_1, x_2, … x_n.

1136. Our confidence building model is based on some simplified assumptions:

- Cap and liner are not taken into account. Flow rates (sub-vertical q_leachate and subhorizontal q_aquifer) are constant in time.
- One radionuclide (and if applicable its daughter) is modelled at the time.

1137. It is possible to refine this model further, e.g. by using time dependencies, adding unsaturated zone and aquifer pathway, etc. However, this was not within the scope of the confidence building exercise. Parameter values have been copied from the GoldSim model.

1138. The flow rates (*q*) driving advective transport are:

$$q_{leachate} = A_{basal} \cdot H_{Clay}$$

$$q_{aquifer} = W_{Aquifer} \cdot d_{Aquifer} \cdot K_{limestone} \cdot Grad$$

with:

- A_{basal} the basal area of the landfill;
- H_{clay} the hydraulic conductivity of clay;
- $W_{Aquifer}$ the width of the aquifer strip;
- $d_{Aquifer}$ the height of the aquifer;
- $K_{limestone}$ the hydraulic conductivity of limestone; and,
- $Grad$ the hydraulic gradient.

F.2. Cl-36 model

1139. The Cl-36 activity in the waste cell, $A_{cell}(^{36}Cl)$, varies according to the following equation:

$$\frac{dA_{cell}(^{36}Cl)}{dt} = -\left(r_{cell}(^{36}Cl) + \lambda(^{36}Cl)\right) \cdot A_{cell}(^{36}Cl)$$

with loss terms for leaching and radioactive decay.

1140. Leaching is described by:

$$-r_{cell}\left({}^{36}Cl\right)\cdot A_{cell}\left({}^{36}Cl\right) = -\frac{q_{leachate}}{V_{cell}\cdot\varepsilon_{cell} + V_{cell}\cdot\rho_{soil}\cdot f_{soil,cell}\cdot k_{d,soil}(Cl)}\cdot A_{cell}\left({}^{36}Cl\right)$$

with:

- $r_{cell}({}^{36}Cl)$ the waste cell leaching rate;
- V_{cell} the volume of the waste cells (m^3);
- ε_{cell} the porosity of the waste cells (dimensionless);
- ρ_{soil} the density of soil (kg m^{-3});
- $f_{soil,cell}$ the fraction of soil in the waste cells (dimensionless); and,
- $k_{d,soil(Cl)}$ the distribution coefficient of soil relative to water for chlorine (dimensionless).

1141. Radioactive decay is described by:

$$-\lambda\left({}^{36}Cl\right)\cdot A_{cell}\left({}^{36}Cl\right) = -\frac{ln(2)}{t_{1/2}\left({}^{36}Cl\right)}\cdot A_{cell}\left({}^{36}Cl\right)$$

with:

- $\lambda({}^{36}Cl)$ the decay constant of Cl-36; and,
- $t_{1/2}({}^{36}Cl)$ the half-life of Cl-36.

1142. The Cl-36 activity in the clay barrier (A_B) varies according to the following equation:

$$\frac{dA_B\left({}^{36}Cl\right)}{dt} = r_{cell}\left({}^{36}Cl\right)\cdot A_{cell}\left({}^{36}Cl\right) - \left(r_B\left({}^{36}Cl\right) + \lambda\left({}^{36}Cl\right)\right)\cdot A_B\left({}^{36}Cl\right)$$

with source term:

$$r_{cell}\left({}^{36}Cl\right)\cdot A_{cell}\left({}^{36}Cl\right) = \frac{q_{leachate}}{V_{cell}\cdot\varepsilon_{cell} + V_{cell}\cdot\rho_{soil}\cdot f_{soil,cell}\cdot k_{d,soil}(Cl)}\cdot A_{cell}\left({}^{36}Cl\right)$$

and, with loss terms (B = Barrier):

$$-r_B\left({}^{36}Cl\right)\cdot A_B\left({}^{36}Cl\right) = -\frac{q_{leachate}}{V_B\cdot\varepsilon_B + V_B\cdot\rho_{clay}\cdot k_{d,clay}(Cl)}\cdot A_B\left({}^{36}Cl\right)$$

$$-\lambda\left({}^{36}Cl\right)\cdot A_B\left({}^{36}Cl\right) = -\frac{ln(2)}{t_{1/2}\left({}^{36}Cl\right)}\cdot A_B\left({}^{36}Cl\right)$$

1143. The Cl-36 activity in the unsaturated zone (A_{Unsat}) varies according to the following equation:

$$\frac{dA_{Unsat}(^{36}Cl)}{dt} = r_B(^{36}Cl) \cdot A_B(^{36}Cl) - \left(r_{Unsat}(^{36}Cl) + \lambda(^{36}Cl)\right) \cdot A_{Unsat}(^{36}Cl)$$

with source term:

$$r_B(^{36}Cl) \cdot A_B(^{36}Cl) = \frac{q_{leachate}}{V_B \cdot \varepsilon_B + V_B \cdot \rho_{clay} \cdot k_{d,clay}(Cl)} \cdot A_B(^{36}Cl)$$

and, with loss terms:

$$-r_{Unsat}(^{36}Cl) \cdot A_{Unsat}(^{36}Cl) = -\frac{q_{leachate}}{V_{Unsat} \cdot \varepsilon_{Unsat}} \cdot A_{Unsat}(^{36}Cl)$$

$$-\lambda(^{36}Cl) \cdot A_{Unsat}(^{36}Cl) = -\frac{ln(2)}{t_{1/2}(^{36}Cl)} \cdot A_{Unsat}(^{36}Cl)$$

1144. The Cl-36 activity in the aquifer ($A_{aquifer}$) varies according to the following equation:

$$\frac{dA_{aquifer}(^{36}Cl)}{dt} = r_{Unsat}(^{36}Cl) \cdot A_{Unsat}(^{36}Cl) - \left(r_{aquifer}(^{36}Cl) + \lambda(^{36}Cl)\right) \cdot A_{aquifer}(^{36}Cl)$$

with source term:

$$r_{Unsat}(^{36}Cl) \cdot A_{Unsat}(^{36}Cl) = \frac{q_{leachate}}{V_{Unsat} \cdot \varepsilon_{Unsat}} \cdot A_{Unsat}(^{36}Cl)$$

and, with loss terms:

$$-r_{aquifer}(^{36}Cl) \cdot A_{aquifer}(^{36}Cl) = -\frac{q_{aquifer}}{V_{aquifer} \cdot \varepsilon_{aquifer}} \cdot A_{aquifer}(^{36}Cl)$$

$$-\lambda(^{36}Cl) \cdot A_{aquifer}(^{36}Cl) = -\frac{ln(2)}{t_{1/2}(^{36}Cl)} \cdot A_{aquifer}(^{36}Cl)$$

1145. Boundary conditions at t=0 are:

- 1 MBq of Cl-36 in the waste cell;
- 0 MBq of Cl-36 in the barrier; and,
- 0 MBq of Cl-36 is the aquifer.

1146. The model is run in steps of 10 years up to 1,000,000 years. After the model run activity values can be translated into activity concentrations in the water fractions using the following equations:

$$C_{cell}(^{36}Cl) = \frac{A_{cell}(^{36}Cl)}{V_{cell} \cdot \varepsilon_{cell} + V_{cell} \cdot \rho_{soil} \cdot f_{soil,cell} \cdot k_{d,soil}(Cl)}$$

$$C_B(^{36}Cl) = \frac{A_B(^{36}Cl)}{V_B \cdot \varepsilon_B + V_B \cdot \rho_{clay} \cdot k_{d,clay}(Cl)}$$

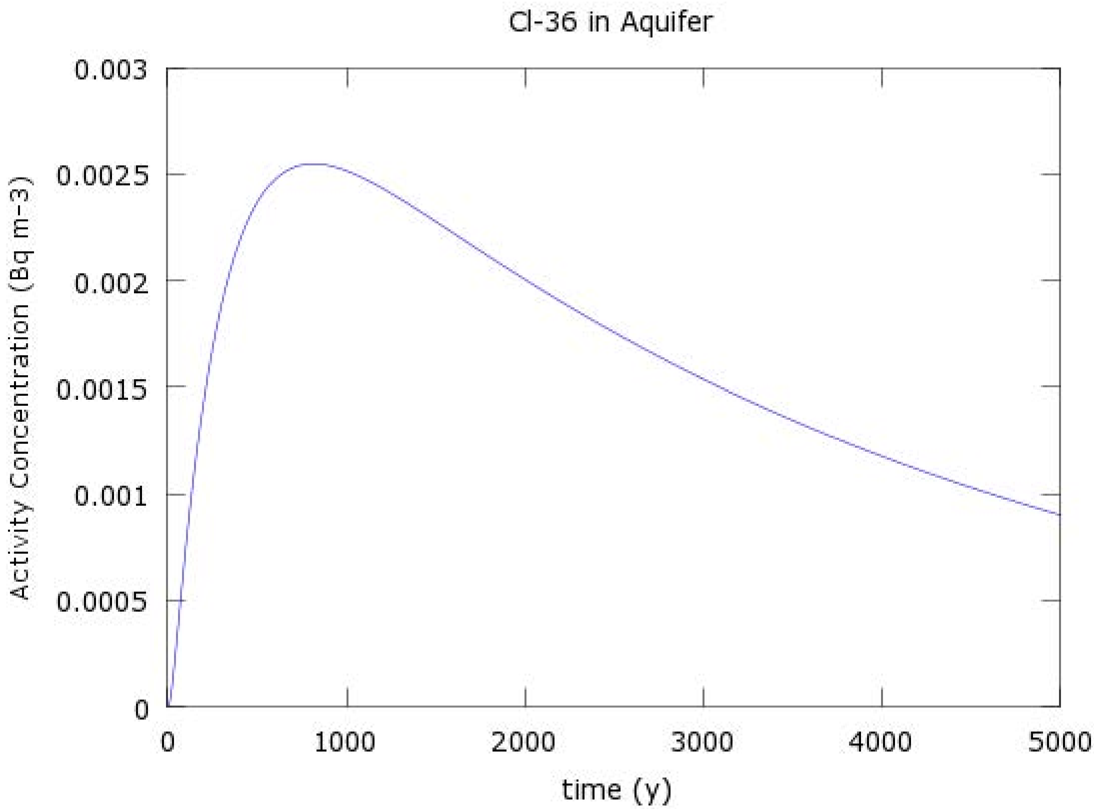
$$C_{Unsat}\left({}^{36}Cl\right) = \frac{A_{Unsat}\left({}^{36}Cl\right)}{V_{Unsat} \cdot \varepsilon_{Unsat}}$$

$$C_{aquifer}\left({}^{36}Cl\right) = \frac{A_{aquifer}\left({}^{36}Cl\right)}{V_{aquifer} \cdot \varepsilon_{aquifer}}$$

F.2.1. Results and comparison with GoldSim model

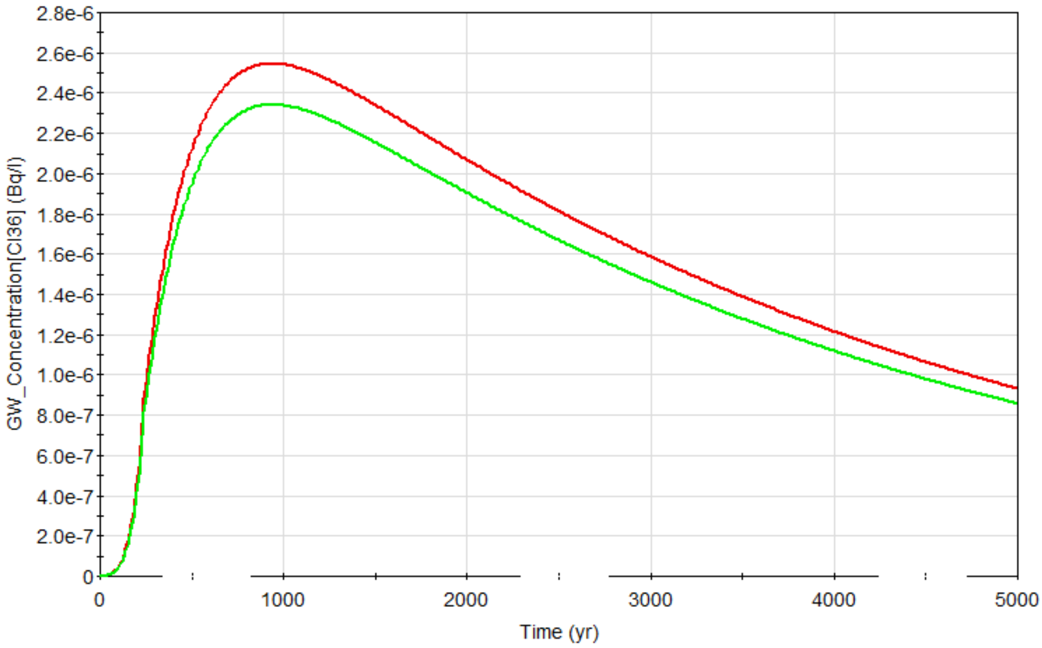
1147. Figure 32 shows the result of the GNU Octave model and Figure 33 shows the corresponding result of the GoldSim model. Note that the units in the y-axes for the two graphs are different.

Figure 32. GNU Octave estimation of Cl-36 activity concentration in groundwater below the landfill



1148. The GNU Octave model does not take the cap and basal HDPE liner into account and therefore leaching is assumed to start immediately; the GoldSim model exhibits a delay of about 150 years before leaching starts. This leads to the peak concentration in Goldsim appearing about 150 years later. Because of the long half-life of Cl-36, the peak value is very similar.

Figure 33. Goldsim estimation of Cl-36 activity concentration in groundwater below the landfill (red) and at an abstraction point at the boundary of the site (green)



1149. Scoping calculations were done in Excel based on simple assumptions and parameter values from the GoldSim model.

$$V_{water} = \frac{V_{site}}{\varepsilon_{waste} \cdot f_{waste} + \varepsilon_{soil} \cdot f_{soil}}$$

$$M_{soil} = \rho_{soil} \cdot V_{site} \cdot f_{soil}$$

$$C_{leachate}(^{36}Cl) = \frac{A(^{36}Cl)}{V_{water} + M_{soil} \cdot k_{d,soil}(Cl)}$$

$$C_{gw}(^{36}Cl) = \frac{q_{leachate}}{q_{aquifer}} \cdot C_{leachate}(^{36}Cl)$$

1150. The leachate flux ($q_{leachate}$) is 309.6 m^3 y^{-1} around 1000 years after closure, which is about the time when the peak concentration appears in groundwater. Based on that assumption the equilibrium activity concentration of Cl-36 in groundwater would be 3.15 10^{-3} Bq m^{-3}, slightly higher than the result from either Goldsim or GNU Octave. This is considered to be good agreement. The aquifer flux ($q_{aquifer}$) is 84,059 m^3 y^{-1}.

F.3. Ra-226 model

1151. The main difference between Cl-36 and Ra-226 is the ingrowth of Pb-210. This process is included in the model. Equations are described for nuclei count (N) rather than activity for each radionuclide k.

$$A(k) = \lambda(k) \cdot N(k)$$

1152. The Ra-226 and Pb-210 activity in the waste cell ($N_{cell}(^{226}Ra)$ and $N_{cell}(^{210}Pb)$) varies according to the following equation:

$$\frac{dN_{cell}(^{226}Ra)}{dt} = -\left(r_{cell}(^{226}Ra) + \lambda(^{226}Ra)\right) \cdot N_{cell}(^{226}Ra)$$

$$\frac{dN_{cell}(^{210}Pb)}{dt} = \lambda(^{226}Ra) \cdot N_{cell}(^{226}Ra) - \left(r_{cell}(^{210}Pb) + \lambda(^{210}Pb)\right) \cdot N_{cell}(^{210}Pb)$$

with loss terms for leaching and radioactive decay and a source term for ingrowth of Pb-210.

1153. Leaching of Ra-226 and Pb-210 is described by:

$$-r_{cell}(^{226}Ra) \cdot N_{cell}(^{226}Ra) = -\frac{q_{leachate}}{V_{cell} \cdot \varepsilon_{cell} + V_{cell} \cdot \rho_{soil} \cdot f_{soil,cell} \cdot k_{d,soil}(Ra)} \cdot N_{cell}(^{226}Ra)$$

$$-r_{cell}(^{210}Pb) \cdot N_{cell}(^{210}Pb) = -\frac{q_{leachate}}{V_{cell} \cdot \varepsilon_{cell} + V_{cell} \cdot \rho_{soil} \cdot f_{soil,cell} \cdot k_{d,soil}(Pb)} \cdot N_{cell}(^{210}Pb)$$

with:

- r_{wc} — the leaching rate ($r_{cell}(^{226}Ra)$ or $r_{cell}(^{210}Pb)$)
- V_{cell} — the volume of the waste cells
- ε_{cell} — the porosity of the waste cells
- ρ_{soil} — the density of soil
- $f_{soil,wc}$ — the fraction of soil in the waste cells
- $k_{d,soil}$ — the distribution coefficient of soil relative to water for radium/lead ($k_{d,soil}(Ra)$ or $k_{d,soil}(Pb)$)

1154. Radioactive decay of Ra-226 and Pb-210 is described by:

$$-\lambda(^{226}Ra) \cdot N_{cell}(^{226}Ra) = -\frac{ln(2)}{t_{1/2}(^{226}Ra)} \cdot N_{cell}(^{226}Ra)$$

$$-\lambda(^{210}Pb) \cdot N_{cell}(^{210}Pb) = -\frac{ln(2)}{t_{1/2}(^{210}Pb)} \cdot N_{cell}(^{210}Pb)$$

with:

- $t_{1/2}$ — the half-life ($t_{1/2}(^{226}Ra)$ or $t_{1/2}(^{210}Pb)$)

1155. Ingrowth of Pb-210 from radioactive decay of Ra-226 is described by:

$$\lambda(^{226}Ra) \cdot N_{cell}(^{226}Ra) = \frac{ln(2)}{t_{1/2}(^{226}Ra)} \cdot N_{cell}(^{226}Ra)$$

1156. The Ra-226 and Pb-210 activity in the clay barrier ($N_B(^{226}Ra)$, $N_B(^{210}Pb)$) varies according to the following equation:

$$\frac{dN_B(^{226}Ra)}{dt} = r_{cell}(^{226}Ra) \cdot N_{cell}(^{226}Ra) - \left(r_B(^{226}Ra) + \lambda(^{226}Ra)\right) \cdot N_B(^{226}Ra)$$

$$\frac{dN_B(^{210}Pb)}{dt} = r_{cell}(^{210}Pb) \cdot N_{cell}(^{210}Pb) + \lambda(^{226}Ra) \cdot N_B(^{226}Ra) - \left(r_B(^{210}Pb) + \lambda(^{210}Pb)\right) \cdot N_B(^{210}Pb)$$

with source, loss and ingrowth terms are defined similar to above.

1157. The Ra-226 and Pb-210 activity in the unsaturated zone ($N_{Unsat}(^{226}Ra)$, $N_{Unsat}(^{210}Pb)$) varies according to the following equation:

$$\frac{dN_{Unsat}(^{226}Ra)}{dt} = r_B(^{226}Ra) \cdot N_B(^{226}Ra) - \left(r_{Unsat}(^{226}Ra) + \lambda(^{226}Ra)\right) \cdot N_{Unsat}(^{226}Ra)$$

$$\frac{dN_{Unsat}(^{210}Pb)}{dt} = r_B(^{210}Pb) \cdot N_B(^{210}Pb) + \lambda(^{226}Ra) \cdot N_{Unsat}(^{226}Ra) - \left(r_{Unsat}(^{210}Pb) + \lambda(^{210}Pb)\right) \cdot N_{Unsat}(^{210}Pb)$$

with source, loss and ingrowth terms are defined similar to above.

1158. The Ra-226 and Pb-210 activity in the aquifer ($N_{aquifer}(^{226}Ra)$ and $N_{aquifer}(^{210}Pb)$) varies according to the following equation:

$$\frac{dN_{aquifer}(^{226}Ra)}{dt} = r_{Unsat}(^{226}Ra) \cdot N_{Unsat}(^{226}Ra) - \left(r_{aquifer}(^{226}Ra) + \lambda(^{226}Ra)\right) \cdot N_{aquifer}(^{226}Ra)$$

$$\frac{dN_{aquifer}(^{210}Pb)}{dt} = r_{Unsat}(^{210}Pb) \cdot N_{Unsat}(^{210}Pb) + \lambda(^{226}Ra) \cdot N_{aquifer}(^{226}Ra) - \left(r_{aquifer}(^{210}Pb) + \lambda(^{210}Pb)\right) \cdot N_{aquifer}(^{210}Pb)$$

with source, loss and ingrowth terms are defined similar to above.

1159. Boundary conditions at t=0 are:

- 1 MBq of Ra-226 and 0 MBq of Pb-210 in the waste cell;
- 0 MBq of Ra-226 and 0 MBq of Pb-210 in the barrier; and,
- 0 MBq of Ra-226 and 0 MBq of Pb-210 is the aquifer.

1160. The model is run in steps of 10 years up to 1,000,000 years. After the model run activity values can be translated into activity concentrations in the water fractions.

$$C_{cell}\left({}^{226}Ra\right) = \frac{N_{cell}\left({}^{226}Ra\right) \cdot \lambda\left({}^{226}Ra\right)}{V_{cell} \cdot \varepsilon_{cell} + V_{cell} \cdot \rho_{soil} \cdot f_{soil,cell} \cdot k_{d,soil}(Ra)}$$

$$C_{B}\left({}^{226}Ra\right) = \frac{N_{B}\left({}^{226}Ra\right) \cdot \lambda\left({}^{226}Ra\right)}{V_{B} \cdot \varepsilon_{B} + V_{B} \cdot \rho_{clay} \cdot k_{d,clay}(Ra)}$$

$$C_{Unsat}\left({}^{226}Ra\right) = \frac{N_{Unsat}\left({}^{226}Ra\right) \cdot \lambda\left({}^{226}Ra\right)}{V_{Unsat} \cdot \varepsilon_{Unsat}}$$

$$C_{aquifer}\left({}^{226}Ra\right) = \frac{N_{aquifer}\left({}^{226}Ra\right) \cdot \lambda\left({}^{226}Ra\right)}{V_{aquifer} \cdot \varepsilon_{aquifer}}$$

$$C_{cell}\left({}^{210}Pb\right) = \frac{N_{cell}\left({}^{210}Pb\right) \cdot \lambda\left({}^{210}Pb\right)}{V_{cell} \cdot \varepsilon_{cell} + V_{cell} \cdot \rho_{soil} \cdot f_{soil,cell} \cdot k_{d,soil}(Pb)}$$

$$C_{B}\left({}^{210}Pb\right) = \frac{N_{B}\left({}^{210}Pb\right) \cdot \lambda\left({}^{210}Pb\right)}{V_{B} \cdot \varepsilon_{B} + V_{B} \cdot \rho_{clay} \cdot k_{d,clay}(Pb)}$$

$$C_{Unsat}\left({}^{210}Pb\right) = \frac{N_{Unsat}\left({}^{210}Pb\right) \cdot \lambda\left({}^{210}Pb\right)}{V_{Unsat} \cdot \varepsilon_{Unsat}}$$

$$C_{aquifer}\left({}^{210}Pb\right) = \frac{N_{aquifer}\left({}^{210}Pb\right) \cdot \lambda\left({}^{210}Pb\right)}{V_{aquifer} \cdot \varepsilon_{aquifer}}$$

F.3.1. Results and comparison with GoldSim model

1161. Figure 34 and Figure 35 show the modelling results for Ra-226 and its daughter Pb-210 using the GNU Octave model and the GoldSim model. Note that the units in the y-axes for the two graphs are different. Because the GNU Octave model doesn't take the cap and basal PE liner into account the concentration peaks appear about 150 years earlier than in the GoldSim model. As a consequence the peak concentrations of Ra-226 and Pb-210, calculated in GNU Octave, are slightly higher than those calculated in the GoldSim model.

1162. A similar model for Th-232 and its daughter Ra-228 shows good agreement (see Figure 36 and Figure 37). Note that the units in the y-axes for the two graphs are different.

Figure 34. GNU Octave estimation of Ra-226 (lower curve) and daughter Pb-210 (upper curve) activity concentration in groundwater below the landfill

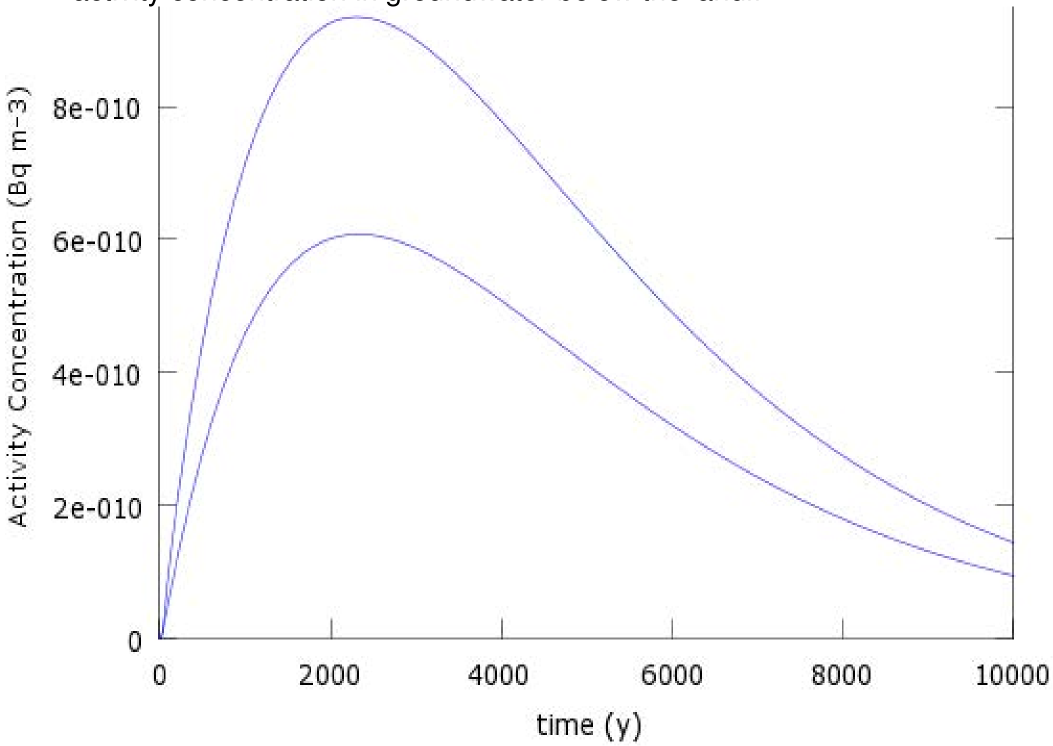


Figure 35. Goldsim estimation of Ra-226 (Red) and daughter Pb-210 (Green) activity concentration in groundwater below the landfill

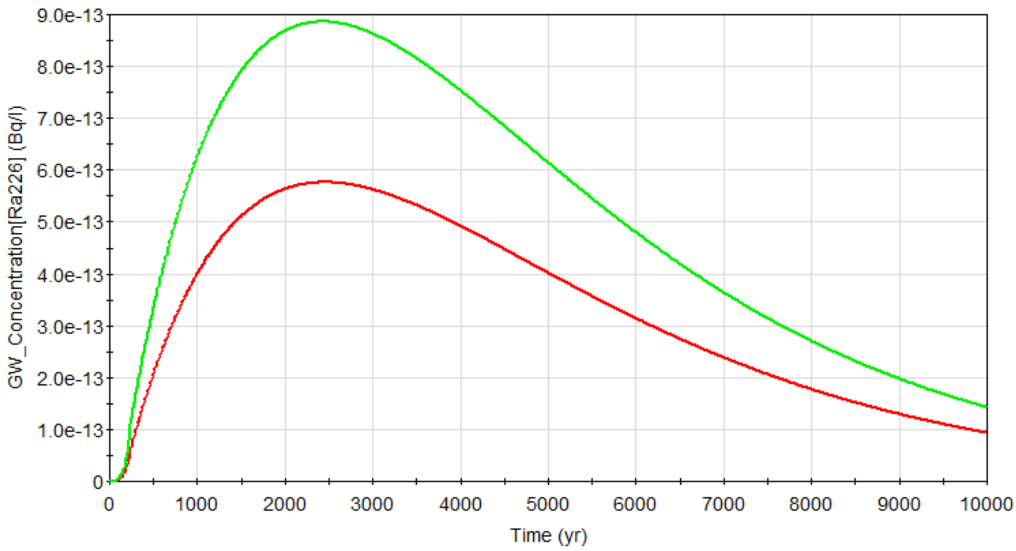


Figure 36. GNU Octave estimation of Th232 (lower curve) and daughter Ra-228 (upper curve) activity concentration in groundwater below the landfill

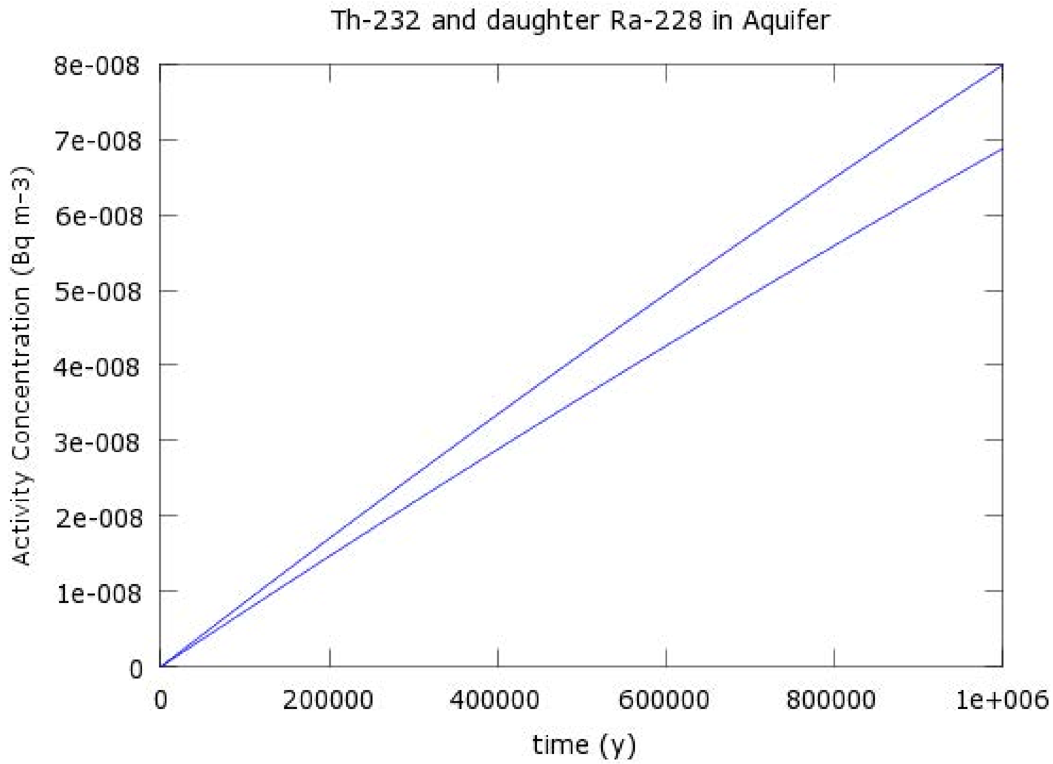
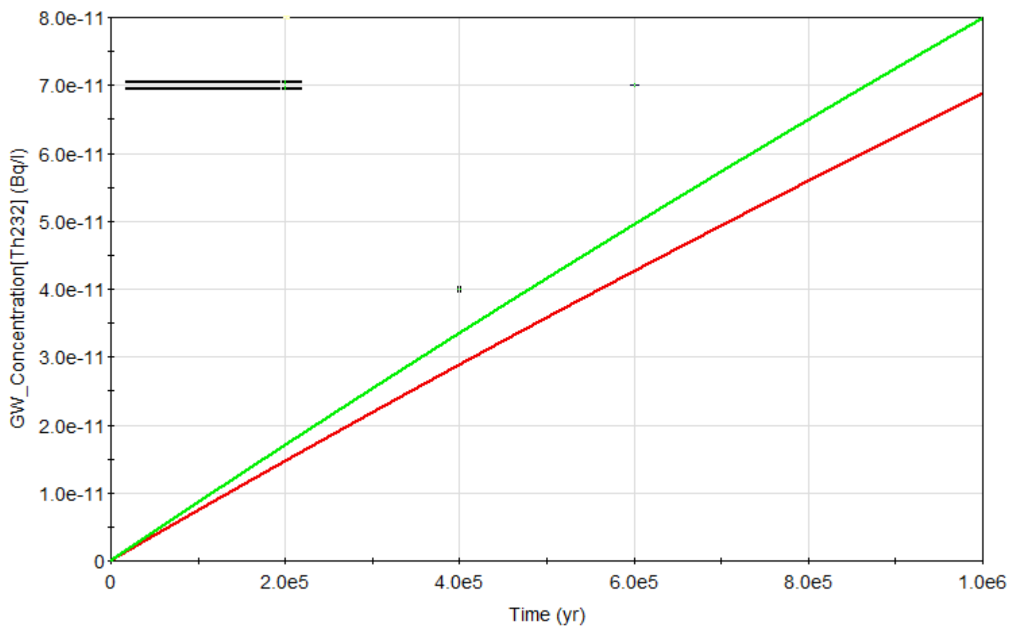


Figure 37. Goldsim estimation of Th-232 (Red) and daughter Ra-228 (Green) activity concentration in groundwater below the landfill



Appendix G. Impact of waste disposal using illustrative waste streams

G.1. Introduction

1163. In developing the safety case two illustrative inventories have been used, these are for wastes originally disposed to the Meashill trenches at Harwell (Augean, 2009a), used as an absolute inventory, and an illustrative NORM inventory based on the composition of a waste stream that has already been disposed at the ENRMF (consignment L12107400007, 2013), assuming 448,000 t of this waste are disposed.

1164. These calculations do not show the total impact of the whole facility, this will be dependent on the waste that is actually received for disposal. However, the calculations illustrate the dose that would arise from waste streams typical of those that might be disposed to the ENRMF. The Meashill Trenches inventory has not been disposed of at the ENRMF. Since the illustrative NORM waste stream contains significant activity concentrations of radium (>5 Bq g^{-1}), it is assumed to be placed at least 5 m below the restored surface. This is not important for the doses during the period of authorisation but is important for the doses after the period of authorisation. The Meashill Trenches Inventory is assumed to be disposed of at any depth. Neither inventory contains particles or large heterogeneously contaminated items.

1165. The radiological impact of the inventory that has already been disposed of at the ENRMF (up to June 2015) is given in Table 30 in terms of the fraction of the site capacity. The doses are also given here.

1166. In addition doses from two extrapolated inventories are presented in the following sections. The specific activity of waste in the current inventory as disposed at the ENRMF has been extrapolated to the maximum tonnage allowed at the site and the composition in national inventory has also been used to determine the dose from a disposal of 89.6 TBq.

Table 174 Inventories for calculation of contributions to the radiological impact

Radionuclide	Harwell Meashill trenches, 2010 (MBq)	NORM (ENRMF L12107400007, 2013) ($Bq\ g^{-1}$)
H-3	3.25	
Co-60	8050	
Cs-137	952	
Ra-226	99.6	110.23
Th-232	40	16.30
U-234	500	
U-235	24	
U-238	500	
Pu-238	37	
Pu-239	400	
Pu-240	400	
Pu-241	38.2	
Am-241	99.2	
Total (MBq)	$1.11\ 10^{4}$	$5.67\ 10^{7}$

G.2. Radiological impact of current disposed inventory

1167. The current permit, disposals to the ENRMF up to June 2015 and the proportion of each radionuclide in the waste disposed in the ENRMF and in the national low level waste inventory are presented in Table 175. Columns 4 and 5 of this Table provides an indication of the likely waste composition that will be disposed at the ENRMF and is referenced on a number of occasions in discussing the likelihood of doses occurring from specific radionuclides. We note that there are significant differences between the composition of waste disposed at the ENRMF and the national low level waste inventory.

Table 175 Activity disposed at the ENRMF and the composition of the national inventory of low level waste

Radionuclide	Permit CD8503 (MBq)	Activity disposed at ENRMF to June 2015 (MBq)	Composition of ENRMF disposals (percentage)	Composition of national LLW inventory (percentage)
H-3	$3.23\ 10^{6}$	$2.38\ 10^{4}$	26.1%	17.6%
C-14	$1.70\ 10^{5}$	$2.24\ 10^{3}$	2.5%	0.5%
Cl-36	$8.50\ 10^{4}$	$3.10\ 10^{1}$	0.0%	0.2%
Fe-55	$5.95\ 10^{5}$	$4.58\ 10^{2}$	0.5%	5.3%
Co-60	$7.32\ 10^{5}$	$1.49\ 10^{3}$	1.6%	6.7%
Ni-63	$1.87\ 10^{5}$	$1.46\ 10^{3}$	1.6%	2.5%
Sr-90	$1.75\ 10^{6}$	$2.73\ 10^{3}$	3.0%	15.1%
Nb-94	$8.50\ 10^{4}$	$2.76\ 10^{-1}$	0.0%	0.0%
Tc-99	$3.74\ 10^{5}$	$1.71\ 10^{1}$	0.0%	0.8%

Radionuclide	Permit CD8503 (MBq)	Activity disposed at ENRMF to June 2015 (MBq)	Composition of ENRMF disposals (percentage)	Composition of national LLW inventory (percentage)
Ru-106	$3.91\ 10^{5}$	$1.80\ 10^{-2}$	0.0%	0.5%
Ag-108m	$8.50\ 10^{4}$	$3.75\ 10^{-1}$	0.0%	0.0%
Sb-125	$8.50\ 10^{4}$	1.15	0.0%	0.1%
Sn-126	$8.50\ 10^{4}$	0	0.0%	0.0%
I-129	$8.50\ 10^{4}$	1.80	0.0%	0.0%
Ba-133	$8.50\ 10^{4}$	$2.11\ 10^{1}$	0.0%	0.0%
Cs-134	$8.50\ 10^{4}$	2.23	0.0%	0.3%
Cs-137	$5.10\ 10^{6}$	$1.59\ 10^{4}$	17.4%	30.8%
Pm-147	$1.19\ 10^{5}$	5.08	0.0%	0.3%
Eu-152	$8.50\ 10^{4}$	$8.77\ 10^{2}$	1.0%	0.0%
Eu-154	$8.50\ 10^{4}$	$5.92\ 10^{1}$	0.1%	0.1%
Eu-155	$8.50\ 10^{4}$	9.04	0.0%	0.0%
Pb-210	$8.50\ 10^{4}$	$1.47\ 10^{4}$	16.1%	0.0%
Ra-226	$3.06\ 10^{5}$	$1.86\ 10^{4}$	20.4%	0.2%
Ra-228	Included in 'other radionuclide'	0	0.0%	0.0%
Ac-227	$8.50\ 10^{4}$	4.43	0.0%	0.0%
Th-229	$8.50\ 10^{4}$	0	0.0%	0.0%
Th-230	$8.50\ 10^{4}$	$1.42\ 10^{2}$	0.2%	0.0%
Th-232	$8.50\ 10^{4}$	$3.98\ 10^{3}$	4.4%	0.0%
Pa-231	$8.50\ 10^{4}$	4.05	0.0%	0.0%
U-232	$8.50\ 10^{4}$	0	0.0%	0.0%
U-233	$8.50\ 10^{4}$	$2.70\ 10^{-2}$	0.0%	0.0%
U-234	$8.50\ 10^{4}$	$1.60\ 10^{2}$	0.2%	0.4%
U-235	$8.50\ 10^{4}$	6.58	0.0%	0.0%
U-236	$8.50\ 10^{4}$	$4.64\ 10^{-1}$	0.0%	0.0%
U-238	$8.50\ 10^{4}$	$3.18\ 10^{2}$	0.3%	0.3%
Np-237	$8.50\ 10^{4}$	0	0.0%	0.0%
Pu-238	$1.02\ 10^{5}$	$6.92\ 10^{1}$	0.1%	1.1%
Pu-239	$1.70\ 10^{5}$	$3.59\ 10^{2}$	0.4%	0.6%
Pu-240	$8.50\ 10^{4}$	$5.17\ 10^{2}$	0.6%	0.3%
Pu-241	$1.19\ 10^{6}$	$2.53\ 10^{3}$	2.8%	5.1%
Pu-242	$8.50\ 10^{4}$	$4.35\ 10^{-1}$	0.0%	0.0%
Am-241	$1.19\ 10^{5}$	$5.81\ 10^{2}$	0.6%	11.1%
Cm-243	$8.50\ 10^{4}$	1.51	0.0%	0.0%
Cm-244	$8.50\ 10^{4}$	$5.56\ 10^{1}$	0.1%	0.0%
Others	$8.50\ 10^{4}$	$8.49\ 10^{1}$	0.1%	0.0%
Total	$1.70\ 10^{7}$	$9.13\ 10^{4}$	100%	100%

1168. The doses calculated for the scenarios that limit disposals at the ENRMF are presented in Table 176 for the current disposed inventory. The doses from the scenarios that are expected to occur (recreational use and a well at the current abstraction point) are 0.0005 μSv y^{-1} and <0.043 μSv y^{-1} (the dose from a well constructed at the site boundary), respectively. These are at least two orders of magnitude below the GRA dose criterion of 20 μSv y^{-1} for these events. The dominant nuclides are H-3, C-14 and 'others' for the recreational scenario, and 'others' for the well scenario. The doses from the scenarios that are not expected to occur (borehole driller and smallholder) are 0.5 μSv y^{-1} and 0.2 μSv y^{-1}, respectively, which can be compared to the corresponding dose criteria of 20,000 μSv y^{-1} and 3,000 μSv y^{-1}, respectively for these events. The main contributions to the borehole driller come from Ra-226 and Th-232, and the main contributions to the dose to the smallholder come from Th-232 and 'others' and compared to the limit of 3000 μSv y^{-1} for this event. The dose from a well constructed at the site boundary would be <0.043 μSv y^{-1} compared to a limit of 20 μSv y^{-1} for this event.

Table 176 Doses from activity disposed at the ENRMF

Radionuclide	Doses from current inventory (June 2015) (μSv y^{-1})			
	Recreational users (0 years)	Borehole drilling (60 years)	Smallholder (200 years)	Well at boundary
H-3	$1.15\ 10^{-4}$	$1.40\ 10^{-8}$	$6.69\ 10^{-7}$	$<2.4\ 10^{-6}$
C-14	$3.74\ 10^{-4}$	$1.32\ 10^{-6}$	$8.32\ 10^{-3}$	$7.83\ 10^{-6}$
Cl-36	$8.26\ 10^{-30}$	$1.33\ 10^{-7}$	$1.72\ 10^{-3}$	$4.20\ 10^{-4}$
Fe-55	0	$2.36\ 10^{-14}$	$3.55\ 10^{-28}$	$<4.6\ 10^{-8}$
Co-60	$7.81\ 10^{-15}$	$1.17\ 10^{-5}$	$2.01\ 10^{-13}$	$<1.5\ 10^{-7}$
Ni-63	0	$1.16\ 10^{-7}$	$1.58\ 10^{-5}$	$<1.5\ 10^{-7}$
Sr-90	$1.09\ 10^{-24}$	$3.48\ 10^{-5}$	$6.60\ 10^{-3}$	$4.16\ 10^{-7}$
Nb-94	$7.70\ 10^{-21}$	$3.45\ 10^{-6}$	$5.76\ 10^{-6}$	$6.16\ 10^{-10}$
Tc-99	$8.62\ 10^{-52}$	$1.82\ 10^{-8}$	$5.65\ 10^{-4}$	$2.16\ 10^{-6}$
Ru-106	$6.81\ 10^{-24}$	$6.66\ 10^{-26}$	$1.20\ 10^{-66}$	$<1.8\ 10^{-12}$
Ag-108m	$6.16\ 10^{-22}$	$4.24\ 10^{-6}$	$5.62\ 10^{-6}$	$3.39\ 10^{-10}$
Sb-125	$1.38\ 10^{-22}$	$1.03\ 10^{-12}$	$9.26\ 10^{-28}$	$<1.1\ 10^{-10}$
Sn-126	0	0	0	0
I-129	$5.39\ 10^{-155}$	$1.36\ 10^{-7}$	$2.00\ 10^{-4}$	$8.62\ 10^{-4}$
Ba-133	$2.88\ 10^{-24}$	$1.04\ 10^{-6}$	$1.74\ 10^{-10}$	$<2.1\ 10^{-9}$
Cs-134	$2.21\ 10^{-20}$	$4.90\ 10^{-14}$	$4.29\ 10^{-34}$	$<2.2\ 10^{-10}$
Cs-137	$2.95\ 10^{-17}$	$1.77\ 10^{-2}$	$1.95\ 10^{-3}$	$<1.6\ 10^{-6}$
Pm-147	$2.51\ 10^{-51}$	$3.21\ 10^{-16}$	$2.95\ 10^{-31}$	$<5.1\ 10^{-10}$
Eu-152	$1.39\ 10^{-16}$	$3.67\ 10^{-4}$	$4.74\ 10^{-7}$	$<8.8\ 10^{-8}$
Eu-154	$1.35\ 10^{-17}$	$4.65\ 10^{-6}$	$9.71\ 10^{-11}$	$<5.9\ 10^{-9}$
Eu-155	$1.87\ 10^{-43}$	$3.43\ 10^{-10}$	$8.15\ 10^{-19}$	$<9.0\ 10^{-10}$
Pb-210	$1.01\ 10^{-21}$	$2.94\ 10^{-3}$	$2.90\ 10^{-3}$	$6.63\ 10^{-5}$
Ra-226	$2.77\ 10^{-12}$	$3.59\ 10^{-1}$	$2.78\ 10^{-7}$	$9.60\ 10^{-5}$
Ra-228	0	0	0	0
Ac-227	$6.54\ 10^{-24}$	$1.31\ 10^{-5}$	$1.05\ 10^{-7}$	$1.18\ 10^{-8}$
Th-229	0	0	0	0
Th-230	$1.78\ 10^{-39}$	$5.03\ 10^{-4}$	$6.15\ 10^{-3}$	$9.86\ 10^{-6}$
Th-232	$3.33\ 10^{-15}$	$1.06\ 10^{-1}$	$1.67\ 10^{-1}$	$4.89\ 10^{-4}$
Pa-231	$2.45\ 10^{-27}$	$8.81\ 10^{-5}$	$6.54\ 10^{-4}$	$3.66\ 10^{-7}$

Radionuclide	Doses from current inventory (June 2015) ($\mu Sv\ y^{-1}$)			
	Recreational users (0 years)	Borehole drilling (60 years)	Smallholder (200 years)	Well at boundary
U-232	0	0	0	0
U-233	$5.89\ 10^{-36}$	$1.01\ 10^{-8}$	$1.50\ 10^{-8}$	$1.72\ 10^{-8}$
U-234	$2.92\ 10^{-47}$	$4.98\ 10^{-5}$	$6.21\ 10^{-5}$	$5.00\ 10^{-4}$
U-235	$3.84\ 10^{-31}$	$8.34\ 10^{-6}$	$1.73\ 10^{-5}$	$2.68\ 10^{-5}$
U-236	$2.31\ 10^{-45}$	$1.33\ 10^{-7}$	$1.70\ 10^{-7}$	$6.45\ 10^{-8}$
U-238	$3.86\ 10^{-22}$	$1.40\ 10^{-4}$	$2.10\ 10^{-4}$	$2.51\ 10^{-4}$
Np-237	0	0	0	0
Pu-238	$9.12\ 10^{-50}$	$1.48\ 10^{-4}$	$2.20\ 10^{-5}$	$5.73\ 10^{-8}$
Pu-239	$3.07\ 10^{-31}$	$1.35\ 10^{-3}$	$6.01\ 10^{-4}$	$2.38\ 10^{-6}$
Pu-240	$6.10\ 10^{-57}$	$1.93\ 10^{-3}$	$8.51\ 10^{-4}$	$7.83\ 10^{-7}$
Pu-241	$7.20\ 10^{-41}$	$2.38\ 10^{-4}$	$1.14\ 10^{-4}$	$4.86\ 10^{-7}$
Pu-242	$6.06\ 10^{-73}$	$1.50\ 10^{-6}$	$6.90\ 10^{-7}$	$1.77\ 10^{-8}$
Am-241	$7.99\ 10^{-65}$	$1.61\ 10^{-3}$	$7.57\ 10^{-4}$	$5.17\ 10^{-6}$
Cm-243	$1.35\ 10^{-31}$	$1.06\ 10^{-6}$	$3.21\ 10^{-8}$	$1.68\ 10^{-10}$
Cm-244	0	$1.05\ 10^{-5}$	$2.75\ 10^{-7}$	$<5.6\ 10^{-9}$
Others	$1.41\ 10^{-5}$	$2.25\ 10^{-3}$	$1.37\ 10^{-2}$	$4.07\ 10^{-2}$
Total	$5.04\ 10^{-4}$	$4.94\ 10^{-1}$	$2.12\ 10^{-1}$	$<4.35\ 10^{-2}$

1169. The sum of fractions for the current inventory was presented in Table 30 and shows a value of 0.0024 for which the major contribution was from "Any Other Radionuclide" which adopts the lowest radiological capacity calculated for any radionuclide (I-129 from the well scenario). Removing this value (equivalent to giving 'others' a much higher radiological capacity than the one assigned to I-129) produces a sum of fractions of 0.0003. Table 177 shows the sum of fractions for the June 2015 inventory calculated for each of the scenarios contributing to the radiological capacity. Two versions of the sum of fractions are presented, with and without the "Any Other Radionuclide" category.

Table 177 Sum of fractions calculated using current inventory for the four capacity limiting scenarios

Radionuclide	Based on current inventory (June 2015)			
	Recreational users (0 years)	Borehole drilling (60 years)	Smallholder (200 years)	Well at boundary
Sum of fractions	$2.52\ 10^{-5}$	$1.65\ 10^{-4}$	$7.06\ 10^{-5}$	$2.17\ 10^{-3}$
Sum of fractions (Excluding Others)	$2.45\ 10^{-5}$	$1.64\ 10^{-4}$	$6.61\ 10^{-5}$	$1.37\ 10^{-4}$

1170. This indicates that, for the June 2015 disposed inventory, the capacity utilisation for the well at the boundary scenario is sensitive to the value of radiological capacity assigned to 'others'. The capacity utilisation for the other three scenarios is insensitive to the radiological capacity assigned to 'others'. A more sophisticated approach to assigning the capacity to 'others' (by taking into account the half-life, for example) or deriving values for additional radionuclides would lead to a more accurate estimate of the true fraction of the capacity that has been utilised.

1171. Doses have also been calculated assuming that a total of 448,000 t of waste with the same composition as the current inventory that has been disposed at the ENRMF is disposed of at the site by the time of closure. This applies the specific activity of current disposals to the maximum tonnage allowed at the site.

Table 178 Dose from an extrpolated final inventory based on current disposals for the four capacity limiting scenarios

Inventory (TBq)	Dose ($\mu Sv\ y^{-1}$)			
	Recreational users (0 years)	Borehole drilling (60 years)	Smallholder (200 years)	Well at boundary
3.79	$2.09\ 10^{-2}$	$2.05\ 10^{1}$	8.81	1.81

1172. The doses from the scenarios that are not expected to occur (borehole driller and smallholder) are 20 $\mu Sv\ y^{-1}$ and 8.8 $\mu Sv\ y^{-1}$, respectively, which can be compared to the corresponding dose criteria of 20,000 $\mu Sv\ y^{-1}$ and 3,000 $\mu Sv\ y^{-1}$, respectively for these events. The dose from a well constructed at the site boundary would be 1.8 $\mu Sv\ y^{-1}$ compared to a limit of 20 $\mu Sv\ y^{-1}$ for this event (with 1.7 $\mu Sv\ y^{-1}$ from "Other radionuclides").

G.3. Illustrative radiological impact during the period of authorisation

1173. In Table 179 and Table 180 the results of assessment calculations for the period of authorisation are applied to two illustrative inventories to indicate the potential radiological impact of waste disposal. Both likely and unlikely events are considered. The doses to members of the public and workers from disposal of the Meashill Trenches inventory at the ENRMF are shown in Table 179 and the corresponding doses from wastes with the same radionuclide mix (fingerprint) as an ENRMF NORM waste stream that has been disposed of at the site are shown in Table 180. Doses to the Fisherman are not shown because these are orders of magnitude lower than the Farming family.

1174. The calculated doses to workers and members of the public using the Meashill Trenches inventory are all significantly below 20 $\mu Sv\ y^{-1}$. The peak dose when summed across all radionuclides is $7.8\ 10^{-3}$ $\mu Sv\ y^{-1}$ to a leachate treatment facility worker; this is an insignificant dose. The groundwater pathway also results in an insignificant level of dose, estimated at $<1\ 10^{-6}$ $\mu Sv\ y^{-1}$. The 'less than' values for the well at the boundary (e.g. for Co-60) arise due to the cut-off used in the GoldSim model. GoldSim output uses an arbitrary lower limit of $1\ 10^{-10}$ $\mu Sv\ y^{-1}\ MBq^{-1}$ and this is applied to short lived radionuclides (half-life of less than about 5 years) when radioactive decay reduces the activity to very low levels or when transport of radionuclides in groundwater to the well, and hence to the exposed group, is low. The less than dose rate values have been omitted from the total dose shown. The Meashill Trenches inventory has not been disposed of at the ENRMF.

Table 179 Doses arising during the period of authorisation based on disposal of the Meashill Trenches waste inventory

Radionuclide	Inventory (MBq)	Dose ($\mu Sv\ y^{-1}$)				
		Well at boundary – PoA	Off-site gas (Operations)	Leachate spillage - Farming family	Leachate treatment - Facility worker	Leachate treatment - Farming family
H-3	3.25	<3.3 10^{-10}	1.65 10^{-8}	8.50 10^{-9}	3.95 10^{-11}	1.99 10^{-12}
Co-60	8.05 10^{3}	<8.0 10^{-7}		6.27 10^{-5}	7.73 10^{-3}	4.31 10^{-5}
Cs-137	9.52 10^{2}	<9.5 10^{-8}		3.96 10^{-5}	1.88 10^{-5}	6.35 10^{-8}
Ra-226	9.96 10^{1}	<1.0 10^{-8}	2.08 10^{-5}	2.18 10^{-5}	6.18 10^{-6}	8.55 10^{-8}
Th-232	4.00 10^{1}	<4.0 10^{-9}		5.76 10^{-7}	8.66 10^{-9}	9.96 10^{-9}
U-234	5.00 10^{2}	<5.0 10^{-8}		2.38 10^{-5}	2.23 10^{-7}	1.07 10^{-9}
U-235	2.40 10^{1}	<2.4 10^{-9}		1.10 10^{-6}	3.48 10^{-7}	3.30 10^{-10}
U-238	5.00 10^{2}	<5.0 10^{-8}		2.35 10^{-5}	1.67 10^{-6}	2.18 10^{-9}
Pu-238	3.70 10^{1}	<3.7 10^{-9}		4.83 10^{-7}	4.01 10^{-8}	7.49 10^{-10}
Pu-239	4.00 10^{2}	<4.0 10^{-8}		5.72 10^{-6}	4.75 10^{-7}	9.36 10^{-9}
Pu-240	4.00 10^{2}	<4.0 10^{-8}		5.72 10^{-6}	4.75 10^{-7}	9.15 10^{-9}
Pu-241	3.82 10^{1}	<3.8 10^{-9}		1.00 10^{-8}	1.42 10^{-9}	2.79 10^{-11}
Am-241	9.92 10^{1}	<9.9 10^{-9}		4.03 10^{-7}	5.50 10^{-8}	1.83 10^{-9}
Total	1.11 10^{4}	<1.1 10^{-6}	2.08 10^{-5}	1.85 10^{-4}	7.76 10^{-3}	4.33 10^{-5}

1175. The illustrative NORM waste inventory is based on a NORM waste stream containing 110 Bq g^{-1} Ra-226 and 16.3 Bq g^{-1} Th-232 and comprised of NORM contaminated pipe scale encapsulated in cement. These are the highest NORM activity concentrations that had been deposited in the landfill to the end of December 2014. Ra-226 and Th-232 are modelled explicitly in the calculations and for all times after disposal their daughter radionuclides are assumed to be in secular equilibrium with the parent; all daughters are therefore included in the dose conversion factors applied in the assessment models (see Table 1). The illustrative NORM waste inventory is assumed to comprise 448,000 t at the activity concentrations specified above. It is assumed to be placed at least 5m below the restored surface, although this emplacement strategy does not affect the doses during the period of authorisation.

Table 180 Doses arising during the period of authorisation based on an illustrative NORM waste inventory

Radionuclide	Inventory (MBq)	Dose ($\mu Sv\ y^{-1}$)				
		Well at boundary – PoA	Off-site gas (Operations)	Leachate spillage - Farming family	Leachate treatment - Facility worker	Leachate treatment - Farming family
Ra-226	4.94 10^{7}	<4.9 10^{-3}	1.03 10^{1}	1.08 10^{1}	3.06	4.24 10^{-2}
Th-232	7.30 10^{6}	<7.3 10^{-4}	0	1.05 10^{-1}	1.58 10^{-3}	1.82 10^{-3}
Total	5.67 10^{7}	<5.7 10^{-3}	1.03 10^{1}	1.09 10^{1}	3.06	4.42 10^{-2}

1176. The peak dose when summed across all radionuclides is 1.09 10^{1} $\mu Sv\ y^{-1}$ and occurs to the farming family after a leachate spillage during transport to an off-site leachate treatment facility; an event that is considered unlikely to occur. The highest doses from leachate treatment are to workers at the hazardous waste treatment facility, at 3 µSv y^{-1}. As stated above, this level of exposure is not expected to occur since the leachate is usually used on site. Leachate at the ENRMF is also routinely monitored to assess the need for a discharge permit. The estimated dose from groundwater abstraction during the period of authorisation is insignificant. The doses from the NORM waste inventory that has been disposed of to the ENRMF are a small fraction of these values since the quantity disposed of was much less than the 448,000 t assumed above.

1177. The doses from the current inventory and two extrapolated inventories are shown in Table 181. The specific activity of waste in the current inventory as disposed at the ENRMF has been extrapolated to the maximum tonnage allowed at the site and the composition in national inventory has also been used to determine the dose from a disposal of 89.6 TBq.

Table 181 Dose during the period of authorisation based on three illustrative inventories

Illustrative inventory	Inventory (GBq)	Dose (µSv y^{-1})				
		Well at boundary – PoA	Off-site gas (Operations)	Leachate spillage - Farming family	Leachate treatment - Facility worker	Leachate treatment - Farming family
Current inventory	$9.13\ 10^{1}$	$4.35\ 10^{-2}$	$4.39\ 10^{-3}$	$2.12\ 10^{-2}$	$3.10\ 10^{-3}$	$3.76\ 10^{-5}$
Extrapolated from specific activity of current disposals	$3.79\ 10^{3}$	1.81	$1.83\ 10^{-1}$	$8.82\ 10^{-1}$	$1.29\ 10^{-1}$	$1.56\ 10^{-3}$
Extrapolated from national inventory	$8.96\ 10^{4}$	5.98	$1.98\ 10^{-1}$	3.22	6.40	$4.98\ 10^{-2}$

G.4. Illustrative radiological impact after the period of authorisation

1178. In Table 182 and Table 183 the results of the assessment calculations of the impact after the period of authorisation are applied to the two illustrative inventories to indicate the potential radiological impact of waste disposal. The Meashill Trenches inventory is used in Table 182 and an illustrative NORM inventory based on an ENRMF NORM waste stream is shown in Table 183.

1179. The calculated doses using the Meashill Trenches inventory are all significantly below 20 µSv y^{-1}. The peak dose when summed across all radionuclides is about $2\ 10^{-3}$ µSv y^{-1} and this is to the users of the well at the site boundary who are assumed to extract groundwater for drinking, crop irrigation and livestock. As mentioned previously, this sum over radionuclides will overestimate the dose as the peak doses for each radionuclide from groundwater abstraction will occur at different times.

Table 182 Doses arising after the period of authorisation based on disposal of the Meashill Trenches waste inventory

Radionuclide	Inventory (MBq)	Dose (μSv y^{-1})			
		Well at boundary	Well at 1500m	Bathtubbing	Recreational users (0 years)
H-3	3.25	<3.3 10^{-10}	<3.3 10^{-10}	<3.3 10^{-10}	1.58 10^{-8}
Co-60	8.05 10^{3}	<8.0 10^{-7}	<8.0 10^{-7}	<8.0 10^{-7}	4.21 10^{-14}
Cs-137	9.52 10^{2}	<9.5 10^{-8}	<9.5 10^{-8}	<9.5 10^{-8}	1.77 10^{-18}
Ra-226	9.96 10^{1}	5.13 10^{-7}	2.18 10^{-7}	2.73 10^{-7}	1.48 10^{-14}
Th-232	4.00 10^{1}	4.92 10^{-6}	2.10 10^{-6}	1.23 10^{-7}	3.35 10^{-17}
U-234	5.00 10^{2}	1.56 10^{-3}	6.65 10^{-4}	4.38 10^{-8}	9.11 10^{-47}
U-235	2.40 10^{1}	9.76 10^{-5}	4.16 10^{-5}	3.41 10^{-8}	1.40 10^{-30}
U-238	5.00 10^{2}	3.94 10^{-4}	1.68 10^{-4}	1.54 10^{-7}	6.06 10^{-22}
Pu-238	3.70 10^{1}	3.06 10^{-8}	1.06 10^{-8}	2.96 10^{-11}	4.88 10^{-50}
Pu-239	4.00 10^{2}	2.65 10^{-6}	1.11 10^{-6}	1.39 10^{-8}	3.42 10^{-31}
Pu-240	4.00 10^{2}	6.06 10^{-7}	2.59 10^{-7}	1.34 10^{-8}	4.72 10^{-57}
Pu-241	3.82 10^{1}	7.32 10^{-9}	<3.8 10^{-9}	<3.8 10^{-9}	1.09 10^{-42}
Am-241	9.92 10^{1}	8.84 10^{-7}	3.78 10^{-7}	6.83 10^{-10}	1.37 10^{-65}
Total	1.11 10^{4}	2.06 10^{-3}	8.79 10^{-4}	6.57 10^{-7}	1.58 10^{-8}

1180. The illustrative NORM waste inventory is based on a NORM waste stream containing 110 Bq g^{-1} Ra-226 and 16.3 Bq g^{-1} Th-232 and comprised of NORM contaminated pipe scale encapsulated in cement. These are the highest NORM activity concentrations that had been deposited in the landfill to the end of June 2015. Ra 226 and Th-232 are modelled explicitly in the calculations and their daughter radionuclides are assumed to be in secular equilibrium with the parent; all daughters are therefore included in the dose conversion factors applied in the assessment models (see Table 1). The illustrative NORM waste inventory is assumed to comprise 448,000 t at the activity concentrations specified above. It is assumed to be placed at least 5 m below the restored surface.

Table 183 Doses arising after the period of authorisation based on an illustrative NORM waste inventory

Radionuclide	Inventory (MBq)	Dose (μSv y^{-1})			
		Well at boundary	Well at 1500m	Bathtubbing	Recreational users (0 years)
Ra-226*	4.94 10^{7}	2.54 10^{-1}	1.08 10^{-1}	1.36 10^{-1}	7.35 10^{-9}
Th-232	7.30 10^{6}	8.97 10^{-1}	3.83 10^{-1}	2.25 10^{-2}	6.12 10^{-12}
Total	5.67 10^{7}	1.15	4.91 10^{-1}	1.58 10^{-1}	7.36 10^{-9}

* Dose to recreational users corresponding to a 5 m emplacement depth, all other doses are independent of emplacement depth (see Section 6.3.6).

1181. The peak dose when summed across all radionuclides is 1.15 μSv y^{-1} and this is to the users of the well at the site boundary who are assumed to extract groundwater for drinking, crop irrigation and livestock. The doses from the NORM waste inventory that has been disposed of to the ENRMF are a small fraction of these values since the quantity disposed of was much less than the 448,000 t assumed above and they were disposed of at least 5 m below the restored surface.

1182. The doses from the current inventory and two extrapolated inventories are shown in Table 184. The specific activity of waste in the current inventory as disposed at the ENRMF has been extrapolated to the maximum tonnage allowed at the site and the composition in national inventory has also been used to determine the dose from a disposal of 89.6 TBq.

Table 184 Doses arising after the period of authorisation based on three illustrative inventories

Illustrative inventory	Inventory (GBq)	Dose (µSv y^{-1})			
		Well at boundary	Well at 1500m	Bathtubbing	Recreational users (0 years)
Current inventory	$9.13\ 10^{1}$	$4.35\ 10^{-2}$	$1.86\ 10^{-2}$	$8.84\ 10^{-5}$	$5.04\ 10^{-4}$
Extrapolated from specific activity of current disposals	$3.79\ 10^{3}$	1.81	$7.72\ 10^{-1}$	$3.67\ 10^{-3}$	$2.09\ 10^{-2}$
Extrapolated from national inventory	$8.96\ 10^{4}$	5.98	2.55	$1.66\ 10^{-1}$	$1.47\ 10^{-1}$

* Dose to recreational users corresponding to a 5 m emplacement depth for Ra-226 wastes, all other doses are independent of emplacement depth (see Section 6.3.6).

G.5. Illustrative Radiological Impact for Intrusion Scenarios

1183. In Table 185 and Table 186 the results of assessment calculations are applied to illustrative inventories to indicate the potential radiological impact of waste disposal. The results for the Meashill Trenches inventory are shown in Table 185 and an illustrative NORM inventory based on an ENRMF waste stream are shown in Table 186. It is assumed in these calculations that the NORM inventory is buried at least 5 m below the restored ground level and the Meashill inventory is buried at any depth.

1184. The calculated doses using the Meashill Trenches inventory are all significantly below 3 mSv y^{-1}. The peak dose when summed across all radionuclides is about 8 10^{-6} mSv y^{-1} to the borehole driller.

Table 185 Intrusion doses based on the Meashill Trenches waste inventory

Radionuclide	Inventory (MBq)	Dose (mSv y^{-1})				
		Small holder 200 years	Resident 150 years	Borehole drilling 60 years	Excavator (House) 150 years	Laboratory analyst 60 years
H-3	3.25	$9.15\ 10^{-14}$	$1.15\ 10^{-12}$	$1.92\ 10^{-15}$	$1.21\ 10^{-17}$	$8.08\ 10^{-16}$
Co-60	$8.05\ 10^{3}$	$1.08\ 10^{-15}$	$6.65\ 10^{-13}$	$6.31\ 10^{-8}$	$4.56\ 10^{-13}$	$3.16\ 10^{-9}$
Cs-137	$9.52\ 10^{2}$	$1.17\ 10^{-7}$	$2.08\ 10^{-7}$	$1.06\ 10^{-6}$	$1.33\ 10^{-7}$	$5.34\ 10^{-8}$
Ra-226	$9.96\ 10^{1}$	$1.48\ 10^{-12}$	$1.62\ 10^{-12}$	$1.92\ 10^{-6}$	$1.51\ 10^{-6}$	$1.65\ 10^{-7}$
Th-232	$4.00\ 10^{1}$	$1.68\ 10^{-6}$	$1.30\ 10^{-6}$	$1.06\ 10^{-6}$	$1.06\ 10^{-6}$	$1.76\ 10^{-7}$
U-234	$5.00\ 10^{2}$	$1.94\ 10^{-7}$	$6.82\ 10^{-8}$	$1.56\ 10^{-7}$	$1.57\ 10^{-7}$	$9.25\ 10^{-8}$
U-235	$2.40\ 10^{1}$	$6.32\ 10^{-8}$	$3.98\ 10^{-8}$	$3.04\ 10^{-8}$	$3.06\ 10^{-8}$	$5.26\ 10^{-9}$
U-238	$5.00\ 10^{2}$	$3.29\ 10^{-7}$	$1.88\ 10^{-7}$	$2.20\ 10^{-7}$	$2.19\ 10^{-7}$	$8.31\ 10^{-8}$

Pu-238	$3.70\ 10^{1}$	$1.18\ 10^{-8}$	$9.37\ 10^{-9}$	$7.92\ 10^{-8}$	$3.88\ 10^{-8}$	$4.85\ 10^{-8}$
Pu-239	$4.00\ 10^{2}$	$6.69\ 10^{-7}$	$3.60\ 10^{-7}$	$1.50\ 10^{-6}$	$1.49\ 10^{-6}$	$9.18\ 10^{-7}$
Pu-240	$4.00\ 10^{2}$	$6.58\ 10^{-7}$	$3.55\ 10^{-7}$	$1.49\ 10^{-6}$	$1.47\ 10^{-6}$	$9.14\ 10^{-7}$
Pu-241	$3.82\ 10^{1}$	$1.72\ 10^{-9}$	$9.12\ 10^{-10}$	$3.58\ 10^{-9}$	$3.16\ 10^{-9}$	$2.16\ 10^{-9}$
Am-241	$9.92\ 10^{1}$	$1.29\ 10^{-7}$	$6.88\ 10^{-8}$	$2.75\ 10^{-7}$	$2.38\ 10^{-7}$	$1.66\ 10^{-7}$
Total	$1.11\ 10^{4}$	$3.85\ 10^{-6}$	$2.59\ 10^{-6}$	$7.86\ 10^{-6}$	$4.85\ 10^{-6}$	$2.63\ 10^{-6}$

1185. The illustrative NORM waste inventory is based on a NORM waste stream containing 110 Bq g^{-1} Ra-226 and 16.3 Bq g^{-1} Th-232 and comprised of NORM contaminated pipe scale encapsulated in cement. These are the highest NORM activity concentrations that have been deposited in the landfill to the end of June 2015. Ra-226 and Th-232 are modelled explicitly in the calculations and their daughter radionuclides are assumed to be in secular equilibrium with the parent; all daughters are therefore included in the dose conversion factors applied in the assessment models (see Table 1). The illustrative NORM waste inventory is assumed to comprise 448,000 t at the activity concentrations specified above. It is assumed to be buried at least 5 m below the restored surface of the site.

Table 186 Doses arising from intrusion based on a NORM waste inventory

Radionuclide	Inventory (MBq)	Dose (mSv y^{-1})				
		Smallholder 200 years	Resident 150 years	Excavator (Borehole) 60 years	Excavator (House) 150 years	Laboratory analyst 60 years
Ra-226	$4.94\ 10^{7}$	$7.36\ 10^{-7}$	$8.02\ 10^{-7}$	$9.52\ 10^{-1}$	0	$8.18\ 10^{-2}$
Th-232	$7.30\ 10^{6}$	0	0	$1.94\ 10^{-1}$	0	$3.21\ 10^{-2}$
Total	$5.67\ 10^{7}$	$7.36\ 10^{-7}$	$8.02\ 10^{-7}$	1.15	0	$1.14\ 10^{-1}$

1186. The calculated doses using the NORM waste inventory are below the dose guidance level for intrusion (prolonged exposure) of 3 mSv y^{-1}. The peak dose when summed across all radionuclides is 1.15 Sv y^{-1} to the borehole driller. The doses from the NORM waste inventory that has been disposed of to the ENRMF are a small fraction of these values since the quantity disposed of was much less than the 448,000 t assumed above.

1187. The doses from the current inventory and two extrapolated inventories are shown in Table 184. The specific activity of waste in the current inventory as disposed at the ENRMF has been extrapolated to the maximum tonnage allowed at the site and the composition in national inventory has also been used to determine the dose from a disposal of 89.6 TBq.

Table 187 Doses arising from intrusion based on three illustrative inventories

Illustrative inventory	Inventory (GBq)	Dose (mSv y^{-1})				
		Smallholder 200 years	Resident 150 years	Excavator (Borehole) 60 years	Excavator (House) 150 years	Laboratory analyst 60 years
Current inventory	$9.13\ 10^{1}$	$2.12\ 10^{-4}$	$1.55\ 10^{-4}$	$4.94\ 10^{-4}$	$4.61\ 10^{-4}$	$5.49\ 10^{-5}$
Extrapolated from specific activity of current disposals	$3.79\ 10^{3}$	$8.81\ 10^{-3}$	$6.45\ 10^{-3}$	$2.05\ 10^{-2}$	$1.92\ 10^{-2}$	$2.28\ 10^{-3}$
Extrapolated from national inventory	$8.96\ 10^{4}$	$8.81\ 10^{-2}$	$3.82\ 10^{-2}$	$6.87\ 10^{-2}$	$3.67\ 10^{-2}$	$2.21\ 10^{-2}$

COMMERCIAL

Environmental Safety Case:

Disposal of Low Activity Low Level Radioactive Waste at East Northants Resource Management Facility

Final: ENE-154/001

Appendices H to L

July 2015

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Appendix H. ENRMF, IRRs 1999, Radiation Risk Assessment for Low Level Waste Disposal (HPA)

EAST NORTHANTS RESOURCE MANAGEMENT FACILITY

IONISING RADIATIONS REGULATIONS 1999

RADIATION RISK ASSESSMENT FOR LLW WITH A SPECIFIC ACTIVITY UP TO 200Bq/g

1 SCOPE AND DEFINITIONS

1.1 INTRODUCTION

The East Northants Resource Management Facility (ENRMF) operated by Augean plc is intending to dispose of low level radioactive wastes (LLW) with a specific activity of up to 200Bq/g. An application under the Radioactive Substances Act 1993 has been prepared, and this includes an assessment of the potential radiation exposure of workers and members of the public. In addition to this, the Ionising Radiations Regulations 1999 (IRR99) require that a radiological risk assessment is undertaken for any work involving ionising radiation. Specifically, Regulation 7 requires the radiation employer (Augean plc) to carry out a prior risk assessment before commencing work with radioactive materials at the ENRMF site. This document is intended to meet the requirements of this Regulation in relation to the operational phase of the controlled burial operation.

1.2 RADIOACTIVE MATERIALS AND RADIATION HAZARDS

The type and quantities of radioactive materials that may be accepted at ENRMF are described in the RSA93 application and supporting documents. In brief, the application includes a range of potential radionuclides from nuclear and non-nuclear practices (including radionuclides of natural origin) with a maximum total activity concentration of 200 Bq/g. This assessment pessimistically assumes that the waste received contains radionuclides at the maximum activity concentrations, which is unlikely to be the case in practice.

The radionuclides considered emit a combination of alpha and beta particles and gamma rays. The handling of these materials can potentially give rise to a radiation hazard from:

- external gamma exposure from proximity to the waste (either during handling waste containers or occupancy of the disposal areas);
- internal radiation exposure from the inhalation of contaminated dust (air contamination) arising during the work;
- internal radiation from the transfer and inadvertent ingestion of material (surface contamination) during the work; and
- internal radiation from any contaminated wounds incurred during the work.

This risk assessment focuses on the exposure of workers and other persons visiting the ENRMF site. The potential radiation exposure of persons off-site (i.e. members of the public) from a range of exposure pathways has been considered in detail in the RSA93 application. This demonstrated that the maximum dose to a member of the public is expected to be below 0.02 mSv per year[1]. This is well below the relevant IRR99 dose limit of 1 mSv per

[1] A higher dose of up to 1 mSv was associated with accidental (public) intrusion into the landfill. This is a post-closure scenario and is beyond the scope of this risk assessment.

year, and consequently doses to persons off-site are not considered in detail in this risk assessment.

1.2 RISK ASSESSMENT REQUIREMENTS

The purpose of this risk assessment is to identify the measures needed to restrict the exposure of employees and other persons to ionising radiation from the controlled burial waste (LLW) operations at ENRMF.

IRR99 Regulation 7 also requires that potential radiation accidents are identified and quantified, and that steps are taken to prevent accidents, limit the consequences of any accidents that do occur, and to provide any necessary information, instruction, training and equipment to deal with such accidents.

Paragraph 44 of the Approved Code of Practice to IRR99 recommends that the following matters should be considered when carrying out this risk assessment. The parts of this document that correspond to these matters are listed in the table below.

Nature of the radiation source	1.2
Estimated radiation dose rates to which anyone can be exposed	2.1
Likelihood of contamination arising and being spread	2.2
Results of previous personal dosimetry or area monitoring	2.1.1
Advice from manufacturers or suppliers	4.5
Engineering control measures and design features	4.1
Any planned systems of work	4.1, 4.4.1
Estimated levels of airborne and surface contamination	2.2.1, 3
Effectiveness and suitability of personal protective equipment	4.5
Extent of unrestricted access to working areas where dose rates or contamination levels are likely to be significant	4.2
Possible accident situations, their likelihood and potential severity	3
The failure of control measures or systems of work	3
Steps to prevent identified accident situations or limit their consequences	3, 4.1.3

Paragraph 45 of the ACoP states that the risk assessment should enable the radiation employer to determine the following outcomes. Again, the relevant parts of this risk assessment are indicated in the table below.

What action is needed to ensure that radiation exposures are as low as reasonably practicable (ALARP)	4.1
What engineering controls, design features, safety and warning devices, and systems of work are needed	4.1
Whether it is appropriate to provide personal protective equipment	4.1, 4.5
Whether dose constraints for planning purposes are needed	4.1.4
The need to alter the working conditions of any female employee who declares she is pregnant or breastfeeding	4.1.6
A dose investigation level to check that exposures are ALARP	4.1.5
What maintenance and testing schedules are required	4.5
What contingency plans are necessary	4.1
The training of classified and non-classified employees	4.6
The need to designate specific areas as controlled or supervised and the need for local rules	4.2
The actions needed to ensure restriction of access for controlled or supervised areas	4.2, 4.4.1
The need to designate certain employees as classified persons	4.3
The need for individual dose assessment	4.3
The responsibilities of managers	4.4.2
An appropriate programme of monitoring or auditing of arrangements.	4.7

2 RADIATION RISKS FROM NORMAL OPERATIONS

2.1 RADIATION DOSE RATES AND EXTERNAL RADIATION RISKS

2.1.1 Augean employees engaged in the LLW operation

The radiation dose rates from a range of radionuclides have been calculated as part of the supporting documents to the RSA93 application. Extracts from these documents, relevant to the estimation of external dose, are reproduced in Appendix 1.

For all external dose scenarios, cobalt-60 is the limiting radionuclide (i.e. it gives rise to the highest dose rates). For the purpose of this risk assessment, the following representative dose rates, working patterns and estimated doses are used.

Work activity	Dose rate (μSv/h)	Occupancy (hours/year)	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 mSv**

Summary

The above estimates are likely to be conservative, and it is unlikely that the same person(s) will be exposed during all the work activities listed above. Nevertheless, it is reasonable to assume, for planning purposes, that annual external doses of the order of 1 mSv per year might be associated with the LLW operation.

2.1.2 External radiation risks to other persons

Such persons might include other employees (i.e. not involved in the LLW operation), visitors to site, etc. Such persons would be unlikely to be exposed to dose rates above 1 μSv/h, and exposure times would be expected to be short. Consequently, it is expected that external doses to such persons should be negligible.

External doses to members of the public (during and after the LLW operation) were estimated in the RSA93 application, and are a small fraction of the 20 μSv/y dose constraint.

2.2 CONTAMINATION LEVELS AND INTERNAL RADIATION RISKS

The waste will be in closed containers (either steel drums or bulk bags) throughout the LLW operation. Furthermore, waste consigners will be required to demonstrate that the external surfaces of these containers are effectively free of loose contamination. Consequently, contamination levels, and hence internal radiation doses, during normal operations are expected to be negligible.

There is the potential for contamination and internal exposures arising from accidents, in particular a damaged container. This is considered in the Section 3.

3 RADIATION RISKS FROM ACCIDENTS

The following reasonably foreseeable incidents/accidents have been identified:

3.1 The delivery of waste containing unexpectedly high levels of radioactivity

The likelihood of receiving waste that is more radioactive than expected is limited by the strict pre-acceptance criteria and associated procedures that are to be put in place. In addition, it is expected that incoming consignments will be monitored, and a dose rate acceptance test applied. Thus any radiation exposures from this scenario, should be limited to a brief external exposure to increased dose rates at the receiving stage. Even if the dose rate is 10x the acceptance criteria, the resulting doses to workers from the monitoring and subsequent quarantine of the consignment would be expected to be negligible.

3.2 Dropping or otherwise damaging a container of waste and spilling the contents

The "dropped bag" scenario is specifically considered in the RSA93 application using a pessimistic dispersion model to estimate the radiation doses (from dust inhalation) to workers and persons off-site. This assessment is principally concerned with the exposure of workers, in particular those that may be involved in cleaning up any spills. Consequently, for this risk assessment the following general "spillage" scenario is assumed:

- either type of waste container (drum or bag) could be damaged;
- contaminated dust is released producing a localised dust loading of 10 mg/m^3, which is considered a pessimistic assumption for an accident outdoors;
- workers remain in the above dust loading for a total of 4 hours (to allow for any clean-up).
- the worker breathing rate is 1.2 m^3/h and no respiratory protective equipment (RPE) is worn; and
- dust is inadvertently ingested (e.g. during the clean-up) at a rate of 3.45×10^{-5} kg/h (the same rate as assumed in the RSA application for excavation scenarios)

The above assumptions produce an inhaled dust mass of 48 mg, and an ingested dust mass of 138 mg. ICRP dose coefficients for inhalation and ingestion (the same as those used in the RSA93 application) are given in the Appendix to this risk assessment. Combining these with the mass of dust inhaled and ingested, and an activity concentration of 200 Bq/g (i.e. a worst case assumption) gives the following (rounded) internal doses:

Radionuclide	Estimated internal dose from a single spillage (mSv)		
	Inhalation	**Ingestion**	**Total**
Ac-227	5		5
Th-229	2		2
Th-230,232 Pa-231 Pu-238, 239, 240, 242 Am-241	1	All < 0.05 mSv	1
Ra-228, Th-228 U-232, Np-237 Cm-243, 244	0.1 to 1		0.1 to 1
All other radionuclides	<0.1		<0.1

Thus, dust inhalation is the dominant exposure pathway. The highest estimated doses are for actinium-227 and thorium-229. However, it is considered highly unlikely that waste would contain these radionuclides at 200 Bq/g.

There would also be an external dose associated with the clean up. Assuming a 4 hour exposure to an average dose rate of 10 µSv/h gives an external dose of 0.04 mSv. Taking all these factors into account, **it is concluded that the radiation exposure (internal plus external) from a spillage of waste containing up to 200 Bq/g is unlikely to exceed 1 mSv**. This includes any exposures from the subsequent clean-up of the spill.

3.3 Internal exposure from contaminated wounds

Under normal circumstances this is not a reasonably foreseeable exposure scenario. However, if contamination does arise, for example because of the spill scenario in 3.2 above, then this additional accident exposure pathway becomes a possibility. It is considered that doses from this pathway would be likely to be the same order of magnitude as from inadvertent ingestion, i.e., less than 0.1 mSv.

The UKAEA Safety Assessment Handbook (UKAEA/SAH/D9, Issue 1, March 2006) gives dose factors for contaminated wounds. Assuming that 0.1 g of material (at 200 Bq/g) becomes incorporated into a wound, the highest estimated dose is approximately 3 mSv, from actinium-227. As mentioned above, this radionuclide is most unlikely to predominate, and it is concluded that internal doses from a contaminated wound would be very unlikely to exceed 1 mSv in practice.

4. RECOMMENDED ACTIONS - REQUIREMENTS OF IRR99

4.1 RESTRICTION OF EXPOSURE (IRR99 REGULATION 8)

4.1.1 Summary of estimated doses

Regulation 8 requires that every radiation employer shall take all necessary steps to ensure that the radiation exposure of employees and other persons is as low as reasonably practicable (ALARP). The preceding dose assessment produced the following estimated effective doses:

Augean LLW workers	
Normal operations:	1 mSv/y from external exposure
Accidents:	Negligible (<0.1 mSv) increase in external dose due to receipt of consignments containing higher than expected activity concentrations. Up to 1 mSv (principally from inhalation) from dealing with spills of loose waste material (without special precautions). Up to 1 mSv from contaminated wounds incurred when dealing with spills of loose waste material (without special precautions).
Other persons	
Normal operations and accidents	Doses are expected to be negligible, but even in the worst case should be much less than those estimated above for workers.

The estimated doses are a small fraction of the annual dose limits of 20 mSv (for workers) and 1 mSv (for other persons) specified in IRR99. Notwithstanding this, there is still a requirement to keep exposures ALARP, and the recommended steps needed to achieve this are given below.

4.1.2 Protection during normal operations

Augean LLW workers

- All LLW waste should be received, handled and disposed of in closed containers. Consignments of waste should be checked and verified as they arrive on site.
- External radiation exposures can be restricted by setting a limit on the dose rate from each waste container. Based on this risk assessment, a limit of 10 μSv/h at 1 metre from the container is recommended. A quarantine area should be provided for holding containers that exceed this level.
- External radiation exposures can be restricted through time and distance. Procedures should be in place to ensure that LLW consignments are disposed of as quickly as possible. Workers should be instructed to avoid loitering near waste containers.
- Deposited waste should be covered with a compacted 300mm inactive layer, as soon as practicable. The dose rate above the covered waste should be monitored and should not exceed 2 μSv/h. If necessary, additional covering material should be applied until this is achieved.
- Although contamination is not expected during normal operations, it is good practice for workers to wear suitable overalls and gloves during the LLW work, which will provide

protection in the event of a spillage of waste. This is existing practice at the ENRMF site for al operatives.

- Radiation monitoring (individual and environmental) is required – see 4.3 and 4.7 below.
- Local rules and training should be provided - see 4.4 and 4.6 below.

Other persons

- Other persons should be excluded from the immediate area during the LLW operation.
- The dust suppression measures for spills, as described below, should also ensure that the spread of airborne dust is minimised. No other specific protection measures are required.

4.1.3 Accidents – prevention and mitigation

- Dose rate checks on incoming consignments of waste should be undertaken, as recommended above.
- Contingency plans should be prepared for dealing with spillages of waste. These should include the following precautions:
    - Simple dust suppression measures (e.g. damping down, and avoiding dust resuspension during clean-up) should be applied, where practicable.
    - As an additional precaution, workers should wear respiratory protective equipment when cleaning up spills – see 4.5 below.
    - Spilled material must be placed into suitable containers for disposal, and the affected area should be monitored to ensure that all contaminated material has been removed.

  The above precautions should ensure that the radiation doses from accidents are negligible (<0.1 mSv).

4.1.4 Dose constraints

Regulation 8(3) requires that dose constraints are considered at the planning stage of radiation protection. This has also been considered as part of the RSA93 application. Based on the original application, and on the implementation of the findings of this risk assessment, the following dose constraints are recommended:

LLW workers: 1 mSv/y

Other persons: 0.02 mSv/y

4.1.5 Dose investigation level

Regulation 8(7) requires that employers should set an investigation level for the purposes of determining whether exposures are being kept ALARP. Based upon the findings of this risk assessment it is recommended that Augean set a dose investigation level of 1 mSv for its employees.

The monitoring required to compare exposures against the investigation level is discussed in 4.3 below.

If the dose investigation level is exceeded, Augean in consultation with their RPA should undertake an investigation to determine whether the steps being taken to restrict exposures are sufficient.

4.1.6 Pregnant and breast-feeding employees

Regulation 8(5) contains additional dose restriction provisions for such employees. For pregnant women, it is recommended the dose to the foetus should be kept below 1 mSv. Whilst exposures of over 1 mSv are unlikely to occur, as a precaution it is recommended that pregnant employees are not allowed in the LLW work areas.

For breastfeeding women, it is recommended that they avoid situations where significant bodily contamination might occur. As a general precaution, it is recommended that such women are not allowed in the LLW work areas.

The risks associated with radiological hazards should be incorporated in the company risk assessment for pregnant and breastfeeding employees.

4.2 DESIGNATED AREAS

4.2.1 Controlled areas

Regulation 16 requires the designation of a controlled area where either:

a) radiation doses are likely to exceed three-tenths of the annual dose limits for workers (e.g. 6mSv/y effective dose); or

b) special working procedures are required to restrict radiation exposures.

Worker doses are not expected to exceed 6 mSv/y. However, it is considered that special working procedures (as defined in Regulation 16(1)) are appropriate in respect of certain operations. Consequently the following recommendations are made:

- Incoming waste consignments should be rapidly processed, and should not remain in any one area for an extended period of time. On this basis, a controlled area (for example, around arriving vehicles) is not recommended.
- A quarantine area should be provided for waste consignments that do not meet the dose rate limits described previously, and this should be designated as a controlled area whenever such consignments are quarantined. It should be ensured that the dose rate outside this area is below 2 µSv/h.
- During the deposition of waste containers, elevated dose rates are present, and there is the potential for accidents (dropped containers, etc.). It is recommended that this area is designated as a controlled area during the disposal operation, and should remain designated until a satisfactory covering layer has been applied (see 4.1.2).

Controlled areas should, where practicable be physically demarcated and warning signs posted at the points of entry. For the above areas, the following is recommended:

- **Quarantine area**: the perimeter should be fully demarcated, ideally with fencing, but if not, with rope barriers or similar. A controlled area warning sign should be posted at all points of potential access.
- **Disposal area**: during the operational period the area will be occupied or under surveillance and it is considered sufficient to temporarily post controlled area warning signs at the access points to the area.

Access to the controlled areas should be restricted to authorised personnel. Local rules and Radiation Protection Supervisors (Regulation 17) should be provided for controlled areas – see 4.4 below.

4.2.1 Supervised areas

The Regulations also require that a supervised area should be designated where it is considered necessary to keep the radiological conditions under periodic review. Although some confirmatory monitoring is recommended outside the controlled areas (see 4.7 below), the designation of a supervised area is not considered necessary provided that the aforementioned dose rate limits are met.

4.3 CLASSIFIED PERSONS AND INDIVIDUAL MONITORING

4.3.1 Designation of classified persons

Regulation 20 requires workers to be designated as classified persons if they are likely to receive an effective dose in excess of 6 mSv per year. This risk assessment indicates that doses are expected to be well below this value and, therefore, it is **not** recommended that Augean employees are designated as classified persons.

4.3.2 Monitoring of individual dose

As a means of confirming the restriction of exposures, and for checking against the Dose Investigation Level, it is recommended that a programme of individual dose monitoring is implemented for all Augean employees engaged in the LLW operation. For monitoring external exposure, it is recommended that passive whole body dosemeters (e.g. TLDs) are worn, and Augean should make the necessary arrangements with an appropriate dosimetry service.

Internal exposures during normal operations are expected to be negligible, and the precautions listed in Section 4.1.3 should ensure that this is also the case for internal exposures from accidents. The systematic assessment of individual internal dose is not, therefore, warranted (see ACoP paragraph 386).

4.4 WORKING PROCEDURES AND SUPERVISION

4.4.1 Local rules

Regulation 17 requires that Local Rules are written for work in controlled areas. Augean should draft Local Rules, consulting the RPA as required, to ensure that the format and content of the rules (as specified in IRR99) are appropriate. The Local Rules must include:

- the dose investigation level;
- a description of each controlled area, and the means by which access is restricted;
- names of the Radiation Protection Supervisors (see below);
- for each controlled area, appropriate working instructions (PPE, good working practice, monitoring arrangements, etc) including written arrangements for the entry of non-classified persons into the controlled areas;
- details of any contingency arrangements, for example for dealing with spillages.

4.4.2 Radiation Protection Supervisors (RPSs)

Regulation 17 requires that Augean appoint one or more employees as RPS. The main role of the RPS is to ensure that the Local Rules are being observed, and whoever is appointed should be suitable for the role. In practice, this means that they are appropriately trained and are able to properly supervise the work being undertaken. There should be an RPS present on the ENRMF site whenever LLW is being processed.

It should be noted that the RPSs are not a substitute for line-management responsibilities. Augean must ensure that line managers involved in the LLW operation project are familiar with the contents of this risk assessment and the local rules, and their responsibilities for health and safety.

4.5 PERSONAL PROTECTIVE EQUIPMENT

The internal dose to Augean employees from inhalation of dust during normal operations is expected to be negligible. As indicated in Section 3.2, the inhalation dose from dealing with a waste spill is likely to be below 1 mSv. Although this is well below the 20 mSv/y dose limit, it is recommended that respiratory protective equipment be worn in the interests of keeping exposures ALARP, and to ensure compliance with the dose constraint and dose investigation level.

The RPE should be readily available in the event of a spill occurring, and must be put on before attempting to clean up any spilt LLW material.

RPE with a minimum protection factor of 5 is recommended: this, combined with the dust suppression measures described in Section 4.1.3, should ensure that inhalation doses are below 0.1 mSv. In addition:

- RPE must be CE marked;
- the comfort of the wearer should be taken into account when choosing a particular type of respirator;
- RPE should be fit-tested to ensure a good seal to individual faces;
- If the RPE is reusable, it should be thoroughly examined at suitable intervals and properly maintained in accordance with the manufacturer's instructions, and as required by Regulation 10(2). Suitable records of examinations and maintenance should be made and kept for at least 2 years; and (very importantly)
- training in the proper use and maintenance of RPE must be provided.

In addition to RPE, protective clothing should also be worn by Augean employees when working in the area, as follows:

- coveralls must be worn, the type being selected according to the nature of the work.
- protective gloves must be worn, the type being selected according to the nature of the work. Gloves should be impermeable and be sufficiently strong to withstand wear and tear and provide protection against cuts/wounds;
- footwear – normal safety footwear is considered sufficient; and
- suitable washing and changing facilities should be provided for use by workers before lunch breaks, ends of shift, etc.. This should include facilities for separate storage of

clean and dirty clothing, and hand/face washing facilities with elbow-operated taps. It is suggested that a contamination monitor should also be considered for reassurance purposes, i.e. so that workers can check themselves if they wish.

After dealing with a spill, coveralls and gloves may need to be disposed of. It is suggested that disposable outer coveralls and gloves should be provided for use when cleaning up spills. Gloves should be taped to coveralls where there is a risk of up-sleeve contamination during a clean-up. Footwear should be washed down before leaving the area.

4.6 INFORMATION, INSTRUCTION AND TRAINING

To meet the requirements of Regulation 14, the following arrangements are recommended:

- **All Augean employees** engaged in LLW work should receive training in radiation protection prior to the work. This should cover:
    - the nature of the radiation hazards associated with LLW;
    - the risks to health associated with exposure to radiation;
    - the precautions that need to be taken to restrict exposures, including the contents of this risk assessment and the local rules;
    - the correct use of RPE; and
    - the regulatory requirements associated with the work, and the importance of complying with these requirements.
- In addition, **specifically appointed Augean employees** should receive additional training to act as a Radiation Protection Supervisor(s) and (if applicable) to examine and maintain RPE.
- **Other persons working on the ENRMF site** should be provided with information to indicate that the certain areas are designated as controlled areas, that access to these areas is restricted, and that warning signs should be observed.

4.7 WORKPLACE MONITORING

The following programme of workplace monitoring is recommended.

Dose rates

- All incoming LLW containers should be subject to dose rate monitoring, and the results recorded. The dose rate a 1 metre from a container must not exceed 10 μSv/h.
- Any containers that do not meet the above criteria should be placed in quarantine. The dose rate around the perimeter of the quarantine area must be measured (and recorded) whenever containers are placed inside. The dose rate at the perimeter must not exceed 2 μSv/h.
- The dose rate on top of any newly deposited material must be measured after the minimum 300mm cover is applied. The dose rate at a height of 1 metre must not exceed 2 μSv/h. If necessary, additional cover should be applied. The measured dose rate and the thickness of cover applied should be recorded.

- Annual environmental-level dose rate monitoring will be undertaken by the RPA at representative locations around the site boundary.

Surface contamination monitoring

- Surface contamination is not expected to arise during routine operations. However, confirmatory monitoring should be undertaken once every month in the following areas:
    - At the exit point from the disposal area
    - After the vehicle wheel wash
    - Change rooms including PPE
    - At the main exit from the site.
- In addition, contamination monitoring should be undertaken after cleaning up any waste spillages. This should include:
    - Monitoring the affected area, i.e. to confirm that all contamination has been removed.
    - Monitoring all persons and items leaving the area to ensure that the spread of contamination is avoided.

P V Shaw

14 July 2009

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Version 3: 14 July 2009. Revised by RPA to incorporate comments.

APPENDIX TO ENRMF RISK ASSESSMENT

SUPPORTING RADIOLOGICAL DATA TAKEN FROM RSA93 APPLICATION

A.1 EXTERNAL DOSE DATA

WASTE IN CONTAINERS

Specific calculations were undertaken for cobalt-60 (the most restrictive radionuclide) and caesium-137 (for comparison) at 200 Bq/g – for both high and low density waste in drums and bulk bags. A summary of the results is given below.

Exposure scenario	Estimated dose rate (µSv/h)[2]	
	Cobalt-60	Caesium-137
Drums		
- contact (1 cm)	100	25
- 1 metre	6	1.5
- 2 metres	2	0.5
Bulk waste bags		
- contact (1 cm)	125	6
- 1 metre	14	3
- 2 metres	5	1

DEPOSITED WASTE

Specific calculations were also undertaken to estimate the dose rate above deposited waste covered with 30 cm of compacted topsoil. The results are summarised in the following table.

Radionuclides	Calculated dose rate (µSv/h)
Cobalt-60 at 200 Bq/g	5 to 10
Other radionuclides at 200 Bq/g	<1

In this assessment a maximum dose rate of 2 µSv/h above the covered waste has been recommended (see 4.1.2). This value has, therefore, been used (in part 2.1.1) to estimate doses to workers.

[2] The values have been rounded and represent the average dose rate calculated for high density (2g/cm^3) and low density (1 g/m^3) waste. In the case of cylindrical drums, the average values calculated for the (curved) sides and (flat) ends are given.

A.2 INTERNAL DOSE DATA - ICRP INTERNAL DOSE COEFFICIENTS

For consistency purposes, the data below are the same as those used in the RSA93 application, and are the relevant ICRP dose coefficients for members of the public. The ICRP dose coefficients for workers are slightly different, but this does not materially affect the outcome of this risk assessment.

Radionuclide	Dose coefficient (Sv/Bq)	
	Inhalation	Ingestion
H-3	2.6E-10	1.8E-11
C-14	5.8E-09	5.8E-10
Cl-36	7.3E-09	9.3E-10
Fe-55	7.7E-10	3.3E-10
Co-60	3.1E-08	3.4E-09
Ni-63	4.8E-10	1.5E-10
Sr-90	1.6E-07	2.8E-08
Nb-94	1.1E-08	1.1E-08
Tc-99	1.3E-08	6.4E-10
Ru-106	6.6E-08	7.0E-09
Ag-108m	3.7E-08	2.3E-09
Sb-125	5.5E-08	3.1E-09
Sn-126	3.1E-08	7.1E-09
I-129	3.6E-08	1.1E-07
Ba-133	3.1E-09	1.5E-09
Cs-134	6.8E-09	1.9E-08
Cs-137	3.9E-08	1.3E-08
Pm-147	5.0E-09	2.6E-10
Eu-152	4.2E-08	1.4E-09
Eu-154	5.3E-08	2.0E-09
Eu-155	6.9E-09	3.2E-10
Pb-210	5.6E-06	6.9E-07
Ra-226	9.5E-06	2.8E-07
Ac-227	5.5E-04	1.1E-06
Th-229	2.6E-04	6.1E-07
Th-230	1.0E-04	2.1E-07
Pa-231	1.4E-04	7.1E-07
Th-232	1.1E-04	2.3E-07
U-232	4.7E-05	4.6E-07
U-233	9.6E-06	5.1E-08
U-234	9.4E-06	4.9E-08
U-235	8.5E-06	4.7E-08
U-236	3.2E-06	4.7E-08
U-238	8.0E-06	4.5E-08
Np-237	5.0E-05	1.1E-07

Radionuclide	Dose coefficient (Sv/Bq)	
	Inhalation	Ingestion
Pu-238	1.1E-04	2.3E-07
Pu-239	1.2E-04	2.5E-07
Pu-240	1.2E-04	2.5E-07
Pu-241	2.3E-06	4.8E-09
Pu-242	1.1E-04	2.4E-07
Am-241	9.6E-05	2.0E-07
Cm-243	3.1E-05	1.5E-07
Cu-244	2.7E-05	1.2E-07

Appendix I. Dose Rate Calculations in Support of Low Level Waste Disposal Authorisation, TSG(09)0487 (UKAEA)

NOT PROTECTIVELY MARKED

Technical Services Group	Reference:	TSG(09)0487
	Issue:	Issue 2
	Date:	15th July 2009

DOSE RATE CALCULATIONS IN SUPPORT OF A LOW LEVEL WASTE DISPOSAL AUTHORISATION

UK-10497

SUMMARY

Dose rate calculations were performed in MicroShield to support a low level waste disposal authorisation. The dose rate was calculated on contact, 1m and 2m from a 200-litre drum and a bulk waste bag of soil and rubble waste. Dose was found to be highest when dealing with a ^{60}Co source.

	Name and Organisation	Signature	Date
Prepared By:	Tony Lansdell TSG	ELECTRONIC COPY	
Checked By:	Barry Cook TSG		
Approved By:	Gráinne Carpenter TSG		

Table of Contents

1 INTRODUCTION

Dose rate calculations were required to support a low level waste disposal authorisation. Cases were run using MicroShield v7.02 [1] to determine the maximum dose rate at a series of distances from the wasteform, for two different wasteform geometries.

2 METHODOLOGY

2.1 Background

MicroShield was used to determine the maximum dose rates at various distances from packaged contaminated soil and rubble waste. Two cases were defined, one for waste packaged in a 200-litre drum, and one for waste in a flexible bulk waste bag. In each case, the dose rate was required on contact, at 1m and at 2m from the wasteform.

2.2 Materials

Two sub-cases were defined; one for soil/rubble waste containing 200 Bq/g of ^{60}Co, and one for soil/rubble containing 200 Bq/g of ^{137}Cs. As soil is not a material type available to MicroShield, concrete was chosen to represent the waste material composition.

The bulk density of the soil and rubble wastes will vary depending on the composition of the waste, the level of compaction used, and the packing efficiency. Cases were assessed for two wasteform density values to provide bounding results.

A search of literature revealed that the bulk density of soil was typically 1.0 g/cm^3 for loose soil, 1.3 g/cm^3 for undisturbed soil, and 1.6 g/cm^3 for compacted soil [2]. Concrete rubble was assumed to be the same as normal density concrete, 2.35 g/cm^3.
The minimum density case was taken to be packaged loose soil, with a density of 1.0 g/cm^3.

The maximum density wasteform was taken to contain the maximum amount of concrete rubble, with the remaining space taken up by compacted soil. It was assumed solid pieces of rubble would have a packing efficiency no better than 50%, hence 50% of the volume was assumed to be rubble (2.35 g/cm^3), with the remaining 50% consisting of compacted soil (1.6 g/cm^3). The maximum density of the wasteform was therefore predicted to be 1.98 g/cm^3, and 2.0 g/cm^3 was used for simplicity. The maximum range of wasteform density used was therefore between 1 and 2 g/cm^3.

2.3 200-litre drum case

200-litre drums are steel-walled cylindrical drums of diameter of 67 cm and height 87 cm. The shielding effect of the drum was ignored to be conservative, hence the drum wall was not modelled in MicroShield, and the wasteform was taken to be a cylindrical volume of the above drum dimensions. Dose points were positioned on contact, 1m, and 2m from the wasteform, both in a radial direction (Figure 1) and an axial direction (Figure 2). Radial dose points were located at half the height of the cylinder, where the dose rate is maximised. Axial dose points were on axis with the centre of the cylinder, where the dose rate is maximised.

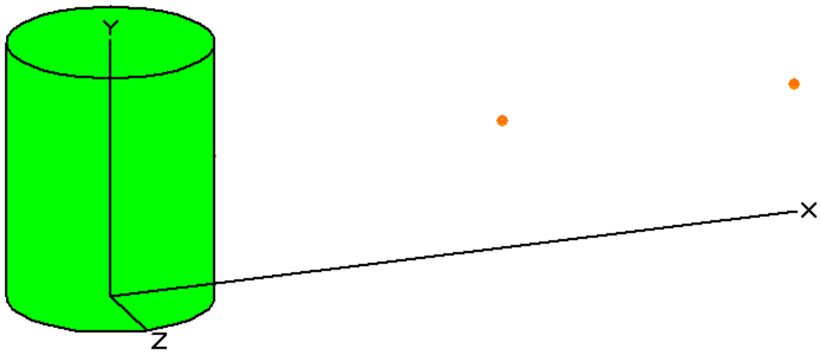

Figure 1: Radial dose points for 200-litre drum (images from MicroShield)

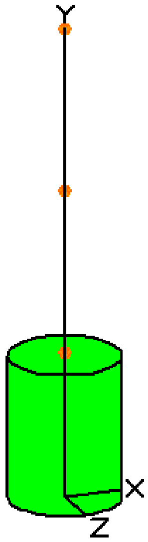

Figure 2: Axial dose points for 200-litre drum

2.4 Bulk waste bag case

The bulk waste bag is a cube of side length 1m, and the wasteform was modelled in MicroShield as a rectangular volume with all sides 1m in length (Figure 3). Again, the wall of the bag was not explicitly modelled to be conservative. The dose points were positioned in line with the centre of a flat face, where the dose rate is maximised.

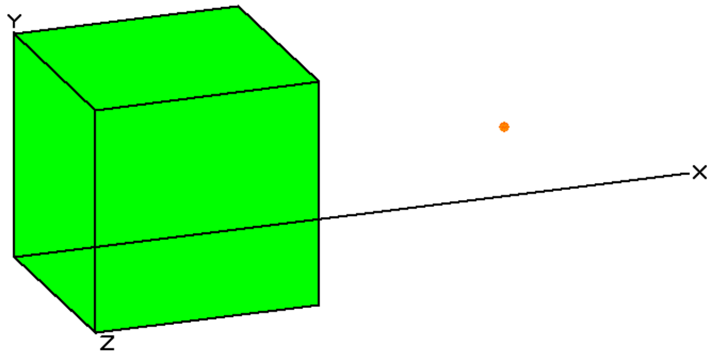

Figure 3: Dose points for bulk waste bag

2.5 MicroShield calculation details and uncertainties

Energy deposition to dose rate conversion was performed automatically in MicroShield using built-in tables of effective dose rate, taken from ICRP-51 [3]. This presents a series of possible dose rates depending on the assumed irradiation geometry. The highest biological dose rate is produced assuming anterior-posterior geometry (with the gamma rays entering a person from the front and exiting through the back), and to be conservative it was this maximum dose rate that was reported. Dose rates can vary by approximately 30%, depending on which geometry is assumed.

MicroShield approximates the contribution of scattered radiation to the resulting dose rate by the use of build-up tables. The dose rate is dependent on which material is chosen as the dominant scattering medium. In accordance with the MicroShield manual, the material containing the highest number of gamma ray mean free paths should be used as the build-up material – hence in these cases, the source was chosen as build-up material. If the air gap is chosen as the scattering medium, it was found that the resulting dose rates increased by 6% for ^{60}Co cases, and increased by 12% for ^{137}Cs cases, but these results would be over-pessimistic.

MicroShield uses a point-kernel integration technique to determine the dose rate. This involves splitting the geometry into pieces (kernels). The quadrature order of the calculation determines the number of kernels used and hence the accuracy of the approximation; the default quadrature order was used for the reported results. The order of the calculation was increased by a factor of two in each dimension, and the contact results only increased by 0.3%, which is well within the range of other sources of uncertainty in the calculation. Further increases in accuracy produced no change to the results.

In all cases assessed, the 'contact' dose rate point was actually positioned at 1 cm from the surface, as the method of calculation used by MicroShield is known to become unstable at distances closer than 1 cm, though this will strongly depend on the integration order used..

^{137}Cs is a beta emitter. Its daughter, ^{137m}Ba is the source of the gamma radiation. Where a source containing ^{137}Cs was specified, its daughter product ^{137m}Ba was also included in equilibrium concentration with ^{137}Cs. Since the half-life of ^{137m}Ba is short (2.5 minutes), it will almost always be found in equilibrium with its parent radionuclide.

3 RESULTS

3.1 Low density case

Density = 1.0 g/cm^3, specific activity = 200 Bq/g = 200 Bq/cm^3.

Case	Dose Rate (μSv/hr)						
	Curved cylinder face				Flat cylinder face		
	Contact*	100 cm	200 cm		Contact*	100 cm	200 cm
^{60}Co	91.65	6.019	1.95		98.35	4.766	1.465
^{137}Cs	21.6	1.421	0.458		23.57	1.109	0.335

Case	Dose Rate (μSv/hr)		
	Flat cube face		
	Contact*	100 cm	200 cm
^{60}Co	120.7	12.45	4.065
^{137}Cs	27.6	2.875	0.925

3.2 High density case

Density = 2 g/cm^3, specific activity = 200 Bq/g = 400 Bq/cm^3.

Case	Dose Rate (μSv/hr)						
	Curved cylinder face				Flat cylinder face		
	Contact*	100 cm	200 cm		Contact*	100 cm	200 cm
^{60}Co	109.2	7.293	2.316		123.9	5.682	1.654
^{137}Cs	24.35	1.639	0.515		28.01	1.283	0.368

Case	Dose Rate (μSv/hr)		
	Flat cube face		
	Contact*	100 cm	200 cm
^{60}Co	131.3	14.5	4.526
^{137}Cs	28.72	3.261	1.01

** Contact doses were located at 1 cm from the wasteform.*

4 REFERENCES

1 MicroShield v7.02, Grove Software Inc, 2007
2 "Soils and Soil Fertility", page 54, Sixth Edition, F.R. Troeh and L.M. Thompson, Blackwell Publishing, 1979.
3 ICRP-51 (1987) Data for use in protection against external radiation

Appendix J. Calculation of Dose Rate at Landfill, TSG(09)0488 (UKAEA)

Technical Services Group	Reference:	TSG(09)0488
	Issue:	Issue 2
	Date:	15th July 2009

CALCULATION OF DOSE RATE AT LANDFILL IN SUPPORT OF A LOW LEVEL WASTE DISPOSAL AUTHORISATION

UK-10497

SUMMARY

Dose rate calculations were performed in MicroShield to support a low level waste disposal authorisation. An estimate of the dose rate at the landfill site was calculated based on lightly-contaminated rubble being covered by a 30 cm layer of soil material. Dose was found to dependent on the soil material density and largely independent on the distance from the source.

	Name and Organisation	Signature	Date
Prepared By:	Tony Lansdell TSG	ELECTRONIC COPY	
Checked By:	Barry Cook TSG		
Approved By:	Gráinne Carpenter TSG		

Table of Contents

1 INTRODUCTION

Dose rate calculations were required to support a low level waste disposal authorisation. Cases were run using MicroShield v7.02 [1], to determine the dose rate above the layer of lightly contaminated soil at a landfill waste disposal site.

2 METHODOLOGY

2.1 Background

MicroShield was used to determine the maximum resulting dose rate from disposal of soil and rubble to a landfill site. This can be assumed to be uniformly contaminated to 200 Bq/g of either ^{60}Co or ^{137}Cs. An infinite slab of contaminated soil and rubble was assumed to be covered with 30 cm of uncontaminated soil material, and the dose rate assessed.

2.2 Case details

MicroShield was used to model a slab of waste, infinite in horizontal extent, and 100 cm thick. Preliminary study found that if the slab thickness was increased above 50 cm thick, the resulting dose rate was effectively unchanged, and hence a thickness of 100 cm was used to be conservative.

Preliminary studies also indicated that when dose rate was determined on contact, 1m and 2m above the shielding soil layer, dose rate was independent of dose point height and so only the contact dose was reported. This work can be found in Appendix A.

It has been outlined that MicroShield did not correctly include the effects of build-up (scattered flux) when using the infinite slab geometry, and that the calculation was instead performed using a (finite) rectangular slab that was chosen to have a very large extent such that it was effectively infinite. The extent was chosen such that the results were unchanged with further increases in size, and it was found that beyond 200 cm in width the dose rate on contact was effectively constant, and 1000 cm was chosen to be conservative.

After these initial tests, on the assumption that dose rates were in the worst case for each nuclide greater than 2.5 µSv/hr, it was to be found what thickness of soil material would result in a dose rate of 2.5 µSv/hr.

2.3 MicroShield calculation details, uncertainties and assumptions

Energy deposition to dose rate conversion was performed automatically in MicroShield using built-in tables of effective dose rate, taken from ICRP-51 [2]. This presents a series of possible dose rates depending on the assumed irradiation geometry. The highest biological dose rate is produced assuming anterior-posterior geometry (with the gamma rays entering a person from the front and exiting through the back), and to be conservative it was this maximum dose rate that was reported. Dose rates can vary by approximately 30%, depending on which geometry is assumed.

MicroShield approximates the contribution of scattered radiation to the resulting dose rate by the use of build-up tables. The dose rate is dependent on which material is chosen as the dominant scattering medium. In accordance with the MicroShield manual, the material containing the highest number of gamma ray mean free paths should be used as the build-up material – hence in these cases, the source was chosen as build-up material. It was found that choosing the shielding soil as the build-up material produced identical results; hence the results are insensitive to this assumption.

MicroShield uses a point-kernel integration technique to determine the dose rate. This involves splitting the geometry into pieces (kernels). The quadrature order of the calculation determines the number of kernels used and hence the accuracy of the approximation, at the expense of a longer calculation time. Due to the extent of the source relative to the dose rate distance, the quadrature order was increased in the y and z axes until the result was unchanged. Beyond a quadrature of 30, the results were unchanged, and 50 was used to be conservative.

In all cases assessed, the ‘contact’ dose rate point was actually positioned at 1 cm from the surface, as the method of calculation used by MicroShield is known to become unstable at distances closer than 1 cm. The dose rate was found to be nearly independent of distance, with only a 1-2% drop in dose rate from contact to 2m, hence the results are insensitive to this assumption as well.

3 RESULTS

3.1 Cobalt-60 case

The contact dose rate from high density (2 g/cm^3) soil containing 200 Bq/g ^{60}Co covered with 30 cm of uncontaminated soil was determined for a series of soil material densities from 1.0 to 1.6 g/cm^3. The results are given in Table 1.

Soil material density (g/cm^3)	Contact dose rate (µSv/hr)
1.0	18.35
1.2	13.02
1.4	9.28
1.6	6.63

Table 1: Contact dose rates for various soil material densities

The resulting dose rate above the shielding layer of soil will be between 18.35 µSv/hr for loose soil and 6.63 µSv/hr if the shielding surface soil has been compacted.

3.2 Caesium-137 case

The contact dose rate from high density (2 g/cm^3) soil containing 200 Bq/g ^{137}Cs covered with 30 cm of uncontaminated soil was determined for a series of soil material densities from 1.0 to 1.6 g/cm^3. ^{137}Cs is a beta emitter. Its daughter, ^{137m}Ba is the source of the gamma radiation. Where a source containing ^{137}Cs was specified, its daughter product ^{137m}Ba was also included in equilibrium concentration with ^{137}Cs. Since the half-life of ^{137m}Ba is short (2.5 minutes), it will almost always be found in equilibrium with its parent radionuclide. The results are given in Table 2.

Soil material density (g/cm^3)	Contact dose rate (µSv/hr)
1.0	2.58
1.2	1.67
1.4	1.08
1.6	0.70

Table 2: Contact dose rates for various soil material densities

The resulting dose rate above the shielding layer of soil will be between 2.58 µSv/hr for loose soil and 0.70 µSv/hr if the shielding surface soil has been compacted.

3.3 Covering soil material thickness

Following on from the scenario outlined in Section 3.1, the contact dose rate from high density (2 g/cm^3) contaminated soil and rubble containing 200 Bq/g ^{60}Co at a density of 2 g/cm^3 covered by uncontaminated soil of various thickness was determined to find a relationship between the shielding material thickness and dose rate. The density of the covering soil material was taken as 1 g/cm^3, which presented the worst case in section 3.1. Figure 1 and Table 3 show the relationship between dose and soil material thickness.

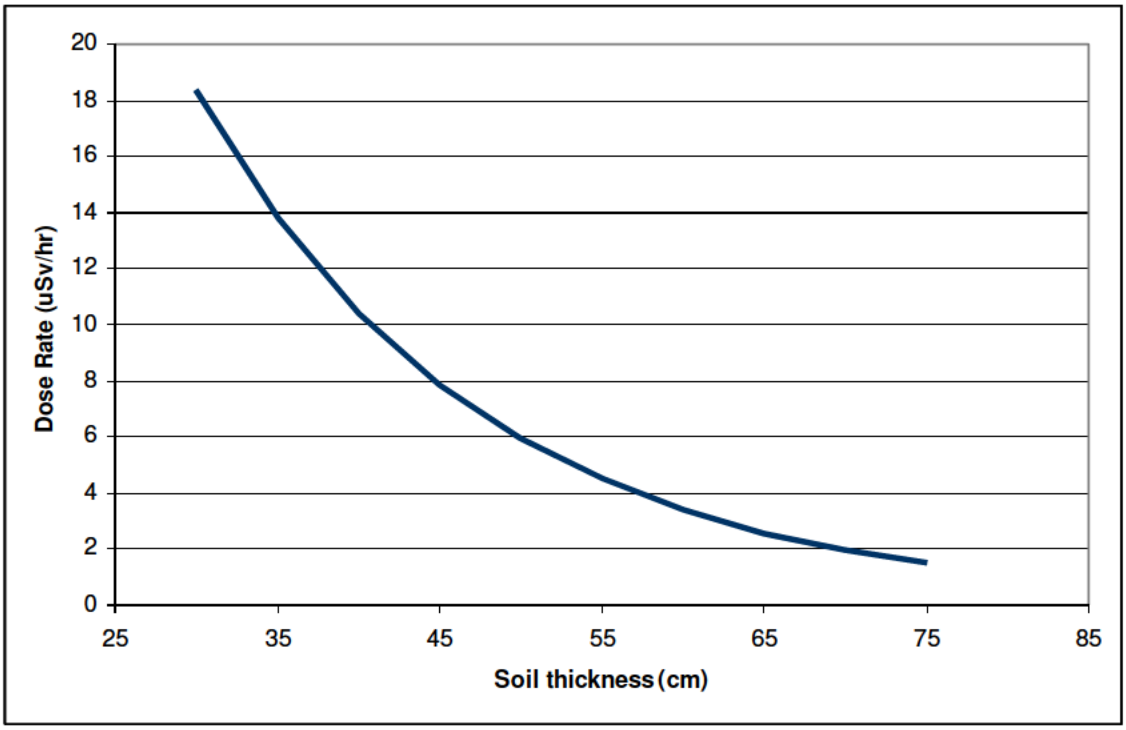

Figure 1: The relationship between soil thickness and contact dose rate for soil and rubble contaminated by ^{60}Co isotopes resulting in a uniform activity of 200 Bq/g

Soil Thickness (cm)	Dose Rate (µSv/hr)
30	18.35
35	13.78
40	10.39
45	7.84
50	5.93
55	4.49
60	3.40
65	2.58
70	1.96
75	1.48

Table 3: Tabulated data for Figure 1

These data show that to get a dose rate of 10 µSv/hr, the soil material must be at least 40cm thick, and to get a dose rate of 2.5 µSv/hr, the soil material must be at least 65cm thick [3].

4 REFERENCES

1 MicroShield v7.02, Grove Software Inc, 2007
2 ICRP-51 (1987) Data for use in protection against external radiation
3 Personal communication, Paul Atyeo, 23rd March, 2009

APPENDIX A – Calculations to show that dose is geometry and air distance independent

Depth: 200cm Length:
2000cm Breadth: 2000cm

Based on the soil material having a thickness of 30cm and a density of 1 g/cm3

Dose point	Dose (µSv/h)
'Contact'	18.35
1m	18.21
2m	17.82

Appendix K. Application form

Application for an environmental permit for a radioactive substances activity Part RSR-A – About you and your premises

Environment Agency

Please read through this part of the form and the Part RSR-A guidance notes before you fill it in.

It will take about two hours to fill in this form.

Fill in this part of the form for all applications relating to a radioactive substances activity.

Please keep a copy of this part of the form once you have filled it in for the first time. You will be able to reuse it if you need to apply for a variation or another radioactive substances activity permit.

If you are reusing this part of the form, please highlight any changes you have made to the information you provided previously. You do not need to resubmit any documents (questions 9c, 12, 13) unless something has changed.

For a transfer application, this part of the form should be filled in by the person the permit will be transferred to.

Contents

1 Site reference number

Fill in your site reference number if you know it.

Site reference number

2 About you

Are you applying as:

an individual ☐
Now go to section 3

an organisation of individuals (for example, a partnership) ☐
Now go to section 4

a registered company or limited liability partnership ☑
Now go to section 5

a public body or other corporate body ☐
Now go to section 6

3 Applications from an individual

3a Please give us the following details

Title (Mr, Mrs, Miss and so on)

First name

Last name

Now go to section 7.

4 Applications from an organisation of individuals

4a Type of organisation (for example, partnership)

4 Applications from an organisation of individuals, continued

4b Details of the organisation

Please give the details of the main representative of the organisation below. Provide details of the other members on a separate sheet and tell us the document reference you have given this sheet.

Document reference

Title (Mr, Mrs, Miss and so on)

First name

Last name

Now go to section 7.

5 Applications from companies or limited liability partnerships (LLPs)

5a Name of the company or LLP
Augean South Limited

5b Company/LLP registration number
4636789

Now go to section 7.

6 Applications from public or other corporate bodies

6a Type of organisation (for example, NHS trust, university)

6b Name of the organisation

6 Applications from public or other corporate bodies, continued

6c Position of the person who acts as the secretary or clerk of the organisation

7 Your address

7a Your main (registered office) business address
For companies or LLPs this is the address on record at Companies House.

Address

[redacted]

Contact numbers, including the area code

Phone [redacted]

Mobile

Fax

Email

7b Main UK business address if different from above

Address

Postcode

Contact numbers, including the area code

Phone

Mobile

Fax

Email

Position of most senior person at this address

8 Contact details

8a Who can we contact about this application?

Title (Mr, Mrs, Miss and so on) Dr

First name Gene

Last name Wilson

8 Contact details, continued

Address

East Northants Resource Management Facility

Stamford Road

Kings Cliffe

Northamptonshire

Postcode PE8 6XX

Contact numbers, including the area code

Phone [redacted]

Mobile

Fax

Email

8b Who can we contact about your radioactive substances activity?

Title (Mr, Mrs, Miss and so on) Dr

First name Gene

Last name Wilson

Address

East Northants Resource Management Facility

Stamford Road

Kings Cliffe

Northamptonshire

Postcode PE8 6XX

Contact numbers, including the area code

Phone [redacted]

Mobile

Fax

Email

8c Who can we contact about your billing/invoice?

Title (Mr, Mrs, Miss and so on) Dr

First name Gene

Last name Wilson

Address

East Northants Resource Management Facility

Stamford

Kings Cliffe

Northamptonshire

Postcode PE8 6XX

8 Contact details, continued

Contact numbers, including the area code

Phone [redacted]

Mobile

Fax

Email

9 About the premises

9a What is the name and address of the premises where you intend to carry out a radioactive substances activity?

If you only keep and use mobile radioactive apparatus, give details of the premises where that apparatus is normally kept when not in use.

Name: East Northants Resource Management Facility

Address: Stamford Road
Kings Cliffe
Northamptonshire

Postcode: PE8 6XX

National grid reference for the premises
For example, ST 12345 67890: TF 010 000

9b Is a nuclear site licence under section 1 of the Nuclear Installations Act 1965 needed for the premises?

No ☑

Yes ☐ What is your status?

Licensee (or potential licensee) ☐

Tenant ☐

9c Please provide a plan of the site, marking the site boundary in green

This is not required if you only keep and use mobile radioactive apparatus.

Document reference: see Figure 5 of Environmental Safety Case (ESC)

10 Consultation

10a Which local authority area are the premises in?

Give the name of your district council, borough council, city council, metropolitan district council, Welsh county borough, Welsh county council, London borough or other unitary authority, as appropriate.: East Northamptonshire Council

10b Who is the sewerage undertaker for the premises?

You do not need to answer this if you only need a standard rules permit, or your premises are on a nuclear licensed site and you do not discharge radioactive waste to public sewer.

Name: Leachate Tankered to off-site treatment works - see ESC 2.3

11 Justification status

11a Does your work with radioactive materials and/or radioactive waste relate to:

11a.1 An existing practice, or work that is not subject to the requirement for justification?

No ☐

Yes ☑ Go to question 11b

11 Justification status, continued

11a.2 A new practice that the Justifying Authority has determined to be justified?

No ☐

Yes ☐ Give date and reference number of decision (DD/MM/YYYY), then go to section 12

Document reference

11a.3 A practice that is currently being considered by the Justifying Authority?

No ☐

Yes ☐ Give date and reference number of application (DD/MM/YYYY), then go to section 12

Document reference

11b Nature of practice or work

11b.1 Tell us the number and purpose of the practice which applies to your work with radioactive material and/or radioactive waste

See the Government guidance. If there is more than one practice that applies to you, give the information for each one.

Number	Purpose

11b.2 If your work is not listed in the Government guidance, tick the appropriate box to show if it involves any of the following:

You use NORM (substances that are naturally radioactive) as a chemical in a laboratory ☐

You create NORM as a result of producing gas and oil ☐

You use NORM for some other reason ☐

The Ministry of Defence (MOD) or the armed forces use radioactive substances on the premises ☐

A contractor to the MOD uses radioactive substances for military purposes ☐

Other ☑ Please give details below

Radioactive waste is accepted for disposal (Environmental Permit CD8503) but not produced at the premises. The premises are the final disposal facility for radioactive waste of low specific activity produced from various justified practices where disposal by landfill is demonstrated to be BAT over other options . The premises are an existing permitted hazardous waste disposal landfill (Environmental Permit TP3430GW) and the site is the subject of a Development Consent Order made by the Secretary of State on 10 July 2013 (The East Northamptonshire Resource Management Facility Order 2013).

11c Associated activities

Tick the appropriate boxes to show which activities associated with the practice(s) are carried out on your premises

Research and development ☐

Manufacturing products ☐

Carrying out repairs ☐

Carrying out maintenance ☐

Supplying radioactive substances ☐

Assembling items that include radioactive substances ☐

Handling radioactive substances ☐

Testing radioactive substances for quality standards ☐

11c Associated activities, continued

Storing radioactive substances ☐
Using radioactive substances ☐
Disposing of waste ☑
Other ☐ Please give details below

12 Your ability as an operator – management systems

You do not need to answer this if you only need a standard rules permit or are applying to surrender your permit.

Provide a summary of your management system or, if you are a nuclear site licensee, provide your management prospectus.

Document reference or references: ESC: Section 5

Is your management system accredited?

No ☐

Yes ☑ Under what scheme or standard? ISO 9001, ISO 14001, OHSAS 18001, PAS 99

13 Existing site contamination

Tell us about any existing contamination on the premises, if appropriate (see guidance)

Document reference

Now go to

Part RSR-B1 if you are applying for a standard rules permit.

Part RSR-B2 if you are applying for a bespoke permit to carry out a radioactive substances activity involving sealed sources (including waste sealed sources).

Part RSR-B3 if you are applying for a permit to carry out a radioactive substances activity, on a nuclear site, involving radioactive material (open sources) and/or radioactive waste. (Part RSR-B5 may also be necessary.)

Part RSR-B4 if you are applying for a permit to carry out a radioactive substances activity, not on a nuclear site, involving radioactive material (open sources) and/or radioactive waste. (Part RSR-B5 may also be necessary.)

Part RSR-B5 if you are applying for a permit to carry out a radioactive substances activity involving on-site disposal of solid radioactive waste to land.

Part RSR-C2 if you are applying to vary a permit for a radioactive substances activity involving sealed sources (including waste sealed sources).

Part RSR-C3 if you are applying to vary a permit for a radioactive substances activity, on a nuclear site, involving radioactive material (open sources) and/or radioactive waste.

Part RSR-C4 if you are applying to vary a permit for a radioactive substances activity, not on a nuclear site, involving radioactive material (open sources) and/or radioactive waste.

Part RSR-C5 if you are applying to vary the conditions of a permit relating to on-site disposal of solid radioactive waste to land.

Part RSR-D2 if you are applying to transfer a permit for a radioactive substances activity involving sealed sources (including waste sealed sources).

Part RSR-D3 if you are applying to transfer a permit for a radioactive substances activity involving radioactive material (open sources) and/or radioactive waste (including on-site disposal of radioactive waste).

Part RSR-E2 if you are applying to surrender a permit for a radioactive substances activity involving sealed sources (including waste sealed sources).

Part RSR-E4 if you are applying to surrender a permit for a radioactive substances activity, involving radioactive material (open sources) and/or radioactive waste (including on-site disposal of radioactive waste).

You will also need to complete part RSR-F.

14 How to contact us

If you need help filling in this form, please contact the person who sent it to you or contact us as shown below.

Premises not on a nuclear site
Phone: 01142 800 678 or 01142 800 682
Email: RSR.Rotherham2.NE@environment-agency.gov.uk

Premises on a nuclear site

Nuclear regulatory group (North)
Phone: 01768 215 991
Email: nrg.north@environment-agency.gov.uk

Nuclear regulatory group (South)
Phone: 01491 828 629
Email: nrg.south@environment-agency.gov.uk

Website: www.environment-agency.gov.uk

If you are happy with our service, please tell us. It helps us to identify good practice and encourages our staff. If you're not happy with our service, please tell us how we can improve it.

Feedback

(You don't have to answer this part of the form, but it will help us improve our forms if you do.)

We want to make our forms easy to fill in and our guidance notes easy to understand. Please use the space below to give us any comments you may have about this form or the guidance notes that came with it.

How long did it take you to fill in this form?

We will use your feedback to improve our forms and guidance notes, and to tell the Government how regulations could be made simpler.

For Environment Agency use only

Date received (DD/MM/YYYY)

Our reference number

Payment received?

No ☐

Yes ☐ Amount received

£

Do you want to remove the 'Restricted – Regulatory' text from the header and footer?

Yes ☐

No ☐

Application for an environmental permit Part RSR-B5 – New bespoke radioactive substances activity permit (burial of radioactive waste)

Environment Agency

This form has been completed as an application to vary Environmental Permit CD8503, dated 25 May 2011. It is intended to replace form RSR-C5.

Please read through this part of the form and the part RSR-B5 guidance notes before you fill it in.

If you are applying for a new bespoke permit for a radioactive substances activity involving the burial of radioactive waste, fill in this part of the form, together with parts RSR-A and RSR-F. If you want to disposal of radioactive waste other than by burial, also fill in part RSR-B3.

We have published guidance on the disposal of radioactive waste in near surface facilities at http://publications.environment-agency.gov.uk/pdf/GEHO0209BPJL-e-e.pdf.

We strongly advise you to read that guidance and the guidance to this form, and then to discuss your proposals with us before you make an application.

It will take less than three hours to fill in this part of the application form.

Contents

The ESC document referred to below is: "Environmental Safety Case: Disposal of low level radioactive waste at East Northants Resource Management Facility. July 2015. ENE 154/001 Eden Nuclear and Environment Ltd."

1 Other applications

1a Have you recently made, or do you intend to make, an application for an environmental permit to operate a regulated facility, other than a radioactive substances activity, on the premises?

No ☐
Yes ☑

2 About the activities

2a What activities are you applying for?

Tick the relevant boxes in Table 1 to show which radioactive substances activities you are applying for.

Table 1 – radioactive substances activities

Schedule 23 Part 2 paragraph reference	Description	
5(2)(b)	Dispose of radioactive waste on or from premises used for the purposes of an undertaking	☐
5(4)(a)	Receive radioactive waste for the purposes of disposing of it	☑
5(2)(c)	Accumulate radioactive waste on the premises (non-nuclear sites only)	☐

2b Is a submission to the European Commission under Article 37 of the Euratom treaty required for these activities?

No ☐
Yes ☑ If yes, what is its status?

In draft ☐
Submitted to EC ☑
Give date of submission (DD/MM/YYYY) 06/07/2015
Opinion received ☐
Give date of opinion (DD/MM/YYYY)

2c Provide a technical description of your activities

Document reference ESC: Sections 1 and 2

3 Operating techniques

3a Describe how you manage the disposal of radioactive waste by burial to protect the environment and to optimise the protection of members of the public

Document reference: ESC: Section 1.5, Section 5.2.5, Section 6.4

3b Describe how you manage the disposal of radioactive waste by burial to protect members of the public and the environment from any non-radiological hazards of the radioactive waste

Document reference: ESC: Section 7.1: Site has Permit for hazardous wastes

4 Disposal of radioactive waste

4a Provide a description and quantitative estimates of the radioactive waste to be disposed of by burial

Document reference: ESC: Section 3, Appendix G for illustrative inventories

4b Provide your proposed limits for the disposal of radioactive waste

Document reference: ESC: Section 7.4 -Table 26

5 Monitoring

5a Provide a description of the sampling arrangements, techniques and systems for measurement and assessment of discharges of radioactive waste and other emissions from the facility

Document reference: ESC: Section 7.5

5b Provide a description of your environmental monitoring programme

Document reference: ESC: Section 7.5

6 Radiological assessment

6a Provide a prospective dose assessment at the proposed limits for burial

Document reference: ESC: Appendix E, Appendix G for illustrative inventories

6b Provide an assessment of the impact on non-human species at the proposed limits for burial

Document reference: ESC: Appendix E - Section E6

7 Non-radiological assessment

7a Provide an assessment of the impacts arising from the non radiological properties of the radioactive waste

Document reference: Not applicable: Site has Permit for hazardous waste

8 Radioactive waste acceptance criteria

8a Provide details of your waste acceptance criteria and procedures for the control of the receipt and burial of radioactive waste

Document reference: ESC: Section 7.4

Now fill in part RSR-F.

9 How to contact us

If you need help filling in this form, please contact the person who sent it to you or contact us as shown below.

Nuclear regulatory group (North)
Phone: 01768 215991
Email: nrg.north@environment-agency.gov.uk

Nuclear regulatory group (South)
Phone: 01491 828629
Email: nrg.south@environment-agency.gov.uk

Website: www.environment-agency.gov.uk

If you are happy with our service, please tell us. It helps us to identify good practice and encourages our staff. If you're not happy with our service, please tell us how we can improve it.

For Environment Agency use only

Date received (DD/MM/YYYY)

Our reference number

Payment received?

No ☐

Yes ☐ Amount received

£

Application for an environmental permit (radioactive substances activity) Part RSR-F – Charges and declarations

Please read through this part of the form and the part RSR-F guidance notes before you fill it in.

It will take about two hours to fill in this part of the form.

Fill in this part for all applications for a radioactive substances activity.

Contents

1 Permit type

If your application relates to permit type G or H (and is not an application for transfer or surrender) provide a copy of your calculations showing how you determined the permit type.

Document reference

2 Working out charges

2a Is your application for a variation to change a fixed condition registration to a standard rules permit?

Yes ☐ Answer question 2e, then go to section 4 (there is no charge)

No ☑

2b Is your application for an administrative variation only?

Yes ☐ Go to section 4 (there is no charge)

No ☑

2c Does your application relate to a radioactive substances activity on a nuclear licensed site?

Yes ☐ Go to section 4 (we will charge you on a time and materials basis)

No ☑

2d Does your application relate to the disposal of solid low-level radioactive waste (including high-volume very low-level waste) to land (either at a conventional landfill site or at a dedicated radioactive waste disposal site)?

Yes ☑ Go to section 4 (we will charge you on a time and materials basis)

No ☐ Fill in the table below

RSR Permit type (see note 1)	Application type (see note 2)	Charges due (£) (see note 3)

Note 1 A – H as described in the charging scheme guidance.

Note 2 New, variation, transfer or surrender.

Note 3 As specified in the charging scheme guidance. Please print or copy this page as confirmation of the application charge payable and for use in raising the payment. We will not be sending you an invoice to cover this charge.

2e If your permit type is A, is each source you hold a gaseous tritium light device?

Yes ☐ (this does not affect the application fee but may affect your subsistence charge)

No ☐

2 Working out charges, continued

2f If you are claiming the reduced fee for a 'minor technical' variation (permit types D, G and H only), give your reasons

2g If you are claiming the reduced fee for a surrender application (permit types G and H only), give your reasons

3 Payment

Tick below to show how you will make the payments.

Cheque ☐

Postal order ☐

Credit or debit card ☐

Electronic transfer (for example, BACS) ☐

Expected date of transfer (DD/MM/YYYY) ______

How to pay

Paying by cheque or postal order

Cheque details

Cheque made payable to ______

Cheque number ______

Amount £ ______

You should make cheques or postal orders payable to 'Environment Agency' or 'Environment Agency Wales' as appropriate and make sure they have 'A/c Payee' written across them if it is not already printed on.

Please write the name of your company and a reference number (this can be the site reference or permit reference – contact us if you don't know either of these) on the back of your cheque or postal order.
We will **not** accept cheques with a future date on them.

Paying by credit or debit card

If you are paying by credit or debit card, please fill in the separate form CC1 (available from www.environment-agency.gov.uk/business/sectors/117043.aspx). We will destroy your card details once we have processed your payment. We can accept payments by Visa, MasterCard or Maestro cards only.

Paying by electronic transfer

If you choose to pay by electronic transfer:

- use the relevant information (dependent on whether the premises your application relates to is in England or Wales) from the table below to make your payment;
- payments made from outside the United Kingdom must be in sterling – use the relevant IBAN/SWIFTBIC number;
- also send your payment details and a reference number (this can be the site reference or permit reference – contact us if you don't know either of these) to the relevant email address or fax number.

If you do not quote your reference number (this can be the site reference or permit reference), there may be a delay in processing your payment and application.

3 Payment, continued

	Premises in England	Premises in Wales
Company name	Environment Agency	Environment Agency Wales
Company address	Income Dept 311, PO Box 263, Peterborough, PE2 8YD	PO Box 663, Cardiff, CF24 0TP
Bank	Citibank, Citigroup Centre, Canada Square, London, E14 5LB	Citibank, Citigroup Centre, Canada Square, London, E14 5LB
Sort code	08-33-00	08-33-00
Account number	12800543	12800578
Payment reference number	xxxxxxxxxxxxx	xxxxxxxxxxxxx
IBAN number	GB23 CITI0833 0012 8005 43	GB48 CITI0833 0012 8005 78
SWIFTBIC number	CITI GB2LXXX	CITI GB2LXXX
Email details to	FSC-Income@environment-agency.gov.uk and RSR.Rotherham2.NE@environment-agency.gov.uk	online@environment-agency.wales.gov.uk and RSR.Rotherham2.NE@environment-agency.gov.uk
or fax details to	01733 464 892	02920 466 404

4 The Data Protection Act 1998

We, the Environment Agency, will process the information you provide so that we can:

- deal with your application;
- make sure you keep to the conditions of the licence, permit or registration;
- process renewals; and
- keep the public registers up to date.

We may also process or release the information to:

- offer you documents or services relating to environmental matters;
- consult the public, public organisations and other organisations (for example, the Health and Safety Executive, local authorities, the emergency services, the Department for Environment, Food and Rural Affairs) on environmental issues;
- carry out research and development work on environmental issues;
- provide information from the public register to anyone who asks;
- prevent anyone from breaking environmental law, investigate cases where environmental law may have been broken, and take any action that is needed;
- assess whether customers are satisfied with our service, and to improve our service; and
- respond to requests for information under the Freedom of Information Act 2000 and the Environmental Information Regulations 2004 (if the Data Protection Act allows). We may pass the information on to our agents or representatives to do these things for us.

5 Confidentiality and national security

We will normally put all the information in your application on a public register of environmental information. However, we may not include certain information in the public register if it's in the interests of national security, or because the information is confidential.

You can ask for information to be confidential by enclosing a letter with your application giving your reasons. If we agree with your request, we will tell you and not include the information in the public register. If we do not agree with your request, we will let you know how to appeal against our decision, or you can withdraw your application.

You can tell the Secretary of State (premises in England) or Welsh Ministers (premises in Wales) that you believe including information on a public register would not be in the interests of national security. You must enclose a letter with your application telling us that you have told the Secretary of State or Welsh Ministers and you must still include the information in your application. We will not include the information in the public register unless the Secretary of State or Welsh Ministers decides that it should be included.

Tick the box if you wish to claim confidentiality for your application

Please treat the information in my application as confidential. ☐

Tick the box if you wish to claim national security for your application

(Note: All applications relating to sealed sources are automatically subject to national security restrictions – only tick the box if there is some other reason for claiming national security.)

I believe that including my information in the public register would not be in the interests of national security. ☐

6 Declaration

If you knowingly or carelessly make a statement that is false or misleading to help you get an environmental permit (for yourself or anyone else), you may be committing an offence under the Environmental Permitting (England and Wales) Regulations 2010 and may be liable to prosecution.

A relevant person should make the declaration. **If you are transferring all or part of your permit, both you and the person receiving the permit must make the declaration.**

I declare the information in this application is true to the best of my knowledge and belief. I understand this application may be refused or approval withdrawn if I give false or incomplete information.

Tick this box to confirm that you understand and agree with the declaration above ☑

I confirm that my standard facility will fully meet the rules that I have applied for. (This only applies if the application is for a standard rules permit.) ☐

Name

Title (Mr, Mrs and so on) Dr

First name Gene

Last name Wilson

Position in organisation Director of Corporate Stewardship

Today's date (DD/MM/YYYY) 24/07/2015

For transfers only – declaration for person receiving the permit

I declare the information in this application to transfer an environmental permit to me is true to the best of my knowledge and belief. I understand this application may be refused or approval withdrawn if I give false or incomplete information.

If you deliberately make a statement that is false or misleading in order to obtain approval you may be liable to prosecution.

Tick this box to confirm that you understand and agree with the declaration above ☐

Name

Title (Mr, Mrs and so on)

First name

Last name

Position in organisation

Today's date (DD/MM/YYYY)

7 Application checklist

You must fill in this section.

Tell us what you have sent with this application.

The correct application fee under our charging scheme ☐ (Tick the box to say you have included the fee – only applicable if you have completed the table in question 2d)

List all the documents you have included. If necessary, continue on a separate sheet and tell us the document reference you have given it below.

Continuation sheet reference

Question reference	Document title	Document reference
RSR-A 9c, 10b, 12	Environmental Safety Case: Disposal	ESC
RSR-B 2c,3a,3b,4a,4b,5a,5b,6a,6b,8a	of Low Activity Low Level Radioactive	
	Waste at East Northants Resource	
	Management Facility. July 2015.	
	ENE 154/001	
	Eden Nuclear and Environment	

8 How to contact us

If you need help filling in this form, please contact the person who sent it to you or contact us as shown below.

Phone: 01142 800 678 or 01142 800 682
Email: RSR.Rotherham2.NE@environment-agency.gov.uk
Website: www.environment-agency.gov.uk

If you are happy with our service, please tell us. It helps us to identify good practice and encourages our staff. If you're not happy with our service, please tell us how we can improve it.

9 Where to send your application

Please send all parts of your filled-in application form and supporting documents to:

Environment Agency
PO Box 4404
Sheffield
S9 9DA
If your application **does not relate to sealed radioactive sources** you may email it to:
RSR.Rotherham2.NE@environment-agency.gov.uk

Feedback

(You don't have to answer this part of the form, but it will help us improve our forms if you do.)

We want to make our forms easy to fill in and our guidance notes easy to understand. Please use the space below to give us any comments you may have about this form or the guidance notes that came with it.

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For Environment Agency use only

Date received (DD/MM/YYYY)

Our reference number

Payment received?

No ☐

Yes ☐ Amount received

£

Do you want to remove the 'Restricted - Regulatory' text from the header and footer?

Yes ☐

No ☐

Appendix L. Monitoring results

L.1. Groundwater

1188. Groundwater samples were collected after an appropriate volume of water had been purged using the waterra tubing installed in the boreholes or a clean sampling bailer. A sample was then collected and placed straight into a 1 litre sampling bottle. This was then placed in a coolbox until it was transferred into packaging to be sent off to Public Health England (PHE, formally HPA) within sample stability times.

1189. In the following tables, “<” indicates a result is less than or equal to a test methods Limit of Detection (LOD) for that parameter at the time of analysis.

Table 188 Analysis of radioactivity in groundwater samples from location K02a

Location Id	Component	Units	22/02/2012	22/10/2012	08/05/2013	04/12/2013	Average	Max
K02a	Total alpha	Bq/g	<0.001	<0.0001	<0.001	0.00002	N/A	0.00002
K02a	Total beta	Bq/g	<0.001	<0.001	0.001	<0.00003	N/A	0.001
K02a	Total gamma	Bq/L	<1	<1	<1.0	2.01	N/A	2.01
K02a	Tritium Liquids	Bq/L	<2	<5	<5	<4.2	N/A	<LOD
K02a	Total Actinium-228	Bq/g	<0.008	N/A	N/A	<0.001	N/A	<LOD
K02a	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	<LOD
K02a	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	<LOD
K02a	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	<LOD
K02a	Total Potassium-40	Bq/g	<0.03	N/A	N/A	<0.0033	N/A	<LOD
K02a	Total Lead-210	Bq/g	<0.01	N/A	N/A	<0.0023	N/A	<LOD
K02a	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	<LOD
K02a	Total Lead-214	Bq/g	<0.004	N/A	N/A	0.002	N/A	0.002
K02a	Total Radium-224	Bq/g	<0.02	N/A	N/A	<0.0017	N/A	<LOD
K02a	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0037	N/A	<LOD
K02a	Total Thorium-234	Bq/g	<0.01	N/A	N/A	<0.0028	N/A	<LOD
K02a	Total Uranium-235	Bq/g	<0.004	N/A	N/A	<0.001	N/A	<LOD

Table 189 Analysis of radioactivity in groundwater samples from location K03

Location Id	Component	Units	22/02/2012	22/10/2012	08/05/2013	Average	Max
K03	Total alpha	Bq/g	<0.001	<0.00012	<0.001	N/A	N/A
K03	Total beta	Bq/g	<0.001	<0.001	0.003	N/A	N/A
K03	Total gamma	Bq/L	<2	<1	<1.0	N/A	N/A
K03	Tritium Liquids	Bq/L	<2	<5	<5	N/A	N/A
K03	Total Actinium-228	Bq/g	<0.009	N/A	N/A	N/A	N/A
K03	Total Americium-241	Bq/g	<0.001	N/A	N/A	N/A	N/A
K03	Total Cobalt-60	Bq/g	<0.003	N/A	N/A	N/A	N/A
K03	Total Caesium-137	Bq/g	<0.002	<0.001	N/A	N/A	N/A
K03	Total Potassium-40	Bq/g	<0.04	N/A	N/A	N/A	N/A
K03	Total Lead-210	Bq/g	<0.01	N/A	N/A	N/A	N/A
K03	Total Lead-212	Bq/g	<0.001	N/A	N/A	N/A	N/A
K03	Total Lead-214	Bq/g	<0.004	N/A	N/A	N/A	N/A
K03	Total Radium-224	Bq/g	<0.02	N/A	N/A	N/A	N/A
K03	Total Radium-226	Bq/g	<0.02	N/A	N/A	N/A	N/A
K03	Total Thorium-234	Bq/g	<0.01	N/A	N/A	N/A	N/A
K03	Total Uranium-235	Bq/g	<0.004	N/A	N/A	N/A	N/A

Table 190 Analysis of radioactivity in groundwater samples from location K03a

Location Id	Component	Units	31/12/2013	Average	Max
K03a	Total alpha	Bq/g	0.00007	N/A	0.00007
K03a	Total beta	Bq/g	0.00004	N/A	0.00004
K03a	Total gamma	Bq/L	<1.0	N/A	N/A
K03a	Tritium Liquids	Bq/L	<4.1	N/A	N/A
K03a	Total Actinium-228	Bq/g	<0.001	N/A	N/A
K03a	Total Americium-241	Bq/g	<0.001	N/A	N/A
K03a	Total Cobalt-60	Bq/g	<0.001	N/A	N/A
K03a	Total Caesium-137	Bq/g	<0.001	N/A	N/A
K03a	Total Potassium-40	Bq/g	<0.003	N/A	N/A
K03a	Total Lead-210	Bq/g	<0.0022	N/A	N/A
K03a	Total Lead-212	Bq/g	<0.001	N/A	N/A
K03a	Total Lead-214	Bq/g	<0.001	N/A	N/A
K03a	Total Radium-224	Bq/g	<0.0035	N/A	N/A
K03a	Total Radium-226	Bq/g	<0.0034	N/A	N/A
K03a	Total Thorium-234	Bq/g	<0.0027	N/A	N/A
K03a	Total Uranium-235	Bq/g	<0.001	N/A	N/A

Table 191 Analysis of radioactivity in groundwater samples from location K04

Location Id	Component	Units	22/02/2012	22/10/2012	08/05/2013	04/12/2013	Average	Max
K04	Total alpha	Bq/g	<0.001	<0.0001	<0.001	0.00003	N/A	0.00003
K04	Total beta	Bq/g	<0.001	<0.001	<0.001	0.00004	N/A	0.00004
K04	Total gamma	Bq/L	<1	<1	<1.0	<1.0	N/A	N/A
K04	Tritium Liquids	Bq/L	2	<5	<5	<4.2	N/A	2
K04	Total Actinium-228	Bq/g	<0.009	N/A	N/A	<0.001	N/A	N/A
K04	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	N/A
K04	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K04	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	N/A
K04	Total Potassium-40	Bq/g	<0.04	N/A	N/A	<0.0034	N/A	N/A
K04	Total Lead-210	Bq/g	<0.01	N/A	N/A	<0.0021	N/A	N/A
K04	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K04	Total Lead-214	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A
K04	Total Radium-224	Bq/g	<0.02	N/A	N/A	<0.0034	N/A	N/A
K04	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0031	N/A	N/A
K04	Total Thorium-234	Bq/g	<0.02	N/A	N/A	<0.0023	N/A	N/A
K04	Total Uranium-235	Bq/g	<0.005	N/A	N/A	<0.001	N/A	N/A

Table 192 Analysis of radioactivity in groundwater samples from location K05

Location Id	Component	Units	22/02/2012	22/10/2012	17/04/2013	03/12/2013	Average	Max
K05	Total alpha	Bq/g	<0.001	<0.0001	<0.0001	0.00005	N/A	0.00005
K05	Total beta	Bq/g	<0.001	<0.001	<0.001	0.00006	N/A	0.00006
K05	Total gamma	Bq/L	<1	<1	<1.0	2.11	N/A	N/A
K05	Tritium Liquids	Bq/L	<2	<5	<5	<2.3	N/A	N/A
K05	Total Actinium-228	Bq/g	<0.007	N/A	N/A	<0.001	N/A	N/A
K05	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	N/A
K05	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K05	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	N/A
K05	Total Potassium-40	Bq/g	<0.06	N/A	N/A	<0.0055	N/A	N/A
K05	Total Lead-210	Bq/g	<0.009	N/A	N/A	<0.0031	N/A	N/A
K05	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K05	Total Lead-214	Bq/g	<0.003	N/A	N/A	0.0021	N/A	0.0021
K05	Total Radium-224	Bq/g	<0.03	N/A	N/A	<0.0025	N/A	N/A
K05	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0047	N/A	N/A
K05	Total Thorium-234	Bq/g	<0.01	N/A	N/A	<0.0044	N/A	N/A
K05	Total Uranium-235	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A

Table 193 Analysis of radioactivity in groundwater samples from location K06

Location Id	Component	Units	22/02/2012	Average	Max
K06	Total alpha	Bq/g	<0.001	N/A	N/A
K06	Total beta	Bq/g	<0.001	N/A	N/A
K06	Total gamma	Bq/L	<2	N/A	N/A
K06	Tritium Liquids	Bq/L	2	N/A	2
K06	Total Actinium-228	Bq/g	<0.009	N/A	N/A
K06	Total Americium-241	Bq/g	<0.001	N/A	N/A
K06	Total Cobalt-60	Bq/g	<0.003	N/A	N/A
K06	Total Caesium-137	Bq/g	<0.002	N/A	N/A
K06	Total Potassium-40	Bq/g	<0.04	N/A	N/A
K06	Total Lead-210	Bq/g	<0.01	N/A	N/A
K06	Total Lead-212	Bq/g	<0.002	N/A	N/A
K06	Total Lead-214	Bq/g	<0.004	N/A	N/A
K06	Total Radium-224	Bq/g	<0.02	N/A	N/A
K06	Total Radium-226	Bq/g	<0.02	N/A	N/A
K06	Total Thorium-234	Bq/g	<0.01	N/A	N/A
K06	Total Uranium-235	Bq/g	<0.004	N/A	N/A

Table 194 Analysis of radioactivity in groundwater samples from location K06a

Location Id	Component	Units	22/10/2012	17/04/2013	04/12/2013	Average	Max
K06a	Total alpha	Bq/g	<0.0001	<0.0007	0.00002	N/A	0.00002
K06a	Total beta	Bq/g	<0.001	0.0026	0.00004	N/A	0.00004
K06a	Total gamma	Bq/L	<1	<1.0	1.06	N/A	1.06
K06a	Tritium Liquids	Bq/L	<5	<5	<4.2	N/A	N/A
K06a	Total Actinium-228	Bq/g	N/A	N/A	<0.001	N/A	N/A
K06a	Total Americium-241	Bq/g	N/A	N/A	<0.001	N/A	N/A
K06a	Total Cobalt-60	Bq/g	N/A	N/A	<0.001	N/A	N/A
K06a	Total Caesium-137	Bq/g	<0.001	N/A	<0.001	N/A	N/A
K06a	Total Potassium-40	Bq/g	N/A	N/A	<0.0033	N/A	N/A
K06a	Total Lead-210	Bq/g	N/A	N/A	<0.0023	N/A	N/A
K06a	Total Lead-212	Bq/g	N/A	N/A	<0.002	N/A	N/A
K06a	Total Lead-214	Bq/g	N/A	N/A	0.0011	N/A	0.0011
K06a	Total Radium-224	Bq/g	N/A	N/A	<0.0038	N/A	N/A
K06a	Total Radium-226	Bq/g	N/A	N/A	<0.0037	N/A	N/A
K06a	Total Thorium-234	Bq/g	N/A	N/A	<0.0029	N/A	N/A
K06a	Total Uranium-235	Bq/g	N/A	N/A	<0.001	N/A	N/A

Table 195 Analysis of radioactivity in groundwater samples from location K07

Location Id	Component	Units	22/02/2012	22/10/2012	17/04/2013	03/12/2013	Average	Max
K07	Total alpha	Bq/g	<0.001	0.00018	<0.0007	0.00003	0.0006	0.00018
K07	Total beta	Bq/g	<0.001	<0.001	<0.0013	0.00006	N/A	0.00006
K07	Total gamma	Bq/L	<1	<0.001	<1.0	<1.0	N/A	N/A
K07	Tritium Liquids	Bq/L	<3	<5	<5	<2.3	N/A	N/A
K07	Total Actinium-228	Bq/g	<0.008	N/A	N/A	<0.001	N/A	N/A
K07	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	N/A
K07	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K07	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	N/A
K07	Total Potassium-40	Bq/g	<0.03	N/A	N/A	<0.0032	N/A	N/A
K07	Total Lead-210	Bq/g	<0.01	N/A	N/A	<0.0021	N/A	N/A
K07	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K07	Total Lead-214	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A
K07	Total Radium-224	Bq/g	<0.02	N/A	N/A	<0.0034	N/A	N/A
K07	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0035	N/A	N/A
K07	Total Thorium-234	Bq/g	<0.02	N/A	N/A	<0.0031	N/A	N/A
K07	Total Uranium-235	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A

Table 196 Analysis of radioactivity in groundwater samples from location K08

Location Id	Component	Units	22/02/2012	22/10/2012	17/04/2013	03/12/2013	Average	Max
K08	Total alpha	Bq/g	<0.001	<0.0001	<0.0007	0.00004	N/A	0.00004
K08	Total beta	Bq/g	<0.001	<0.001	<0.0012	0.0001	N/A	0.0001
K08	Total gamma	Bq/L	<1	<1	<1.0	<1.0	N/A	N/A
K08	Tritium Liquids	Bq/L	<2	<5	<5	<2.3	N/A	N/A
K08	Total Actinium-228	Bq/g	<0.006	N/A	N/A	<0.001	N/A	N/A
K08	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	N/A
K08	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K08	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	N/A
K08	Total Potassium-40	Bq/g	<0.06	N/A	N/A	<0.0033	N/A	N/A
K08	Total Lead-210	Bq/g	<0.009	N/A	N/A	<0.002	N/A	N/A
K08	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K08	Total Lead-214	Bq/g	<0.003	N/A	N/A	<0.001	N/A	N/A
K08	Total Radium-224	Bq/g	<0.02	N/A	N/A	<0.0034	N/A	N/A
K08	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0033	N/A	N/A
K08	Total Thorium-234	Bq/g	<0.01	N/A	N/A	<0.0025	N/A	N/A
K08	Total Uranium-235	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A

Table 197 Analysis of radioactivity in groundwater samples from location K09

Location Id	Component	Units	22/02/2012	22/10/2012	08/05/2013	04/12/2013	Average	Max
K09	Total alpha	Bq/g	<0.001	<0.0001	<0.001	<0.00002	N/A	N/A
K09	Total beta	Bq/g	<0.001	<0.001	0.004	0.00008	0.00204	0.004
K09	Total gamma	Bq/L	<1	<1	<1.0	<1.0	N/A	N/A
K09	Tritium Liquids	Bq/L	<2	<5	<5	<4.2	N/A	N/A
K09	Total Actinium-228	Bq/g	<0.006	N/A	N/A	<0.001	N/A	N/A
K09	Total Americium-241	Bq/g	<0.001	N/A	N/A	<0.001	N/A	N/A
K09	Total Cobalt-60	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K09	Total Caesium-137	Bq/g	<0.001	<0.001	N/A	<0.001	N/A	N/A
K09	Total Potassium-40	Bq/g	<0.05	N/A	N/A	<0.0055	N/A	N/A
K09	Total Lead-210	Bq/g	<0.008	N/A	N/A	<0.0031	N/A	N/A
K09	Total Lead-212	Bq/g	<0.002	N/A	N/A	<0.001	N/A	N/A
K09	Total Lead-214	Bq/g	<0.003	N/A	N/A	<0.001	N/A	N/A
K09	Total Radium-224	Bq/g	<0.02	N/A	N/A	<0.0043	N/A	N/A
K09	Total Radium-226	Bq/g	<0.02	N/A	N/A	<0.0047	N/A	N/A
K09	Total Thorium-234	Bq/g	<0.01	N/A	N/A	<0.0044	N/A	N/A
K09	Total Uranium-235	Bq/g	<0.004	N/A	N/A	<0.001	N/A	N/A

Table 198 Analysis of radioactivity in groundwater samples from location K14a

Location Id	Component	Units	22/10/2012	08/05/2013	04/12/2013	Average	Max
K14a	Total alpha	Bq/g	0.00014	<0.001	<0.00002	N/A	0.00014
K14a	Total beta	Bq/g	<0.001	<0.001	0.00007	N/A	0.00007
K14a	Total gamma	Bq/L	<1	<1.0	<1.0	N/A	N/A
K14a	Tritium Liquids	Bq/L	<5	<5	<4.1	N/A	N/A
K14a	Total Actinium-228	Bq/g	N/A	N/A	<0.001	N/A	N/A
K14a	Total Americium-241	Bq/g	N/A	N/A	<0.001	N/A	N/A
K14a	Total Cobalt-60	Bq/g	N/A	N/A	<0.001	N/A	N/A
K14a	Total Caesium-137	Bq/g	<0.001	N/A	<0.001	N/A	N/A
K14a	Total Potassium-40	Bq/g	N/A	N/A	<0.0032	N/A	N/A
K14a	Total Lead-210	Bq/g	N/A	N/A	<0.0021	N/A	N/A
K14a	Total Lead-212	Bq/g	N/A	N/A	<0.001	N/A	N/A
K14a	Total Lead-214	Bq/g	N/A	N/A	<0.001	N/A	N/A
K14a	Total Radium-224	Bq/g	N/A	N/A	<0.0035	N/A	N/A
K14a	Total Radium-226	Bq/g	N/A	N/A	<0.0035	N/A	N/A
K14a	Total Thorium-234	Bq/g	N/A	N/A	<0.0031	N/A	N/A
K14a	Total Uranium-235	Bq/g	N/A	N/A	<0.001	N/A	N/A

L.2. Dust sampling

1190. All dust samples were collected during the monthly routine monitoring. De-ionised water was used to rinse the deposited dust from the top of the dust gauge (collection Frisbee) through 227mm pipework into a 5 litre HDPE collection bottle. The entire sample is filtered at the on-site laboratory and the dried filter sent off for analysis at PHE.

Table 199 Analysis of radioactivity in dust samples (Bq/filter) from location KCDD01

Location Id	Component in dust	22/02/2012	22/05/2012	23/08/2012	15/11/2012	15/02/2013	15/05/2013	15/08/2013	15/10/2013	Ave.	Max
KCDD01	Total Ac-228	<0.1	<0.1	<0.1	N/A	N/A	N/A	<0.11	<0.11	N/A	<0.11
KCDD01	Total alpha	<0.04	0.17	<0.1	<0.1	<0.1	0.011	0.02	0.0146	0.0152	0.17
KCDD01	Total Am-241	<0.02	<0.02	<0.02	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD01	Total beta	<0.08	0.6	<0.2	<1	<1.0	0.103	0.1	0.101	0.10133	0.6
KCDD01	Total Co-60	<0.05	0.18	<0.05	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD01	Total Cs-137	<0.03	<0.05	<0.03	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD01	Total gamma	N/A	N/A	N/A	<0.1	0.5	<0.1	<0.1	<0.1	0.5	0.5
KCDD01	Total K-40	<0.8	<0.03	<0.8	N/A	N/A	N/A	<0.45	<0.5	N/A	<0.45
KCDD01	Total Pb-210	<0.2	<0.8	<0.2	N/A	N/A	N/A	<0.26	<0.3	N/A	<0.26
KCDD01	Total Pb-212	<0.02	<0.2	<0.03	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD01	Total Pb-214	<0.06	<0.03	<0.07	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD01	Total Ra-224	<0.3	<0.07	N/A	N/A	N/A	N/A	<0.46	<0.5	N/A	<0.46
KCDD01	Total Ra-226	<0.3	<0.3	<0.3	N/A	N/A	N/A	<0.045	<0.5	N/A	N/A
KCDD01	Total Th-234	<0.2	<0.2	<0.2	N/A	N/A	N/A	<1.4	<1.4	N/A	<1.4
KCDD01	Total U-235	<0.06	<0.07	<0.07	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1

Table 200 Analysis of radioactivity in dust samples (Bq/filter) from location KCDD02

Location Id	Component in dust	22/02/2012	22/05/2012	23/08/2013	15/11/2012	15/02/2013	15/05/2013	15/08/2013	15/10/2013	Ave.	Max
KCDD02	Total Ac-228	<0.1	<0.1	<0.1	N/A	N/A	N/A	<0.11	<0.1	N/A	<0.11
KCDD02	Total alpha	0.05	0.1	0.08	<0.1	<0.1	0.006	0.01	0.0104	0.0088	0.01
KCDD02	Total Am-241	<0.02	<0.02	<0.02	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD02	Total beta	<0.08	0.46	<0.1	<1	<1.0	0.088	0.2	0.16	0.149333	0.16
KCDD02	Total Co-60	<0.04	<0.2	<0.04	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD02	Total Cs-137	<0.02	<0.04	<0.02	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD02	Total gamma	N/A	N/A	N/A	<0.1	0.39	<0.1	0.7	0.65	0.58	0.7
KCDD02	Total K-40	<0.7	<0.02	<0.7	N/A	N/A	N/A	0.65	0.7	N/A	0.65
KCDD02	Total Pb-210	<0.2	<0.7	<0.2	N/A	N/A	N/A	<0.29	<0.3	N/A	<0.29
KCDD02	Total Pb-212	0.04	<0.2	<0.03	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD02	Total Pb-214	<0.06	<0.03	<0.06	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1
KCDD02	Total Ra-224	<0.4	<0.06	N/A	N/A	N/A	N/A	<0.5	<1	N/A	<0.5
KCDD02	Total Ra-226	<0.3	<0.3	<0.3	N/A	N/A	N/A	<0.49	<0.5	N/A	N/A
KCDD02	Total Th-234	<0.2	<0.2	<0.2	N/A	N/A	N/A	<1.6	<1.6	N/A	<1.6
KCDD02	Total U-235	<0.07	<0.07	<0.07	N/A	N/A	N/A	<0.1	<0.1	N/A	<0.1

L.3. Surface soil samples

1191. All soil samples were collected using a Soil Sampler Pro (a cross-sectional soil sampler) to a maximum depth of 10 centimetres at all four locations specified in Permit CD8503. The samples were stored in labelled plastic tubs which were then securely sealed and boxed to be collected and delivered to PHE within the specified stability times.

Table 201 Analysis of radioactivity in soil samples from location KCSL_01

Location Id	Component in soil	Units	21/06/2012	25/06/2013	Ave.	Max
KCSL_01	Total alpha	Bq/g	0.5	0.286	0.393	0.286
KCSL_01	Total beta	Bq/g	0.9	1.52	1.52	1.52
KCSL_01	Total gamma	Bq/g	0.003	0.77	0.77	0.77

Table 202 Analysis of radioactivity in soil samples from location KCSL_02

Location Id	Component in soil	Units	21/06/2012	25/06/2013	Ave.	Max
KCSL_02	Total alpha	Bq/g	0.3	0.2	0.2	0.2
KCSL_02	Total beta	Bq/g	0.9	0.94	0.94	0.94
KCSL_02	Total gamma	Bq/g	0.001	0.304	0.304	0.304

Table 203 Analysis of radioactivity in soil samples from location KCSL_03

Location Id	Component in soil	Units	21/06/2012	25/06/2013	Ave.	Max
KCSL_03	Total alpha	Bq/g	0.3	0.21	0.21	0.21
KCSL_03	Total beta	Bq/g	0.9	1.18	1.18	1.18
KCSL_03	Total gamma	Bq/g	0.003	0.558	0.558	0.558

Table 204 Analysis of radioactivity in soil samples from location KCSL_04

Location Id	Component in soil	Units	21/06/2012	25/06/2013	Ave.	Max
KCSL_04	Total alpha	Bq/g	0.3	0.252	0.252	0.3
KCSL_04	Total beta	Bq/g	1.5	1.39	1.39	1.5
KCSL_04	Total gamma	Bq/g	0.002	0.695	0.695	0.695

L.4. Site perimeter dose rate

1192. The site perimeter dose rate check is carried out by Augean's Environmental Monitoring Technician. In accordance with the Monitoring Action Plan for perimeter dose rate monitoring, the perimeter dose rate analysis was carried out using a fully calibrated AT1121 X Ray and Gamma Radiation Dosimeter. An average reading over a 10 minute period at 1 metre above the locations specified in Permit CD8503 is recorded at each location. Weather conditions including barometric pressure, temperature, wind speed and direction and ground conditions are also recorded.

Table 205 Site perimeter total gamma dose rate (mSv h^{-1}) measurements at the site boundary location

Location Id	23/02/2012	30/05/2012	31/07/2012	30/10/2012	08/02/2013	03/05/2013	29/08/2013	25/11/2013	Average	Max
K03	0.000094	0.000096	0.000081	0.000093	0.000091	0.000088	0.000086	0.000097	9.08E-05	0.000097
KCSOIL02	0.000095	0.000087	0.000081	0.000096	0.000092	0.000097	0.000089	0.000084	9.01E-05	0.000097
KCSOIL01	0.000096	0.000099	0.000082	0.000093	0.000101	0.0001	0.000092	0.000087	9.38E-05	0.0001
K13	0.000095	0.000088	0.000077	0.000096	0.0001	0.000093	0.00008	0.000086	8.94E-05	0.0001
KCSOIL03	0.000084	0.000071	0.000076	0.000077	0.000099	0.000091	0.000078	0.000077	8.16E-05	0.000099

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Addendum to Environmental Safety Case: Disposal of Low Activity Low Level Radioactive Waste at East Northants Resource Management Facility

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List of Terms and Acronyms

ALARA As low As Reasonably Achievable
AOD Above Ordnance Datum
bgl Below ground level
BGS British Geological Survey
Biosphere The part of the environment where living organisms exist, or which is capable of supporting life
BPM Best Practicable Means
Cefas Centre for Environment, Fisheries and Aquaculture Science
CLEA Contaminated Land Exposure Assessment Model
CoRWM Committee on Radioactive Waste Management
DfT Department for Transport
EA Environment Agency
EFEPs External features, Events and Processes
EH&S Environment, Health and Safety
EHS&QM Environment, Health, Safety and Quality Manager
EIA Environmental Impact Assessment
EMP Environmental Management Plan
EMS Environmental Management System
ESC Environmental Safety Case
ESS Environmental Safety Strategy
Far Field The geosphere external to the engineered features of the disposal system.
FEPs Features, Events and Processes
GDF Geological Disposal Facility
Geosphere The solid component of the earth (rock and soil etc.)
NS-GRA Guidance on the requirements for authorisation for near-surface disposal facilities on land for solid radioactive wastes
PHE Public Health England
IAEA International Atomic Energy Agency
ICRP International Commission on Radiological Protection
IPPC Integrated Pollution Prevention and Control
LA-LLW Lower Activity Low Level Waste
LLW Low Level Waste
LOAEL Lowest Observable Adverse Effects Level
MDI Mean Daily Intake
NDA Nuclear Decommissioning Authority
NE Normal Evolution (a description of the reference case for the evolution of the disposal system)
Near Field The wastes, waste packages and engineered barriers within the disposal system.
NHB Non-Human Biota
NOAEL No Observable Adverse Effects Level
PA Performance Assessment
PEG Potentially Exposed Group
QMS Quality Management System
RWMD Radioactive Waste Management Directorate (part of the NDA)
SEPA Scottish Environment Protection Agency
SQEP Suitably Qualified and Experienced Personnel
UKCP UK Climate Projections
VLLW Very Low Level Waste
WAC Waste Acceptance Criteria
WASSC Waste Safety Standards Committee (part of the IAEA Safety Standards Commission and Committees)

Units and Prefixes

SI units of radiation and radioactivity

Quantity	SI unit and abbreviation
Absorbed dose	Gray (Gy)
Effective Dose	Sievert (Sv)
Radioactivity	Becquerel (Bq)

Multiples and sub-multiples of SI units

Factor	Prefix and abbreviation	Factor	Prefix and abbreviation
10^{18}	exa (E)	10^{-3}	milli (m)
10^{15}	peta (P)	10^{-6}	micro (μ)
10^{12}	tera (T)	10^{-9}	nano (n)
10^{9}	giga (G)	10^{-12}	pico (p)
10^{6}	mega (M)	10^{-15}	femto (f)
10^{3}	kilo (k)	10^{-18}	atto (a)

Contents

Tables and Figures

1 Introduction

1. This document is an Addendum to the Environmental Safety Case (ESC) (Eden Nuclear and Environment Ltd, 2015) presented to the Environment Agency (EA) in July 2015 to support a request for a variation to Environment Agency Permit number CD8503, for receipt and disposal of low level radioactive waste at the East Northants Resource Management Facility (ENRMF), Stamford Road, King's Cliffe, Northamptonshire, PE8 6XX, United Kingdom (the centre of the site lies approximately at OS Grid Reference TF 0084 0002, 52.5887^{o} N 0.5130^{o} W). The application reference for the permit variation is EPR/FB3598DD/V001.

2. Augean South Limited (Augean) is the operator of the ENRMF which comprises a hazardous waste treatment facility at which materials are recycled, recovered and hazardous properties reduced and a landfill at which a range of hazardous wastes and low activity radioactive waste is disposed. The Environment Agency Permit number CD8503 covers disposal in cells 4B, 5A and 5B of the landfill. On 11th July 2013, the Secretary of State (The East Northamptonshire Resource Management Facility Order, 2013) approved the extension of the ENRMF to include an additional void of 1.2 10^{6} m^{3} (1.2 million cubic metres) over an area of approximately 11 ha (hectares) and an increase in the annual capacity of the treatment facility to 150,000 t y^{-1} (tonnes per year). The order permits disposal of 150,000 t y^{-1} of hazardous and low level radioactive waste (LLW) direct to landfill. It states that radioactive waste, to a maximum specific activity of 200 Bq g^{-1} (Becquerel per gramme) may be disposed in cells 4B, 5A and 5B and Phases 6 to 11. LLW input to the site is capped at 448,000 t (tonnes). The permit variation would extend the Environmental Permit for the LLW disposal area to include Phases 6 to 11 as well as cells 4B, 5A and 5B. A revised permit for the disposal of hazardous wastes including phases 6 to 11 was issued on 5th October 2015 (EPR/TP3430GW).

3. The EA requested further information to support the application in correspondence dated 8th October 2015 and 20th November 2015. These requests are reproduced in Annex A. The purpose of this document is to provide the requested information. In order to avoid confusion between the ESC (Eden Nuclear and Environment Ltd, 2015) and the Addendum, the former used Appendices and this document uses Annexes.

Document structure

4. Further information was requested concerning the half-life of radionuclides included in the "Other radionuclides" group, in an email dated 8th October 2015 (Annex A). A response was sent to the EA on 20th November 2015 and this is reproduced here in Annex B. This justifies why the permit should considering only those "Other radionuclides" that have a half-life greater than 1 year.

5. In the EA letter dated 20th November 2015 (Annex A) 11 further requests were provided and these are considered in Section 2 and Annex C. Transcription errors were identified in two ESC tables during this work and corrected versions are provided in Section 3. The ESC included a proposed schedule of limits for the revised permit and an amended version of this table, taking account of EA comments and our responses, is presented and discussed in Section 4.

2 Responses to letter of 20th November 2015

2.1 Large scale intrusion scenario

1) A number of smaller intrusion scenarios have been considered within the Environmental Safety Case provided with the above application.

However, we note that a scenario for a larger intrusion involving substantial amounts of waste being displaced, for example: to make way for a major road, has been excluded from the assessment. This query originates from the fact that a large intrusion was considered in the 2009 ESC and that it was limiting for some radionuclides.

We ask Augean to provide justification as to why a large intrusion scenario has not been considered in the Environmental Safety Case;

6. The 2009 ESC does not consider a large intrusion scenario. It was based on the SNIFFER methodology and considered doses to workers during a small scale intrusion, and redevelopment for housing. The radiological capacity was limited by doses to workers for a small scale intrusion after 60 years for 23 radionuclides, and the dose to residents following intrusion at 60 years limited the radiological capacity of 2 radionuclides. The 2009 ESC results for intrusion have been reproduced using the SNIFFER default parameters combined with any alternative values specified in the 2009 ESC (Section A.3 of Annex A, and Appendix B).

7. A site re-engineering/remediation scenario is 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 the ENRMF, 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. No results for this scenario are presented in 2009 ESC. Hence this scenario was not considered in the 2015 ESC (Section E5, paragraph 767).

8. The ESC considers the period of authorisation (see Figure 15 of the 2015 ESC) and this is assumed to include a period of active management lasting for 60 years after capping. There will be a period following this active management period during which records and knowledge will exist about the site, and during which any development of the site will be undertaken with the knowledge that the site was a hazardous and radioactive waste disposal facility. This is a passive form of control that will apply for a period of time, until records become lost or are not referred to, after which intrusion into the site without knowledge of what is there would be possible.

9. The site will be restored to a mix of woodland, scrub and species rich neutral grassland, in accordance with the planning permission/Development Order. Future development of the site would need planning permission and this would result in associated searches of the records (for example those held by the EA, DECC and the local authority, planning records, the development consent order and local maps). These records will inform the future

development of the site. Any work would therefore be carried out with the knowledge that there is radioactive material on the site and it can be assumed that appropriate precautions against exposure would be adopted and that contaminated material would be disposed of safely. Intrusion with knowledge of the site and its contents is not relevant to the ESC, and would be regulated by the HSE under the Ionising Radiations Regulations. Small exploratory intrusions might occur but large projects are unlikely to get planning permission during this passive management phase.

10. The ESC intrusion scenarios assume that the workers and residents have no knowledge of any contamination and very cautiously considered that controls limiting development or intrusions will be ineffective after 60 years for small scale intrusions such as a Borehole, but would remain effective for up to 150 years for the Resident scenario and up to 200 years for the Smallholder scenario. Whilst it is not possible to predict when passive controls will fail to identify the historic use of the site, 60 years is clearly too early.

11. The resident and smallholder intrusion scenarios considered in the 2015 ESC involve substantial excavation of material, including wastes. However, a large intrusion relating to developments such as motorways, a major road or railways was not explicitly considered as it was not considered to be credible for a site containing hazardous and radioactive wastes such as the ENRMF. This type of development would require planning permission. If planning permission was granted while records were available then the excavation would be performed in full knowledge of the nature of the site and this is not relevant to the ESC. At later times when records may not be available it is possible that development of a road or railway through the site could be considered. However, thorough geotechnical investigations would precede a major development of this type and they would reveal artificial made ground; based on current practice, this finding would not be ignored and further investigations would follow. The presence of wastes and potentially hazardous materials would also be discovered through trial pits and other small intrusions. It needs to be emphasised that this is a hazardous waste landfill, containing identifiable wastes such as asbestos which do not degrade in soil. Developers of brownfield sites, e.g. sites containing artificial made ground, are aware that contamination is possibly an issue and take this into account when planning construction activities e.g. see site investigation (AGS, 2013) and construction industry (CIRIA, 2014) guidance on asbestos in soil and made ground. The GRA considers current technologies (paragraph 6.3.48) and hence it is unreasonable to assume that future groundworks would pay less attention to potential contamination of brownfield sites and made ground than currently. With these arguments in mind a large intrusion such as that associated with a road cutting carried out with no knowledge of site contamination was not considered to be credible and it has not been considered in the ESC.

12. The EA has not yet revoked a disposal permit for a near surface radioactive waste disposal site. There is guidance from SEPA on principles and expectations on the revocation of authorisations for radioactive substances (SEPA, 2014) and we are aware that the EA is looking at development controls that can be put in place through the planning regime: these could be used to prevent development of landfills containing radioactive waste for a specified time or until the EA/SEPA advise that the restriction can be lifted. Such restrictions would give further legal protection against intrusion at early times (60 to 150 years) and are a reliable and legally binding form of passive control.

2.2 Time-dependent trends in groundwater concentrations

2) We note that Appendix F in the Environmental Safety Case includes information relating to interim concentrations for radionuclides in groundwater below the landfill. However, this information is partial and it is unclear to us whether these concentrations have been used in the assessment or are purely indicative.

Interim concentrations of radionuclides in groundwater that have been assumed for the "well constructed at the site boundary" scenario do not appear to have been explicitly provided in the Environmental Safety Case.

In order to provide further clarification and confidence in the assessment we ask Augean to provide us with time trend graphs showing concentrations per unit disposal at close to the well abstraction point for the following radionuclides: H-3, C-14, Cl-36, Tc-99, Sn-126, I-129, Ra-226**, Ra-226, U-234, U-235, U-238, Np-237, Pu-240 and Pu-242.***

13. Interim concentrations of radionuclides in groundwater for the "well constructed at the site boundary" scenario were not explicitly provided in the 2015 ESC. The concentrations in groundwater extracted from a well constructed at the site boundary that correspond to the doses presented in the ESC are presented in Annex C of this addendum report for the radionuclides listed above. The concentrations of Cl-36, Ra-226 and Pb-210 in groundwater that were provided in Appendix F of the 2015 ESC were used for model checking and incorporated a higher waste porosity value, see Section 2.7.

2.3 Infant doses from Cl-36 in groundwater

3) The Environmental Safety Case does not consider infant and child doses but rather provides an analysis that additional calculations are not necessary as previous assessments from the 2009 Environmental Safety Case show that doses to children and infants (plus the foetus/embryo) will be lower than adult doses for the majority of radionuclides (paras 430-431). We note, however, from the 2009 Environmental Safety Case that, in the case of consumption of foodstuffs contaminated with Cl-36, the child/infant doses in the were higher than adult doses by a factor of 2.5 (2009 application document Annex B table C5 – note that child/infant doses for all other radionuclides are lower than corresponding adult doses).

We have considered the results for Cl-36 for the smallholder scenario in the (2015) Environmental Safety Case submitted in support of the variation application in light of the above:

The dose from the maximum inventory of Cl-36 to a smallholder at 200 y is 0.08 mSv/y (Table 18). A factor of 2.5 increase in this dose would be well below the dose guidance level. Similarly, for the leachate treatment scenario, peak impacts to the farming family from Cl-36 are 0.005 uSv/y. However, the limiting capacity for Cl-36 is associated with the well scenario – in these calculations the irrigation pathway (which includes consumption of foodstuffs contaminated with contaminated groundwater) contributes about 80% of dose compared with 20% for the water drinking pathway for Cl-36 (Tables 78, 79).

We take view that the Cl-36 radiological capacity may not be conservative for infants and children. We therefore ask Augean to formally assess impacts to child/infant receptors for Cl-36 in a well[1] scenario and revise the Cl-36 radiological capacity accordingly. The outcome of this assessment should be provided to us.

Note 1. Confirmed 30/11/2015 that the well scenario limited capacity and should be assessed.

14. The dose to infants from groundwater was assessed for a well located at the boundary of the site by scaling Goldsim outputs to account for age related dose coefficients and infant consumption rates. The main contributions to dose for infants and children are from the consumption of root vegetables and milk. Hence, these are at 97.5th percentile intake values and all others are at average intakes.

15. The age-related Cl-36 ingestion dose coefficients for different age groups (Table 1) and consumption rates for infants (Table 2) are presented below. The contribution from inhalation or ingestion of Cl-36 contaminated dusts/soil are both less than 1.0 x 10^{-10} μSv y^{-1} MBq^{-1} and are not therefore considered here.

Table 1 Age related dose coefficients for Cl-36 (ICRP, 1996)

Radionuclide	Ingestion dose coefficient (Sv Bq^{-1})					
	Infant	1 y	5 y	10 y	15 y	Adult
Cl-36	8.8 x 10^{-9}	6.3 x 10^{-9}	3.2 x 10^{-9}	1.9 x 10^{-9}	1.2 x 10^{-9}	9.3 x 10^{-10}

Table 2 Mean and 97.5 percentile consumption rates for infants

Pathway	Infant average	Infant 97.5th	Comment
Milk consumption (l y^{-1})	148	320	From (Smith & Jones, 2003).
Cow meat consumption (kg y^{-1})	3	10	
Sheep meat consumption (kg y^{-1})	0.8	3	
Offal consumption (kg y^{-1})	1	5.5	
Green & other domestic veg consumption (kg y^{-1})	5	15	
Root veg & potatoes consumption (kg y^{-1})	15	45	
Drinking water (m^3 y^{-1})[1]	0.26		

Note 1. A single value based on reference man is provided by (Smith & Jones, 2003).

16. This assessment produces a total dose to an infant that is a factor of 4.0 greater than the adult dose (Table 3). The dose to a 10 year old child was a factor of 1.4 greater than the adult dose and is not therefore discussed further. The relative contribution of the drinking water pathway to the total dose is 17.9% for an adult and 18.3% for an infant.

Table 3 Maximum annual doses for adults, 10 year olds and infants based on a unit inventory of 1 MBq of Cl-36 for a well at the site boundary

Radionuclide	Total dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)			
	Adult	10 y	Infant	Infant/Adult
Cl-36	1.35×10^{-5}	1.89×10^{-5}	5.41×10^{-5}	4.0

17. The radiological capacity based on the dose to an adult is 1.48 TBq, it is 0.370 TBq based on the dose to an infant.

2.4 Monthly leachate treatment

4) Doses for leachate treatment scenarios are significantly lower (factor ~6) in the final 2015 Environmental Safety Case compared with the draft. The original model assumed 2400 m^3/y based on GoldSim groundwater model output. The revised model assumes 403 m^3/y based on 28 m^3 monthly loads with an allowance of 20% for peak rainfall (para 609). We understand that the revised calculations in the Environmental Safety Case submitted with the application use a 'realistic volume of leachate' but the data used to support these calculations is unclear to us.

We ask Augean to provide justification for the reduction in values for the annual quantities of leachate that have been used in the revised model. We ask also that Augean explains the source of the 28m^3 figure that has been used for monthly loads of leachate removed from site.

18. Records indicate that the amount of leachate dispatched to the offsite leachate treatment facility in the last 33 months is irregular and can vary substantially from quarter to quarter.

19. While the Soil Treatment Facility is in operation pumped leachate will be used in the stabilisation process. The amount used on site varies depending on waste throughput at the stabilisation plant with a maximum of about 5200 m^3 per quarter in the last 18 months (email from S. Moyle; 17-12-2015).

20. The 28 m^3 per month used in the ESC was based on the volume of a tanker load, and a view that one tanker per month was a reasonable estimate of the experience at the site at the time the 2015 ESC was finalised, taking into account the fact that no leachate was sent off-site in one quarter in 2013, in one quarter in 2014, and anticipated hazardous waste permit requirements (granted October 2015). Following receipt of the EA comment we have re-evaluated the data and now take the view that the estimate of one tanker per month was based on a period of low leachate production which has not continued, but it may still be a good estimate of the future usage of the off-site treatment facility under the new hazardous waste Permit. We have now revised the estimate used in the calculations as described below.

Table 4 Leachate disposal to Avonmouth

Period	Tonnage	Comment
2013 minimum per quarter (Q2)	0	
2013 maximum per quarter (Q3)	1,021	
2013 total in 2013	2,236	
2014 minimum per quarter (Q1)	0	
2014 maximum per quarter (Q2)	302	
2014 total in 2014	566	
Annual average Q1 2013 to Q4 2014	1,401	Used for leachate treatment calculations
Annual average Q2 2014 to Q1 2015	1,344	
2015 minimum per quarter (Q1)	778	
2015 maximum per quarter (Q2)	4,132	additional leachate to Avonmouth during this period due to one off pumping campaign

21. A review of data available from Q1 2013 to Q3 2015 (email from S. Moyle; 17-12-2015) shows a recent increase in the use of the off-site facility at Avonmouth for leachate processing: a high volume was sent off-site in Q2 2015 due to a campaign aimed at adjusting leachate sump levels within the waste cells. This high level of leachate transfer will not occur again because the new hazardous waste permit (granted in October 2015) allows a greater depth of leachate to accumulate and this will provide a buffer against fluctuations in the use of leachate at the on-site Soil Treatment Facility. Hence, it is expected that future use of the off-site treatment facility will be lower than in the past.

22. The majority of leachate comes from uncapped waste cells and sequential capping will therefore reduce the inventory available to leachate. In general, only 2 cells at any one time would be uncapped (the cell being filled and the cell that has just been completed), though operational considerations may mean that full capping of a cell may be delayed. Under these circumstances only a proportion of the inventory, e.g. 2 out of a total of 15 cells at the ENRMF, produce most of the leachate. Assuming a constant disposal rate for radioactive waste and disposal into cells 4a to 11b (15 cells), it is therefore appropriate to scale the leached inventory during the operational period by a factor corresponding to the uncapped cell fraction. The current configuration at the ENRMF means that an uncapped cell factor of 4/15 is most appropriate.

23. During the operational period past records show that the amount of leachate will vary and the annual average up to the end of Q4 2104 is about 1,401 m^3. We have used this value in the revised calculations for Co-60 presented in Section 2.5 of this addendum. We have not used the Q2 2015 values since they represent an unusual situation that will not be repeated now the new hazardous waste permit has been granted. Once the site is fully capped the 2014 HRA calculates that about 70 m^3 of leachate will be produced per annum and this will need to be disposed off-site.

2.5 Co-60 in leachate

5) The reduced leachate volume suggested in point 4 is also of relevance for Co-60, as the dose for Co-60 at the leachate treatment facility (operational period) is reduced from 584 uSv/y in the draft Environmental Safety Case to 86 uSv/y in the final Environmental Safety Case. Despite this potentially high dose, Augean does not propose to use this scenario to limit the Co-60 capacity because (a) Co-60 accounts for only about 6% of the LLW activity in the UK national inventory, (b) the estimate assumes that all leachate is sent for treatment, which will not be the case and (c) the model is conservative because it does not take into account sorption within waste materials.

While we acknowledge these reasons, we would like Augean to provide additional justification as to why the leachate treatment facility (operational period) scenario has not been used to limit Co-60, in the context of the historic and planned use of the Avonmouth facility for treatment and disposal of leachate from the ENRMF.

24. Leachate is routinely monitored for Co-60. Ad-hoc monitoring of Co-60 by the EA recorded <1 Bq l^{-1} in leachate during 2014 (LGC Ltd, 2014). Routine monitoring of Co-60 by Augean has not found an increase in leachate concentrations as a result of Co-60 disposals.

25. The leachate treatment facility (operational period) scenario was not used to limit Co-60 disposals in the 2015 ESC because at the ENRMF the scenario is not certain since the leachate is predominantly used for the waste stabilisation plant on-site, and because the dose would be very low. Past disposals to the leachate treatment facility at Avonmouth are described in Section 2.4. Sometimes there is not sufficient leachate generated for the waste stabilisation plant and no leachate is transported off-site. However, sometimes there is an excess of leachate and this is the situation that leads to transport of the leachate off-site to the leachate treatment facility at Avonmouth. However, past disposals do not necessarily indicate the future pattern since this will be influenced by the maximum level of leachate that can be maintained in the landfill site and the fluctuations in the requirements of the waste stabilisation plant. There is no intention to increase the use of the leachate treatment facility: in fact the opposite is planned.

26. As described in the ESC, the national inventory of Co-60 (3.7 TBq) is such that the dose from the leachate treatment pathway will not exceed a few µSv y^{-1}. Note that the value of dose to the worker at the leachate treatment facility presented in the ESC (86 µSv y^{-1}) is a gross overestimate, based on disposal of the maximum inventory that can be disposed of at the ENRMF, as specified in the Development Order conditions (89.6 TBq), and ignoring radioactive decay during the operational period of the site, sorption on the waste and the fraction of the site that is uncapped.

27. At the end of the operational period, the dose to a treatment facility worker at Avonmouth from the disposal of 70 m^3 of leachate once the whole site is capped is estimated to be about 15 µSv y^{-1} from 89.6 TBq of Co-60, ignoring any radioactive decay and assuming that all Co-60 in the waste is soluble. This is a gross overestimate.

28. Following the review of leachate disposal to the off-site treatment facility, we have re-evaluated the assumption made in the ESC regarding this scenario. We have now decided to use the scenario to limit the radiological capacity of Co-60. We have evaluated doses to the leachate treatment facility worker for an annual average off-site leachate treatment rate

of 1,401 m^3, applying the uncapped cell factor described in Section 2.4. The results are shown in Table 5 and are overestimates since they ignore sorption on the waste. Results are also given for the leachate off-site treatment rate of 403.2 m^3 per annum given in the 2015 ESC, and the maximum off-site treatment rate of 2,236 m^3 per annum recorded prior to the 2015 Q2 campaign.

29. The radiological capacity of Co-60 in the 2015 ESC is 3.83 10^5 TBq based on the other scenarios. Using the leachate treatment scenario (at 1,400 m^3) to limit the Co-60 radiological capacity results in a radiological capacity of 22.5 TBq, less than the maximum inventory based on the Development Order (89.6 TBq), but still greater than the national inventory of Co-60 (3.7 TBq). Hence, it will now limit the Co-60 that can be received at the site. The results are given in Table 5. For all other radionuclides, doses from this scenario are limited by the maximum inventory (89.6 TBq) to below 20 μSv y^{-1} and are not considered further.

Table 5 Radiological capacity for Co-60 adjusted for various leachate treatment rates and the uncapped cell factor

Radionuclide	Radiological capacity (other scenarios) (TBq)	Radiological capacity based on off-site treatment leachate rates (TBq)		
		Low (403.2 m^3)	Reference (1,400 m^3)	High (2,236 m^3)
Co-60	3.83 10^5	7.81 10^1	2.25 10^1	1.41 10^1

Note: 403.2 m^3 was used in the 2015 ESC

2.6 Timing of bathtubbing event

6) Bathtubbing impacts are calculated in the final Environmental Safety Case at 450 y after closure ('the point in time the groundwater model suggests overtopping will occur', para 198). We note that in the draft Environmental Safety Case these impacts were calculated at 350 y after closure. We note also that modelling in the final Environmental Safety Case suggests that overtopping will occur, whereas the assumption in the draft Environmental Safety Case was that overtopping is uncertain to occur.

We ask Augean to explain these changes to the bathtubbing scenario impacts.

30. The hydrological risk assessment (HRA) asserts that overtopping would not occur at the site since it is a hazardous waste landfill site, leachate monitoring would continue while the Permit was in force, and the Permit would not be revoked if there was a risk of overtopping. However, the scenario was included in the 2015 ESC following discussions with the EA.

31. The draft ESC simply used a notional time of 350 years based on discussions with the authors of the HRA. The final ESC used a time of 450 years based on scoping calculations using the GoldSim groundwater model. This is described in Appendix E (section E4.5) of the 2015 ESC. We do not consider that this scenario is certain to occur and we recognise that the text could be clearer. The text in paragraphs 740 to 743 describes the basis for adopting a 450 year period, and clearly states in the last sentence that this scenario is unlikely to occur. The main report considers bathtubbing in paragraphs 198 to 202 and the

last sentence of paragraph 198 identifies a time at which the Goldsim groundwater model suggests overtopping will occur. This statement was a comment in relation to the scoping calculations performed using the Goldsim model and not an expectation that the event is expected to occur. The language used in the last sentence of paragraph 198 was not meant to suggest that this scenario was considered certain to occur or to contradict the earlier parts of that paragraph that concluded it is unlikely to occur.

32. The scoping calculations using the Goldsim groundwater model are based on the assumption that leachate management ceases 60 years after closure of the site, and the cap and liner gradually degrade beyond that time. The assumed end date for the period of authorisation is 60 years to be consistent with the other assumptions in the ESC. It is, by definition, inconsistent with the HRA since it does not consider that the period of authorisation will last for a longer period, until it is established that overtopping cannot occur (paragraph 740). Bathtubbing would not be expected to occur while an authorisation is in place since leachate monitoring and management would continue. The time at which bathtubbing can occur is based on a balance of the water input into the site (percolation through the cap) and the water output (leachate extraction and percolation through the sides and liner). The GoldSim calculations were cautious since they assumed that the sides remained impervious, maximising the risk of overtopping. Once the basal liner or facility side walls have fully degraded, overtopping cannot occur.

2.7 Basis for parameter changes

7) We have noted a number of changes to the calculation parameters used in the final Environmental Safety Case, in comparison to the values used in the draft:

- Infiltration rate to grassland is reduced by factor of ~5 in the final Environmental Safety Case (Table 43).

- The waste porosity is reduced from 0.5 to 0.1 in the final Environmental Safety Case (Table 46).

- The inhalation and irradiation dose coefficients for Ra-226, when Pb-210 is modelled explicitly, are different. The former has increased by factor of 3 (Table 54). We note that data from the 2011 Low Level Waste Repository Environmental Safety Case data were used in the draft Environmental Safety Case but ICRP data have been used in the final version.

We ask that Augean provides an explanation for these changes.

33. The values used have changed following a review of the database; some have changed to be consistent with the values given in the most recent version of the HRA, which was not available when the draft ESC was prepared.

Infiltration rate

34. The values of both the cap design infiltration rate and infiltration rate to grassland have increased between the draft and final ESC, not reduced as stated by EA. We are using 4.97 mm y^{-1} in the model for the Cap design infiltration rate and 74.3 mm y^{-1} for the infiltration rate to grassland, both the text (paragraph 409) and Table 43 were amended in

the final ESC. The cap design infiltration rate was 0.27 mm y^{-1} in the draft ESC (paragraph 411). The infiltration rate to grassland used in the draft ESC was 18.7 mm y^{-1}, a best estimate based on a log-normal distribution (paragraph 572).

35. The infiltration rate of 4.97 mm y^{-1} is based on the calculated maximum flux through the cap determined in the 2004 HRA based on the use of a composite cap comprising a geosynthetic clay liner (GCL) and a 1 mm thick linear low density polyethylene (LLDPE) geomembrane liner. Use of the calculated maximum flux is conservative and in the 2004 HRA the base case model assuming good cap performance has a flux through the cap of 0.27 mm y^{-1}. Supplementary calculations presented to the EA in support of the 2014 HRA investigated assumptions regarding the head of water above the cap, the density and size of defects in the cap and the range of potential hydraulic conductivity of the sub-grade which will underlie the geomembrane in the cap. The cap infiltration value of 4.97 mm y^{-1} is close to the middle of the range of the supplementary calculated cap infiltration rates. This is the value used in the 2014 HRA and in the final version of the 2015 ESC.

Waste porosity

36. The waste porosity was updated to be consistent with the assumptions used in the 2014 HRA. The 2011 HRA used a range of waste porosity from 0.455 to 0.556 (Table HRA 3) and the draft ESC adopted a mid-point value of 0.5. The final updated 2014 HRA used a range of waste porosity from 0.01 to 0.2 (Table HRA 2) and a mid-range value (0.1) was applied in the final ESC.

37. In the 2014 HRA the range of values assumed with regard to waste porosity is derived from information presented in the literature (McWhorter & Sunada, 1977) and is based on assumptions regarding the nature of the materials that will be deposited. The values used are based on literature values for effective porosity rather than total porosity. The use of effective porosity values in the 2014 HRA and 2015 ESC is more conservative than the use of total porosity values.

Dose coefficients

38. The Ra-226 inhalation value in Table 54 in the 2015 ESC is not correct and is not the value that was used in the calculations: this table was changed in error in the final version of the ESC. The Ra-226 parameters used in the model were the values reported in the draft ESC. Table 54 should read as follows:

Table 54 revised: Radium 226 and Th-232 dose coefficients used when Pb-210 and Ra-228 are modelled explicitly

Radionuclide	Ingestion (Sv Bq^{-1})	Inhalation (Sv Bq^{-1})	External Irradiation from slab (Sv y^{-1} Bq^{-1} kg)
Ra-226	2.8 10^{-7}	3.49 10^{-6}	3.03 10^{-6}
Th-232	2.3 10^{-7}	1.1 10^{-4}	1.41 10^{-10}

39. The LLWR data set provided a convenient source of data for dose coefficients that included doses from short-lived daughter radionuclides. This data set was used in the draft ESC. The final ESC calculated the dose coefficients from first principles using ICRP data and other original papers.

2.8 Groundwater monitoring programme

8) The Environmental Safety Case identifies the main contributors to dose from leachate migration via groundwater are likely to be H-3, Cl-36, Sr-90, I-129 and Pb-210 (para 171). The updated ESC goes on to say that 'groundwater monitoring for these radionuclides and comparison against background levels in groundwater ... would provide an indication of releases into the environment through this pathway'.

We note that Only H-3 and Pb-210 appear to be in the current groundwater analytical suite.

We ask Augean to review their current monitoring programme for groundwater, in light of the above and taking into account the predicted groundwater concentrations in point 2 above, and to provide us with the outcome of this review.

40. We have reviewed the monitoring programme taking into account the predicted groundwater concentrations, the detection limits and the expected doses from the predicted concentrations.

41. The peak groundwater concentrations at the boundary of the site are shown in Table 6 for the radionuclides listed above, assuming disposal of the maximum inventory (the minimum of the inventory permitted by the Development Order and the radiological capacity) for each radionuclide. The peak concentration during the period of authorisation (PoA) and that observed over the whole period modelled are presented. The tritium peak occurs during the PoA and the same concentration therefore appears in both columns.

Table 6 Peak groundwater concentrations at the site boundary, analytical detection limits and water concentrations producing a dose of 20 µSv

Radionuclide	Typical detection limit	Projected groundwater concentration after disposal of the maximum inventory[2]			Drinking water concentration giving a dose of 20 µSv ($Bq\ l^{-1}$)
		Peak ($Bq\ l^{-1}$)	Year of peak	Max during PoA ($Bq\ l^{-1}$)	
H-3	4 $Bq\ l^{-1}$	$2.39\ 10^{-1}$	44	$2.39\ 10^{-1}$	2000
Cl-36[1]	0.29 $Bq\ l^{-1}$	6.95	759	$3.73\ 10^{-2}$	20.8
Sr-90[1]	0.11 $Bq\ l^{-1}$	$7.01\ 10^{-5}$	138	$3.97\ 10^{-5}$	0.8
I-129[1]	0.02 $Bq\ l^{-1}$	$6.90\ 10^{-2}$	2100	$1.15\ 10^{-4}$	0.20
Pb-210	0.002 $Bq\ g^{-1}$	$4.33\ 10^{-8}$	95	$3.95\ 10^{-8}$	0.012

1. Detection limit reported for Sellafield groundwater assessments.
2. Minimum of the Development Order limit (89.6 TBq) and the radiological capacity

42. Typical detection limits are also listed. This shows that even if the maximum inventory (see Section 6 of the 2015 ESC) is disposed of at the site, the only radionuclide that could be detected in groundwater during the PoA is Cl-36. The detection limit corresponds to disposal of 11 TBq of Cl-36 at the site. However, the inventory of Cl-36 in the LLW waste that has been sent to ENRMF is a very small fraction of the total radionuclide inventory, amounting to a total of $3.1\ 10^{-5}$ TBq in June 2015.

43. The last column of the table provides an estimate of activity concentrations in water that result in a dose of 20 µSv y^{-1}, based on HPA assessments (Ewers & Mobbs, 2010). Their value is always greater than the projected groundwater concentration, indicating that doses from groundwater will be lower than 20 µSv y^{-1} even if the listed radionuclide is disposed of at the maximum inventory.

44. This review shows that, based on the maximum inventory that can be disposed of at the site and the radionuclide mix of the wastes, these radionuclides are very unlikely to be detected in groundwater using current techniques. Routine analysis of radionuclides that are expected to be at levels below the detection limits, and are found to be below the detections limits, does not provide any useful information. The two radionuclides that give the earliest peak groundwater concentrations are H-3, which peaks after 44 years, and Pb-210, which peaks after 95 years; both of these are included in routine groundwater analyses. The other radionuclides peak at much later times.

45. There is uncertainty associated with the groundwater model predictions and for this reason the ESC recommended reviewing the list of radionuclides routinely analysed in groundwater as the inventory accumulates (paragraphs 165 and 171). Thus, additional radionuclides would be analysed as the inventory of the radionuclides in the ENRMF increased and passed certain trigger levels. The trigger levels were not specified in the ESC. As part of this review, we have developed trigger levels for the radionuclides identified by EA above.

46. A factor of 30 is the largest ratio of maximum to minimum results in the groundwater model output observed in the sensitivity analyses. The radionuclide inventory corresponding to groundwater concentrations above the analytical detection limit is given in Table 9, together with the inventory assuming a factor of 30 uncertainty. We propose to use the inventory reduced by a factor of 30 (column 4 in Table 9) to trigger inclusion of a radionuclide in the analytical suite.

Table 7 Cumulative inventory to trigger inclusion of additional radionuclides in routine monitoring of groundwater

Radionuclide	Maximum inventory[1] (TBq)	Inventory required to exceed detection limit (TBq)	Inventory to exceed detection limit reduced by a factor 30 (TBq)	Inventory June 2015 (TBq)
H-3	$8.96\ 10^{1}$	$1.50\ 10^{3}$	$5.00\ 10^{1}$	$2.38\ 10^{-2}$
Cl-36	1.48	$1.15\ 10^{1}$	$3.82\ 10^{-1}$	$3.10\ 10^{-5}$
Sr-90	$8.96\ 10^{1}$	$2.48\ 10^{5}$	$8.28\ 10^{3}$	$2.73\ 10^{-3}$
I-129	$4.16\ 10^{-2}$	7.26	$2.42\ 10^{-1}$	$1.80\ 10^{-6}$
Pb-210	$8.96\ 10^{1}$	$4.99\ 10^{6}$	$1.66\ 10^{5}$	$1.47\ 10^{-2}$

Note 1. The maximum inventory is the minimum of the inventory permitted by the Development Order and the radiological capacity for each radionuclide

47. The Sr-90 trigger inventory remains greater than the maximum inventory and hence it will not be included in the routine analysis using this approach.

48. Routine groundwater monitoring will therefore continue with H-3 and Pb-210 as the radionuclides that are analysed for. If the levels are found to be above those expected, then following confirmation of the unexpected results, the analytical approach will be changed to look for all of the radionuclides identified above.

2.9 Timing of smallholder scenario

9) The sensitivity analysis for the smallholder potentially exposed group considers doses at earlier and later times than 200 y (which is the default time used for capacity determination). At times less than 90 y this scenario becomes limiting for Sr-90 (para 1101). At these times the radiological capacity for Sr-90 becomes less than the proposed 89.6 TBq capacity limit for the ENRMF. The current limiting scenario for Sr-90 is the smallholder at 200 y scenario but the proposed capacity of this radionuclide is limited by tonnage and not by dose (tables 24 and 25)

We note that 60 y is the assumed period of authorisation for the ENRMF site and so it seems reasonable that the Sr-90 capacity is brought in line with the smallholder intrusion scenario impacts (at 60 y).

We ask Augean to reassess the Sr-90 capacity based on peak smallholder impacts at 60 y and to provide us with the outcome of this assessment.

49. The timing of human intrusion scenarios is a matter of judgement. The philosophy behind the ESC is to use realistic but conservative assumptions, not bounding assumptions. Section 2.1 has discussed factors that will influence the timing of human intrusion scenarios affecting a site resident. We do not believe it is reasonable to assume that residential or smallholder developments will occur at the site 60 years after site closure i.e. at the time that it is assumed that the authorisation is revoked. The future development of the site will need planning permission and there are multiple public records (for example held by the EA, DECC and the local authority, planning records, the development consent order and local maps) showing the existence of the hazardous landfill site that will inform any future development at the site. Furthermore, the environment agencies are developing revocation guidance that will clarify the interaction between the environment agencies and the planning authorities at the time of the revocation of the Permit or Authorisation. This will specifically identify any planning controls that are required. The revocation of the Permit or Authorisation will involve assessment of the risks presented by the ENRMF, both from the point of view of the LLW wastes and of the hazardous wastes within the site. It is assumed for the purposes of the calculations that the Permit will be revoked 60 years after closure: if the EA consider that there is a need to retain controls for longer, e.g. to continue monitoring or to prevent intrusions into the site, then either the Permit will remain for longer or suitable planning controls will be put in place. Hence the timescales for intrusion considered in the ESC take into account the characteristics of the site and are appropriate. Intrusion with knowledge and understanding of the nature of the site is not relevant to the intrusion calculations in the ESC as the appropriate precautions will be taken.

50. Other assessments of near surface disposal facilities, e.g. the IAEA ISAM study (IAEA, 2004) discuss the active management and subsequent passive control periods for a near surface disposal site, suggesting periods of 100 years for the Permit followed by a period of 200 years during which passive (planning) controls are effective. This gives a total of 300 years post closure before intrusion is considered, considerably longer than the 200 years

assumed for the Smallholder scenario in the ESC which appears very cautious. LLWR assumed a minimum of 100 years post closure when knowledge of the site is maintained (Thorne, 2009) and we do not think it is reasonable to apply a shorter time span. It should be remembered that these timescales for unintentional or uninformed intrusion are values for the purposes of the calculation. There is no intention to destroy records of the presence of the site at that time, and in fact the control provided by the presence of the records could last for much longer.

51. The analysis of uncertainty in the ESC was undertaken to highlight the sensitivity of assessment calculations to selected parameters, and to illustrate the robustness of the calculations. There was no suggestion that the capacity values could be based on worst case assumptions: indeed to do so would be counter to accepted international guidance on assessment methodologies. It is not surprising that the doses from Sr-90 increase if the intrusion is assumed to occur earlier since Sr-90 has a half-life of about 30 years.

52. As requested by EA, we have calculated the radiological capacity of Sr-90 based on intrusion into the site corresponding to occupancy of the site by a smallholder occurring 60 years after site closure. We have also calculated the radiological capacity based on this intrusion occurring 100 years after site closure. These results, together with the results presented in the ESC for intrusion occurring at 200 years are given in Table 8.

Table 8 Maximum annual doses for adults: Smallholder scenario at 60, 100 and 200 years

Year	Total dose	
	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Radiological capacity (TBq)
60	$7.15\ 10^{-5}$	$4.19\ 10^{1}$
100	$2.72\ 10^{-5}$	$1.10\ 10^{2}$
200	$2.42\ 10^{-6}$	$1.24\ 10^{3}$

2.10 Use of peak doses to determine radiological capacity from intrusion scenarios

10) We note that radiological capacities for a number of other radionuclides are also based on the smallholder scenario at 200 y and this scenario provides the limiting capacity for Th-230, Th-232 and Pa-231. Since we cannot necessarily rely on any controls on land use after the assumed 60 y period of authorisation we consider it reasonable for the capacities for Th-230, Th-232 and Pa-231 to be brought in line with their respective peak doses calculated throughout the assessment period (periods between 60 and 20,000 y, as opposed to the dose at 200 y.

We ask Augean to reassess the capacities for Th-230, Th-232 and Pa-231 based on their respective peak doses calculated over the assessment period and to provide us with the outcome of this assessment.

53. As explained above, we think that it is inappropriate to assume that no controls will be present on the redevelopment of the site at the end of the authorisation period. As part of the planning permission for development of the site, it will be restored to grassland and

mixed scrub/woodland. Any development for residential or smallholding will require planning permission and it is not reasonable to assume that this would be granted immediately after the Permit was revoked unless a risk assessment showed that the resulting doses would be acceptable. This would also take into account the hazardous waste at the site. Nevertheless, we understand that radiological assessment methodologies consider that passive control through records cannot be assumed to last forever and this is the basis of the intrusion assessments. Small scale intrusions would be expected to occur earlier than large scale intrusions that would require planning permission. This is why we have assessed a small scale intrusion at 60 years post closure and larger scale intrusions at later dates.

54. For most radionuclides the peak dose from intrusion scenarios occurs at the earliest time that the intrusion is expected to occur. However, for long lived radionuclides where gradual ingrowth of daughter radionuclides is important, this may not be the case. As requested by EA we have calculated the peak doses from the smallholder scenario for the three radionuclides listed. The results are summarised in Table 9, along with the results assuming intrusion occurs at 200 years.

55. For Th-230 the peak dose arises for intrusion occurring 9,000 years after closure. For Pa-231 and Th-232 the intrusion dose is practically constant for the first 200 years post closure, slowly reducing over time. Hence, the peak intrusion dose over the assessment period for Pa-231 and Th-232 is practically identical to the dose for intrusion at 200 years. However, for Th-230, the peak dose is about an order of magnitude greater than the dose from intrusion at 200 years. Although this peak dose occurs 9,000 years after closure, a very long time after closure, it can be argued that the radiological capacity for Th-230 for the smallholder intrusion scenario should be based on the dose at 9,000 years rather than at 200 years.

Table 9 Maximum annual doses for adults: Smallholder scenario

Radionuclide	Maximum over 60-20,000 years		Results for intrusion at 200 years		Comment
	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Radiological capacity (TBq)	Dose ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Radiological capacity (TBq)	
Pa-231	$1.63\ 10^{-4}$	$1.84\ 10^{1}$	$1.61\ 10^{-4}$	$1.86\ 10^{1}$	practically constant for 200 years
Th-230	$5.17\ 10^{-4}$	5.8	$4.33\ 10^{-5}$	$6.93\ 10^{1}$	Maximum dose occurs at 9,000 y
Th-232	$4.24\ 10^{-5}$	$7.08\ 10^{1}$	$4.19\ 10^{-5}$	$7.16\ 10^{1}$	Practically constant for 200 years

2.11 Exposure of protected species

11) The final Environmental Safety Case includes an assessment of dose to rabbits burrowing into the waste materials. This assessment of this scenario indicates potential doses of > 40 uGy/h to the rabbits at 60 y (max 210 uGy/h). This is a limiting scenario for Pa-231, Cm-243 and Cm-244. The Environmental Safety Case proposes that radiological capacities for these radionuclides could be set for this scenario using the 40 uGy/h action *level - resulting in radiological capacity reduction factors of 4, 6 and 3 for Pa-231, Cm-243 and Cm-244 respectively (table 141).*

We note that rabbits are not a protected species and our assessment does not indicate any significant adverse impact on a European site, Site of Special Scientific Interest, Area of Outstanding Natural Beauty or other conservation site.

However, limiting the capacity of the radionuclides that lead to doses in excess of 40 uGy/h for other burrowing species that are protected, such as badgers, should be precautionary, given that these radionuclides are very unlikely to form a significant part of the ENRMF inventory, and that ecological protection is considered at the population level as opposed to the individual level.

We ask Augean to consider reducing the capacity of Pa-231, Cm-243 and Cm-244, based on the outcome of the assessment of the potential doses to non-human species that may burrow into the waste in future.

56. The assessment undertaken for burrowing animals using the ERICA model is generic and applies to other burrowing species that could burrow deep enough to reach the waste. Badger tunnels can be four metres deep, though most are less than one metre deep. Hence it is appropriate to consider them in the assessment and the results given for rabbits are also applicable to badgers.

57. The waste management principal of ‘concentrate and contain’ will inevitably mean that some sites will contain wastes while the rest of the environment does not. Hence, it could be argued that protection of the environment is already achieved by placing the wastes in a specific facility. However, we note the EA comment that it would be precautionary to apply the radiological capacity reduction factors given above to limit the dose rate to badgers intruding into the waste to 40 µGy/h. Given that the radionuclides that these reduction factors apply to are very unlikely to form a significant part of the ENRMF inventory, we have applied these reduction factors to the radiological capacity. The revised values are given in Table 10.

Table 10 Reduction factors for radiological capacity due to burrowing animals

Radionuclide	Radiological capacity (TBq) 2015 ESC	Reduction factor[1]	Adjusted Radiological capacity (TBq)
Pa-231	$1.86\ 10^{1}$	3.9	4.76
Cm-243	$4.28\ 10^{3}$	5.2	$8.18\ 10^{2}$
Cm-244	$1.59\ 10^{4}$	2.2	$7.24\ 10^{3}$

Note 1. ESC rounded up all values.

3 Corrigenda

58. In responding to the EA comments two tables in the ESC were found to contain transcription errors. These errors were in the lower part of each table and were specific to these tables only. No other tables in the ESC were affected by these errors and the conclusions of the assessment were unchanged. The corrected tables are included here and given their original table numbers. Corrected cells are shaded grey.

Table 21 revised: Scenario radiological capacity calculated for exposures after the period of authorisation

Radionuclide	Bathtubbing		Groundwater (Well at boundary)		Recreational user	
	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)	Dose per MBq (µSv y^{-1} MBq^{-1})	Scenario Radiological Capacity (MBq)
H-3	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$5.95\ 10^{-3}$	$3.01\ 10^{11}$
C-14	$1.10\ 10^{-11}$	$1.82\ 10^{12}$	$3.49\ 10^{-9}$	$5.73\ 10^{9}$	2.97	$6.04\ 10^{8}$
Cl-36	$1.25\ 10^{-7}$	$1.60\ 10^{8}$	$1.35\ 10^{-5}$	$1.48\ 10^{6}$	$3.93\ 10^{-25}$	$7.51\ 10^{31}$
Fe-55	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	0	nd*
Co-60	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.75\ 10^{-13}$	$1.02\ 10^{22}$
Ni-63	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	0	nd*
Sr-90	$1.58\ 10^{-12}$	$1.27\ 10^{13}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$8.48\ 10^{-21}$	$2.11\ 10^{29}$
Nb-94	$3.45\ 10^{-9}$	$5.80\ 10^{9}$	$2.23\ 10^{-9}$	$8.96\ 10^{9}$	$2.50\ 10^{-12}$	$7.18\ 10^{20}$
Tc-99	$2.02\ 10^{-7}$	$9.92\ 10^{7}$	$1.26\ 10^{-7}$	$1.58\ 10^{8}$	$4.52\ 10^{-45}$	$3.97\ 10^{53}$
Ru-106	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$7.49\ 10^{-32}$	$2.39\ 10^{40}$
Ag-108m	$2.93\ 10^{-9}$	$6.83\ 10^{9}$	$9.03\ 10^{-10}$	$2.21\ 10^{10}$	$1.33\ 10^{-13}$	$1.34\ 10^{22}$
Sb-125	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$3.08\ 10^{-21}$	$5.83\ 10^{29}$
Sn-126	$1.21\ 10^{-9}$	$1.66\ 10^{10}$	$9.10\ 10^{-8}$	$2.20\ 10^{8}$	$9.80\ 10^{-14}$	$1.83\ 10^{22}$
I-129	$1.59\ 10^{-7}$	$1.26\ 10^{8}$	$4.80\ 10^{-4}$	$4.17\ 10^{4}$	$1.25\ 10^{-150}$	$6.65\ 10^{155}$
Ba-133	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.35\ 10^{-19}$	$7.63\ 10^{27}$
Cs-134	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.59\ 10^{-21}$	$1.13\ 10^{30}$
Cs-137	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$4.19\ 10^{-14}$	$4.28\ 10^{22}$
Pm-147	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$5.77\ 10^{-51}$	$3.11\ 10^{59}$
Eu-152	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$6.57\ 10^{-13}$	$2.73\ 10^{21}$
Eu-154	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.62\ 10^{-13}$	$1.11\ 10^{22}$
Eu-155	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.98\ 10^{-40}$	$6.02\ 10^{48}$
Pb-210	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$9.45\ 10^{-19}$	$1.90\ 10^{27}$

Radionuclide	Bathtubbing		Groundwater (Well at boundary)		Recreational user	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
Ra-226	$2.74\ 10^{-9}$	$7.29\ 10^{9}$	$5.15\ 10^{-9}$	$3.88\ 10^{9}$	$1.30\ 10^{-8}$	$1.38\ 10^{17}$
Ra-228	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.90\ 10^{-11}$	$6.18\ 10^{19}$
Ac-227	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.96\ 10^{-17}$	$9.15\ 10^{25}$
Th-229	$5.26\ 10^{-11}$	$3.81\ 10^{11}$	$3.68\ 10^{-8}$	$5.44\ 10^{8}$	$6.13\ 10^{-14}$	$2.92\ 10^{22}$
Th-230	$5.74\ 10^{-10}$	$3.48\ 10^{10}$	$6.94\ 10^{-8}$	$2.88\ 10^{8}$	$2.29\ 10^{-29}$	$6.05\ 10^{37}$
Th-232	$3.08\ 10^{-9}$	$6.50\ 10^{9}$	$1.23\ 10^{-7}$	$1.63\ 10^{8}$	$6.01\ 10^{-11}$	$2.38\ 10^{19}$
Pa-231	$1.55\ 10^{-9}$	$1.29\ 10^{10}$	$9.02\ 10^{-8}$	$2.22\ 10^{8}$	$2.33\ 10^{-17}$	$1.60\ 10^{25}$
U-232	$5.16\ 10^{-12}$	$3.87\ 10^{12}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$4.96\ 10^{-34}$	$3.61\ 10^{42}$
U-233	$1.18\ 10^{-10}$	$1.69\ 10^{11}$	$6.38\ 10^{-7}$	$3.13\ 10^{7}$	$1.22\ 10^{-16}$	$5.14\ 10^{24}$
U-234	$8.76\ 10^{-11}$	$2.28\ 10^{11}$	$3.12\ 10^{-6}$	$6.41\ 10^{6}$	$4.33\ 10^{-38}$	$2.96\ 10^{45}$
U-235	$1.42\ 10^{-9}$	$1.41\ 10^{10}$	$4.07\ 10^{-6}$	$4.92\ 10^{6}$	$4.06\ 10^{-24}$	$2.42\ 10^{31}$
U-236	$7.95\ 10^{-11}$	$2.52\ 10^{11}$	$1.39\ 10^{-7}$	$1.44\ 10^{8}$	$2.23\ 10^{-19}$	$8.03\ 10^{27}$
U-238	$3.09\ 10^{-10}$	$6.48\ 10^{10}$	$7.89\ 10^{-7}$	$2.53\ 10^{7}$	$3.07\ 10^{-17}$	$1.65\ 10^{25}$
Np-237	$1.88\ 10^{-8}$	$1.06\ 10^{9}$	$4.43\ 10^{-5}$	$4.52\ 10^{5}$	$2.60\ 10^{-32}$	$3.48\ 10^{38}$
Pu-238	$7.99\ 10^{-13}$	$2.50\ 10^{13}$	$8.28\ 10^{-10}$	$2.42\ 10^{10}$	$7.57\ 10^{-44}$	$2.37\ 10^{52}$
Pu-239	$3.47\ 10^{-11}$	$5.76\ 10^{11}$	$6.62\ 10^{-9}$	$3.02\ 10^{9}$	$7.65\ 10^{-26}$	$2.34\ 10^{34}$
Pu-240	$3.35\ 10^{-11}$	$5.96\ 10^{11}$	$1.51\ 10^{-9}$	$1.32\ 10^{10}$	$7.90\ 10^{-43}$	$2.27\ 10^{51}$
Pu-241	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.92\ 10^{-10}$	$1.04\ 10^{11}$	$1.40\ 10^{-37}$	$1.28\ 10^{46}$
Pu-242	$3.28\ 10^{-11}$	$6.10\ 10^{11}$	$4.06\ 10^{-8}$	$4.93\ 10^{8}$	$1.01\ 10^{-24}$	$1.77\ 10^{33}$
Am-241	$6.89\ 10^{-12}$	$2.90\ 10^{12}$	$8.91\ 10^{-9}$	$2.24\ 10^{9}$	$4.92\ 10^{-38}$	$3.64\ 10^{46}$
Cm-243	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$1.92\ 10^{-24}$	$9.33\ 10^{32}$
Cm-244	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$<1.0\ 10^{-10}$	$>2.0\ 10^{11}$	$2.62\ 10^{-54}$	$6.84\ 10^{62}$

* Where dose is effectively zero the radiological capacity is infinite, marked here as nd (not determined).

Table 23 revised: Scenario radiological capacity calculated for exposures from human intrusion – residents and smallholders

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
H-3	$3.53\ 10^{-10}$	$8.51\ 10^{12}$	$2.81\ 10^{-11}$	$1.07\ 10^{14}$	$2.95\ 10^{-2}$	$6.06\ 10^{10}$
C-14	$4.35\ 10^{-6}$	$6.90\ 10^{8}$	$3.71\ 10^{-6}$	$8.08\ 10^{8}$	$3.84\ 10^{2}$	$4.67\ 10^{6}$
Cl-36	$5.47\ 10^{-6}$	$5.49\ 10^{8}$	$5.55\ 10^{-5}$	$5.40\ 10^{7}$	$1.29\ 10^{-24}$	$2.30\ 10^{31}$
Fe-55	$1.11\ 10^{-26}$	$2.69\ 10^{29}$	$7.76\ 10^{-31}$	$3.87\ 10^{33}$	0	nd*
Co-60	$8.26\ 10^{-14}$	$3.63\ 10^{16}$	$1.35\ 10^{-16}$	$2.23\ 10^{19}$	$4.16\ 10^{-18}$	$4.31\ 10^{26}$
Ni-63	$1.91\ 10^{-9}$	$1.57\ 10^{12}$	$1.08\ 10^{-8}$	$2.78\ 10^{11}$	0	nd*
Sr-90	$1.13\ 10^{-6}$	$2.64\ 10^{9}$	$2.42\ 10^{-6}$	$1.24\ 10^{9}$	$3.18\ 10^{-21}$	$5.64\ 10^{29}$
Nb-94	$1.81\ 10^{-5}$	$1.66\ 10^{8}$	$2.09\ 10^{-5}$	$1.44\ 10^{8}$	$8.14\ 10^{-12}$	$2.20\ 10^{20}$
Tc-99	$7.52\ 10^{-6}$	$3.99\ 10^{8}$	$3.31\ 10^{-5}$	$9.07\ 10^{7}$	$1.48\ 10^{-44}$	$1.21\ 10^{53}$
Ru-106	$1.80\ 10^{-50}$	$1.66\ 10^{53}$	$6.64\ 10^{-65}$	$4.52\ 10^{67}$	$8.05\ 10^{-58}$	$2.23\ 10^{66}$
Ag-108m	$1.41\ 10^{-5}$	$2.13\ 10^{8}$	$1.50\ 10^{-5}$	$2.00\ 10^{8}$	$3.76\ 10^{-13}$	$4.77\ 10^{21}$
Sb-125	$1.98\ 10^{-22}$	$1.51\ 10^{25}$	$8.08\ 10^{-28}$	$3.71\ 10^{30}$	$1.52\ 10^{-30}$	$1.18\ 10^{39}$
Sn-126	$5.35\ 10^{-6}$	$5.60\ 10^{8}$	$8.33\ 10^{-6}$	$3.60\ 10^{8}$	$3.20\ 10^{-13}$	$5.59\ 10^{21}$
I-129	$1.30\ 10^{-5}$	$2.30\ 10^{8}$	$1.12\ 10^{-4}$	$2.69\ 10^{7}$	$4.10\ 10^{-150}$	$2.03\ 10^{155}$
Ba-133	$1.90\ 10^{-10}$	$1.58\ 10^{13}$	$8.22\ 10^{-12}$	$3.65\ 10^{14}$	$2.04\ 10^{-21}$	$8.76\ 10^{29}$
Cs-134	$2.50\ 10^{-27}$	$1.20\ 10^{30}$	$1.92\ 10^{-34}$	$1.56\ 10^{37}$	$3.94\ 10^{-34}$	$4.55\ 10^{42}$
Cs-137	$2.18\ 10^{-7}$	$1.38\ 10^{10}$	$1.23\ 10^{-7}$	$2.44\ 10^{10}$	$1.73\ 10^{-14}$	$1.03\ 10^{23}$
Pm-147	$7.47\ 10^{-27}$	$4.02\ 10^{29}$	$5.81\ 10^{-32}$	$5.16\ 10^{34}$	$8.85\ 10^{-61}$	$2.02\ 10^{69}$
Eu-152	$6.06\ 10^{-9}$	$4.95\ 10^{11}$	$5.41\ 10^{-10}$	$5.54\ 10^{12}$	$2.15\ 10^{-14}$	$8.35\ 10^{22}$
Eu-154	$8.02\ 10^{-11}$	$3.74\ 10^{13}$	$1.64\ 10^{-12}$	$1.83\ 10^{15}$	$3.72\ 10^{-16}$	$4.82\ 10^{24}$
Eu-155	$1.12\ 10^{-16}$	$2.67\ 10^{19}$	$9.02\ 10^{-20}$	$3.33\ 10^{22}$	$1.99\ 10^{-45}$	$9.02\ 10^{53}$
Pb-210	$2.19\ 10^{-7}$	$1.37\ 10^{10}$	$1.97\ 10^{-7}$	$1.52\ 10^{10}$	$1.86\ 10^{-19}$	$9.63\ 10^{27}$
Ra-226**	$1.62\ 10^{-11}$	$1.85\ 10^{14}$	$1.49\ 10^{-11}$	$2.01\ 10^{14}$	$9.81\ 10^{-6}$	$1.83\ 10^{14}$

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)
Ra-226***	$1.20\ 10^{-3}$	$2.51\ 10^{6}$	$5.08\ 10^{-4}$	$5.90\ 10^{6}$	$3.26\ 10^{-2}$	$1.37\ 10^{9}$
Ra-228	$9.87\ 10^{-13}$	$3.04\ 10^{15}$	$7.39\ 10^{-15}$	$4.06\ 10^{17}$	$1.84\ 10^{-15}$	$9.72\ 10^{23}$
Ac-227	$6.63\ 10^{-8}$	$4.53\ 10^{10}$	$2.37\ 10^{-8}$	$1.26\ 10^{11}$	$3.65\ 10^{-18}$	$4.91\ 10^{26}$
Th-229	$4.85\ 10^{-6}$	$6.18\ 10^{8}$	$8.17\ 10^{-6}$	$3.67\ 10^{8}$	$1.99\ 10^{-13}$	$9.01\ 10^{21}$
Th-230	$8.55\ 10^{-6}$	$3.51\ 10^{8}$	$4.33\ 10^{-5}$	$6.93\ 10^{7}$	$2.25\ 10^{-28}$	$6.17\ 10^{36}$
Th-232	$3.24\ 10^{-5}$	$9.26\ 10^{7}$	$4.19\ 10^{-5}$	$7.16\ 10^{7}$	$1.97\ 10^{-10}$	$7.29\ 10^{18}$
Pa-231	$4.25\ 10^{-5}$	$7.06\ 10^{7}$	$1.61\ 10^{-4}$	$1.86\ 10^{7}$	$8.83\ 10^{-17}$	$4.21\ 10^{24}$
U-232	$1.76\ 10^{-7}$	$1.71\ 10^{10}$	$3.27\ 10^{-7}$	$9.17\ 10^{9}$	$6.57\ 10^{-34}$	$2.73\ 10^{42}$
U-233	$2.11\ 10^{-7}$	$1.42\ 10^{10}$	$5.57\ 10^{-7}$	$5.39\ 10^{9}$	$9.93\ 10^{-16}$	$6.31\ 10^{23}$
U-234	$1.36\ 10^{-7}$	$2.20\ 10^{10}$	$3.88\ 10^{-7}$	$7.74\ 10^{9}$	$3.54\ 10^{-37}$	$3.62\ 10^{44}$
U-235	$1.66\ 10^{-6}$	$1.81\ 10^{9}$	$2.63\ 10^{-6}$	$1.14\ 10^{9}$	$3.18\ 10^{-23}$	$3.09\ 10^{30}$
U-236	$1.28\ 10^{-7}$	$2.34\ 10^{10}$	$3.67\ 10^{-7}$	$8.17\ 10^{9}$	$1.83\ 10^{-18}$	$9.81\ 10^{26}$
U-238	$3.76\ 10^{-7}$	$7.99\ 10^{9}$	$6.59\ 10^{-7}$	$4.55\ 10^{9}$	$1.01\ 10^{-16}$	$5.04\ 10^{24}$
Np-237	$2.92\ 10^{-6}$	$1.03\ 10^{9}$	$5.32\ 10^{-6}$	$5.64\ 10^{8}$	$2.12\ 10^{-31}$	$4.26\ 10^{37}$
Pu-238	$2.53\ 10^{-7}$	$1.18\ 10^{10}$	$3.18\ 10^{-7}$	$9.43\ 10^{9}$	$1.31\ 10^{-43}$	$1.36\ 10^{52}$
Pu-239	$8.99\ 10^{-7}$	$3.34\ 10^{9}$	$1.67\ 10^{-6}$	$1.79\ 10^{9}$	$2.50\ 10^{-25}$	$7.18\ 10^{33}$
Pu-240	$8.88\ 10^{-7}$	$3.38\ 10^{9}$	$1.65\ 10^{-6}$	$1.82\ 10^{9}$	$6.44\ 10^{-42}$	$2.78\ 10^{50}$
Pu-241	$2.39\ 10^{-8}$	$1.26\ 10^{11}$	$4.50\ 10^{-8}$	$6.67\ 10^{10}$	$5.95\ 10^{-39}$	$3.01\ 10^{47}$
Pu-242	$8.44\ 10^{-7}$	$3.56\ 10^{9}$	$1.59\ 10^{-6}$	$1.89\ 10^{9}$	$8.26\ 10^{-24}$	$2.17\ 10^{32}$
Am-241	$6.93\ 10^{-7}$	$4.33\ 10^{9}$	$1.30\ 10^{-6}$	$2.30\ 10^{9}$	$3.75\ 10^{-37}$	$4.78\ 10^{45}$
Cm-243	$4.66\ 10^{-8}$	$6.44\ 10^{10}$	$2.13\ 10^{-8}$	$1.41\ 10^{11}$	$7.37\ 10^{-25}$	$2.43\ 10^{33}$
Cm-244	$3.84\ 10^{-9}$	$7.82\ 10^{11}$	$4.95\ 10^{-9}$	$6.06\ 10^{11}$	$9.42\ 10^{-54}$	$1.90\ 10^{62}$

* Where dose is effectively zero the radiological capacity is infinite, marked here as nd (not determined).

Radionuclide	Residential occupant (150 y)		Smallholder (200 y)		Resident – cap intact (150 y)	
	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)	Dose per MBq ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Scenario Radiological Capacity (MBq)

** Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface

*** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

4 Updates resulting from responses to EA questions

59. Annex B presents information and arguments to support a category of “Other radionuclides” that only includes radionuclides with a half-life greater than 1 year. Radiological capacities are also provided in Annex B for radionuclides between 3 months and 1 year. This information could be used either to define a second “Other radionuclides” group for short half-life radionuclides or to include radiological capacities for each of the short-half-life radionuclides discussed.

4.1 Summary of changes

60. The following table summarises the changes discussed in the report (Table 11).

Table 11 Changes suggested in this addendum

Section	Subject	Change to radiological capacity
1, 4 and Annex B	Short half-lives	New radiological capacities that could be applied to short half-life radionuclides in Schedule 3
2.1	Large scale intrusion	No change
2.2	Time dependent trends in GW concentration	No change
2.3	Cl36 infant dose	Reduced radiological capacity for Cl-36
2.4	Monthly leachate treatment	No change
2.5	Co-60 leachate treatment	Reduced radiological capacity for Co-60
2.6	Timing of bathtubbing	No change
2.7	Parameter value changes	No change
2.8	Groundwater monitoring	No change
2.9	Timing of smallholder scenario	No change
2.10	Peak dose for intrusion scenarios	Reduced radiological capacity for 3 radionuclides
2.11	Exposure of protected species	Reduced radiological capacity for 3 radionuclides

61. The revised radiological capacity values are shown in Table 25 and 26. These tables (Table 25 and Table 26) are shown with the original ESC numbers for ease of reference and modified rows are shaded grey.

Table 25 revised: ENRMF Radiological capacity and constraint

Radionuclide	Radiological Capacity (TBq)	Scenario	Constraint*
H-3	$4.12\ 10^{3}$	Recreational (0 years)	
C-14	$1.20\ 10^{2}$	Recreational (0 years)	
Cl-36	$3.70\ 10^{-1}$	Well at boundary (All pathways)	Limiting capacity
Fe-55	$5.83\ 10^{13}$	Excavator (Borehole) 60 years	
Co-60	$2.25\ 10^{1}$	Leachate treatment	Limiting capacity
Ni-63	$2.78\ 10^{5}$	Small holding 200 years	
Sr-90	$1.24\ 10^{3}$	Small holding 200 years	
Nb-94	$1.44\ 10^{2}$	Small holding 200 years	
Tc-99	$9.07\ 10^{1}$	Small holding 200 years	
Ru-106	$5.29\ 10^{16}$	Recreational (0 years)	
Ag-108m	$2.00\ 10^{2}$	Small holding 200 years	
Sb-125	$3.33\ 10^{9}$	Excavator (Borehole) 60 years	
Sn-126	$2.20\ 10^{2}$	Well at boundary (All pathways)	
I-129	$4.17\ 10^{-2}$	Well at boundary (All pathways)	Limiting capacity
Ba-133	$6.11\ 10^{4}$	Excavator (Borehole) 60 years	
Cs-134	$1.37\ 10^{11}$	Excavator (Borehole) 60 years	
Cs-137	$2.70\ 10^{3}$	Excavator (Borehole) 60 years	
Pm-147	$4.74\ 10^{13}$	Excavator (Borehole) 60 years	
Eu-152	$7.16\ 10^{3}$	Excavator (Borehole) 60 years	
Eu-154	$3.81\ 10^{4}$	Excavator (Borehole) 60 years	
Eu-155	$7.91\ 10^{7}$	Excavator (Borehole) 60 years	
Pb-210	$4.44\ 10^{3}$	Well at boundary (All pathways) POA	
Ra-226**	$1.56\ 10^{2}$	Excavator (Borehole) 60 years	
Ra-226***	2.51	Residential 150 years	Limiting capacity
Ra-228	$1.80\ 10^{5}$	Excavator (Borehole) 60 years	
Ac-227	$1.02\ 10^{3}$	Excavator (Borehole) 60 years	
Th-229	$2.98\ 10^{2}$	Excavator (Borehole) 60 years	
Th-230	5.80	Small holding 200 years	Limiting capacity
Th-232	$7.08\ 10^{1}$	Small holding 200 years	Limiting capacity
Pa-231	4.76	Exposure of non-human species	Limiting capacity
U-232	$4.25\ 10^{3}$	Excavator (Borehole) 60 years	
U-233	$3.13\ 10^{1}$	Well at boundary (All pathways)	Limiting capacity
U-234	6.41	Well at boundary (All pathways)	Limiting capacity
U-235	4.92	Well at boundary (All pathways)	Limiting capacity
U-236	$1.44\ 10^{2}$	Well at boundary (All pathways)	
U-238	$2.53\ 10^{1}$	Well at boundary (All pathways)	Limiting capacity
Np-237	$4.52\ 10^{-1}$	Well at boundary (All pathways)	Limiting capacity
Pu-238	$1.40\ 10^{3}$	Excavator (Borehole) 60 years	
Pu-239	$8.01\ 10^{2}$	Excavator (Borehole) 60 years	
Pu-240	$8.05\ 10^{2}$	Excavator (Borehole) 60 years	

Radionuclide	Radiological Capacity (TBq)	Scenario	Constraint*
Pu-241	$3.20\ 10^{4}$	Excavator (Borehole) 60 years	
Pu-242	$4.93\ 10^{2}$	Well at boundary (All pathways)	
Am-241	$1.08\ 10^{3}$	Excavator (Borehole) 60 years	
Cm-243	$8.18\ 10^{2}$	Exposure of non-human species	
Cm-244	$7.24\ 10^{3}$	Exposure of non-human species	

*"Limiting capacity" identifies those radionuclides where the radiological capacity is less than inventory arising from disposing of 448,000 t of LLW at 200 Bq g^{-1}.

** Assuming that wastes containing significant activity concentrations of Ra-226 are 5m below the restored surface

*** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

Table 26 revised: Suggested Schedule 3 – Disposals of radioactive waste and monitoring

Table 3.1 Disposal by burial on the premises				
Waste type	Disposal route	Sum of fractions limits		
		Radionuclide or group of nuclides	Relevant value (TBq)	Maximum total volume
Solid waste with a maximum total activity concentration of 200 Bq/g	Burial on the premises in cells 4B, 5A, 5B, 6, 7, 8, 9, 10 and 11 of the East Northants Resource Management Facility.	H-3	$4.12\ 10^{3}$	Not specified
		C-14	$1.20\ 10^{2}$	
		Cl-36	$3.70\ 10^{-1}$	
		Fe-55	$5.83\ 10^{13}$	
		Co-60	$2.25\ 10^{1}$	
		Ni-63	$2.78\ 10^{5}$	
		Sr-90	$1.24\ 10^{3}$	
		Nb-94	$1.44\ 10^{2}$	
		Tc-99	$9.07\ 10^{1}$	
		Ru-106	$5.29\ 10^{16}$	
		Ag-108m	$2.00\ 10^{2}$	
		Sb-125	$3.33\ 10^{9}$	
		Sn-126	$2.20\ 10^{2}$	
		I-129	$4.17\ 10^{-2}$	
		Ba-133	$6.11\ 10^{4}$	
		Cs-134	$1.37\ 10^{11}$	
		Cs-137	$2.70\ 10^{3}$	
		Pm-147	$4.74\ 10^{13}$	
		Eu-152	$7.16\ 10^{3}$	
		Eu-154	$3.81\ 10^{4}$	
		Eu-155	$7.91\ 10^{7}$	
		Pb-210***	$4.44\ 10^{3}$	

Table 3.1 Disposal by burial on the premises				
Waste type	Disposal route	Sum of fractions limits		
		Radionuclide or group of nuclides	Relevant value (TBq)	Maximum total volume
		Ra-226*	$1.56\ 10^{2}$	
		Ra-226**	2.51	
		Ra-228***	$1.80\ 10^{5}$	
		Ac-227	$1.02\ 10^{3}$	
		Th-229	$2.98\ 10^{2}$	
		Th-230	5.80	
		Th-232	$7.08\ 10^{1}$	
		Pa-231	4.76	
		U-232	$4.25\ 10^{3}$	
		U-233	$3.13\ 10^{1}$	
		U-234	6.41	
		U-235	4.92	
		U-236	$1.44\ 10^{2}$	
		U-238	$2.53\ 10^{1}$	
		Np-237	$4.52\ 10^{-1}$	
		Pu-238	$1.40\ 10^{3}$	
		Pu-239	$8.01\ 10^{2}$	
		Pu-240	$8.05\ 10^{2}$	
		Pu-241	$3.20\ 10^{4}$	
		Pu-242	$4.93\ 10^{2}$	
		Am-241	$1.08\ 10^{3}$	
		Cm-243	$8.18\ 10^{2}$	
		Cm-244	$7.24\ 10^{3}$	
		Any other radionuclide	$4.17\ 10^{-2}$	

* Assuming that wastes containing significant activity concentrations of Ra-226 are 5 m below the restored surface

** Wastes not containing significant activity concentrations of Ra-226 (<5 Bq g^{-1})

*** Only applies to activity that is not supported by the parent

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Annex A. EA Correspondence

From: "Green, Rob"
Date: 8 October 2015 10:05:10 GMT
[redacted]"
Subject: Proposed Definition of 'Other Radionuclides' in ESC to support ENRMF variation application

Dear Gene,

As mentioned previously I have noted in the Environmental Safety Case, submitted as part of Augean's application to vary the permit at the ENRMF site, that there is an assumption that 'other radionuclides' to be limited in the permit should include radionuclides not otherwise listed that have half lives of greater than a year. This is a somewhat longer timescale than the half lives of greater than 3 months definition of 'other radionuclides' that is commonly used in the permits that we have previously issued elsewhere, including for other landfill sites such as Clifton Marsh and Lillyhall.

I spoke briefly to Nick Mitchell regarding this and he suggested that the proposed definition in the ESC for 'other radionuclides' was decided on due to operational reasons. However, he could not be more specific.

You'll appreciate that there is a need for us to ensure that we are consistent in our decision making. Consistency is achieved, in part, by making judgments based on sound evidence.

We don't feel that sufficient evidence or reasoning has been presented in the application as to why the definition for 'other radionuclides' should include those radionuclides that are not otherwise listed in the permit and have half lives of greater than a year, as opposed to greater than 3 months.

However, we are keen to better understand Augean's reasoning for this suggestion and so ask that you write to us with your justification for this proposal, which we will consider as part of our determination of the application.

Yours sincerely

Rob

Rob Green
Nuclear Regulator
Nuclear Regulation Group (South)

Environment Agency

✉ Red Kite House, Howbery Park,
Crowmarsh Gifford, Wallingford, OX10 8BD

[redacted]

Contact/Line Manager: Phil Heaton, Team Leader NRG South

Environment Agency

The Company Secretary
Augean South Limited
4 Rudgate Court
Walton
Wetherby
West Yorkshire
LS23 7BF

Our ref: EPR/FB3598DD/V001
Your ref:

Date: 20 November 2015

Dear Sir or Madam,

Request for further information to support your application

Application reference: EPR/FB3598DD/V001
Operator: Augean South Limited
Facility: East Northants. Resource Management Facility

We have undertaken a review of the Environmental Safety Case that was submitted in support of the above application. There are a number of points that we require further clarification on, as well as some areas where we feel that further work and/or justification is needed. We need this information so that we may continue to process your application.

I enclose a notice that asks you to send us more information to support your application. The notice specifies what you need to send and when you must send it by.

If we do not receive it by the date set out in the notice then we can treat your application as having been withdrawn.

If you have any concerns about being able to provide this information on time please let me know.

If you have any questions about this notice please phone me on 01491 828463 or email [redacted]

Yours faithfully

[redacted]

Rob Green
Nuclear Regulator

cc. Dr Gene Wilson, East Northants Resource Management Facility, Stamford Road, Kings Cliffe, Northamptonshire PE8 6XX

Environment Agency

Notice requiring further information

To: The Company Secretary

Augean South Limited
4 Rudgate Court
Walton
Wetherby
West Yorkshire
LS23 7BF

Application number: EPR/FB3598DD/V001

The Environment Agency, in exercise of its powers under **paragraph 4 of Part 1 of Schedule 5** of the above Regulations, requires you to provide the information detailed in the attached schedule. The information is required in order to determine your permit application dated **24 July 2015**. The information requested should be sent to the following address by **7 December 2015**.

Information should be sent to:

Environment Agency
Nuclear Regulation Group
Red Kite House
Howbery Park
Wallingford
OX10 8BD

Name	Date
Rob Green	20 November 2015

Authorised on behalf of the Environment Agency

Schedule

1. The information requested in supplementary letter 'Annexe 1' that accompanies this notice

Annexe 1 – summary of further information requested from Augean South Limited in support of permit application EPR/FB3598DD/V001

1) A number of smaller intrusion scenarios have been considered within the Environmental Safety Case provided with the above application.

   However, we note that a scenario for a larger intrusion involving substantial amounts of waste being displaced, for example: to make way for a major road, has been excluded from the assessment. This query originates from the fact that a large intrusion was considered in the 2009 ESC and that it was limiting for some radionuclides.

   We ask Augean to provide justification as to why a large intrusion scenario has not been considered in the Environmental Safety Case;

2) We note that Appendix F in the Environmental Safety Case includes information relating to interim concentrations for radionuclides in groundwater below the landfill. However, this information is partial and it is unclear to us whether these concentrations have been used in the assessment or are purely indicative.

   Interim concentrations of radionuclides in groundwater that have been assumed for the "well constructed at the site boundary" scenario do not appear to have been explicitly provided in the Environmental Safety Case.

   In order to provide further clarification and confidence in the assessment **we ask Augean to provide us with time trend graphs showing concentrations per unit disposal at close to the well abstraction point for the following radionuclides: H3, C-14, C-36, Tc-99, Sn-126, I-129, Ra-226**, Ra-226***, U-234, U-235, U-238, Np-237, Pu-240 and Pu-242.**

3) The Environmental Safety Case does not consider infant and child doses but rather provides an analysis that additional calculations are not necessary as previous assessments from the 2009 Environmental Safety Case show that doses to children and infants (plus the foetus/embryo) will be lower than adult doses for the majority of radionuclides (paras 430-431). We note, however, from the 2009 Environmental Safety Case that, in the case of consumption of foodstuffs contaminated with Cl-36, the child/infant doses in the were higher than adult doses by a factor of 2.5 (2009 application document Annex B table C5 – note that child/infant doses for all other radionuclides are lower than corresponding adult doses).

   We have considered the results for Cl-36 for the smallholder scenario in the (2015) Environmental Safety Case submitted in support of the variation application in light of the above:

   The dose from the maximum inventory of Cl-36 to a smallholder at 200 y is 0.08 mSv/y (table 18). A factor of 2.5 increase in this dose would be well below the dose guidance level. Similarly, for the leachate treatment scenario, peak impacts to the farming family from Cl-36 are 0.005 uSv/y. However, the limiting capacity for Cl-36 is associated with the well scenario – in these calculations the irrigation pathway (which includes consumption of foodstuffs contaminated with contaminated groundwater) contributes

1

about 80% of dose compared with 20% for the water drinking pathway for Cl-36 (Tables 78, 79).

We take view that the Cl-36 radiological capacity may not be conservative for infants and children. We therefore ask Augean to formally assess impacts to child/infant receptors for Cl-36 smallholder scenario and revise the Cl-36 radiological capacity accordingly. The outcome of this assessment should be provided to us.

4) Doses for leachate treatment scenarios are significantly lower (factor ~6) in the final 2015 Environmental Safety Case compared with the draft. The original model assumed 2400 m3/y based on GoldSim groundwater model output. The revised model assumes 403 m3/y based on 28 m3 monthly loads with an allowance of 20% for peak rainfall (para 609). We understand that the revised calculations in the Environmental Safety Case submitted with the application use a *'realistic volume of leachate'* but the data used to support these calculations is unclear to us.

   We ask Augean to provide justification for the reduction in values for the annual quantities of leachate that have been used in the revised model. We ask also that Augean explains the source of the 28m3 figure that has been used for monthly loads of leachate removed from site.

5) The reduced leachate volume suggested in point 4 is also of relevance for Co-60, as the dose for Co-60 at the leachate treatment facility (operational period) is reduced from 584 uSv/y in the draft Environmental Safety Case to 86 uSv/y in the final Environmental Safety Case. Despite this potentially high dose, Augean does not propose to use this scenario to limit the Co-60 capacity because (a) Co-60 accounts for only about 6% of the LLW activity in the UK national inventory, (b) the estimate assumes that all leachate is sent for treatment, which will not be the case and (c) the model is conservative because it does not take into account sorption within waste materials.

   While we acknowledge these reasons, we would like Augean to provide additional justification as to why the leachate treatment facility (operational period) scenario has not been used to limit Co-60, in the context of the historic and planned use of the Avonmouth facility for treatment and disposal of leachate from the ENRMF.

6) Bathtubbing impacts are calculated in the final Environmental Safety Case at 450 y after closure (*'the point in time the groundwater model suggests overtopping will occur'*, para 198). We note that in the draft Environmental Safety Case these impacts were calculated at 350 y after closure. We note also that modelling in the final Environmental Safety Case suggests that overtopping **will** occur, whereas the assumption in the draft Environmental Safety Case was that overtopping is uncertain to occur.

   We ask Augean to explain these changes to the bathtubbing scenario impacts.

7) We have noted a number of changes to the calculation parameters used in the final Environmental Safety Case, in comparison to the values used in the draft:

   - Infiltration rate to grassland is reduced by factor of ~5 in the final Environmental Safety Case (Table 43).

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- The waste porosity is reduced from 0.5 to 0.1 in the final Environmental Safety Case (Table 46).
- The inhalation and irradiation dose coefficients for Ra-226, when Pb-210 is modelled explicitly, are different. The former has increased by factor of 3 (Table 54). We note that data from the 2011 Low Level Waste Repository Environmental Safety Case data were used in the draft Environmental Safety Case but ICRP data have been used in the final version.

We ask that Augean provides an explanation for these changes.

8) The Environmental Safety Case identifies the main contributors to dose from leachate migration via groundwater are likely to be H-3, Cl-36, Sr-90, I-129 and Pb-210 (para 171). The updated ESC goes on to say that '*groundwater monitoring for these radionuclides and comparison against background levels in groundwater ... would provide an indication of releases into the environment through this pathway*'.

   We note that Only H-3 and Pb-210 appear to be in the current groundwater analytical suite. **We ask Augean to review their current monitoring programme for groundwater, in light of the above and taking into account the predicted groundwater concentrations in point 2 above, and to provide us with the outcome of this review.**

9) The sensitivity analysis for the smallholder potentially exposed group considers doses at earlier and later times than 200 y (which is the default time used for capacity determination). At times less than 90 y this scenario becomes limiting for Sr-90 (para 1101). At these times the radiological capacity for Sr-90 becomes less than the proposed 89.6 TBq capacity limit for the ENRMF. The current limiting scenario for Sr-90 is the smallholder at 200 y scenario but the proposed capacity of this radionuclide is limited by tonnage and not by dose (tables 24 and 25)

   We note that 60y is the assumed period of authorisation for the ENRMF site and so it seems reasonable that the Sr-90 capacity is brought in line with the smallholder intrusion scenario impacts (at 60y).

   We ask Augean to reassess the Sr-90 capacity based on peak smallholder impacts at 60y and to provide us with the outcome of this assessment.

10) We note that radiological capacities for a number of other radionuclides are also based on the smallholder scenario at 200 y and this scenario provides the limiting capacity for Th-230, Th-232 and Pa-231. Since we cannot necessarily rely on any controls on land use after the assued 60 y period of authorisation we consider it reasonable for the capacities for Th-230, Th-232 and Pa-231 to be brought in line with their respective peak doses calculated throughout the assessment period (periods between 60 and 20,000 y, as opposed to the dose at 200 y.

    We ask Augean to reassess the capacities for Th-230, th-232 and Pa-231 based on their respective peak doses calculated over the assessment period and to provide us with the outcome of this assessment.

    The final Environmental Safety Case includes an assessment of dose to rabbits burrowing into the waste materials. This assessment of this scenario indicates potential doses of > 40 uGy/h to the rabbits at 60 y (max 210 uGy/h). This is a limiting scenario for Pa-231, Cm-243 and Cm-244. The Environmental Safety Case proposes that radiological capacities for these radionuclides could be set for this scenario using the 40 uGy/h action level - resulting in radiological capacity reduction factors of 4, 6 ad 3 for Pa-231, Cm-243 and Cm-244 respectively (table 141).

We note that rabbits are not a protected species and our assessment does not indicate any significant adverse impact on a European site, Site of Special Scientific Interest, Area of Outstanding Natural Beauty or other conservation site.

However, limiting the capacity of the radionuclides that lead to doses in excess of 40 uGy/h for other burrowing species that are protected, such as badgers, should be precautionary, given that these radionuclides are very unlikely to form a significant part of the ENRMF inventory, and that ecological protection is considered at the population level as opposed to the individual level.

We ask Augean to consider reducing the capacity of Pa-231, Cm-243 and Cm-244, based on the outcome of the assessment of the potential doses to non-human species that may burrow into the waste in future.

We would prefer to receive Augean's response to the above requests as a short addendum to the Environmental Safety Case. This should include an updated summary table clarifying the limiting capacities and scenarios for each radionuclide.

Annex B. Consideration of radionuclides with half-lives less than a year

Aim

The EA (Rob Green, 8/10/2015) requested clarification of the definition of 'other radionuclides' proposed in the sum of fractions approach in the ESC presented in July 2015 (Eden Nuclear and Environment Ltd, 2015). The ESC was submitted in support of an application for a variation to the existing (2011) Permit for the ENRMF landfill site. The EA comment that they do not feel that sufficient evidence or reasoning has been presented in the ESC as to why the definition for 'other radionuclides' should include those radionuclides that are not otherwise listed in the permit and have half-lives of greater than a year, as opposed to greater than 3 months.

The radiological assessments supporting the 2015 ESC considered radionuclides with a half-life of greater than 1 year. This note presents assessments for short-lived radionuclides that support the use of a half-life cut-off of 1 year. It provides arguments why a half-life cut-off of 1 year in the sum of fractions approach is appropriate and proportionate at the ENRMF.

Background

Radionuclides with a short half-life are unlikely to lead to exposure of the public before radioactive decay reduces the inventory to insignificant levels. The EA have suggested a 3 month half-life as the cut-off that should apply. For example in the Lillyhall permit:

> *Additionally, we limited the need to consider other radionuclides to those with half-lives greater than three months, on the basis that radionuclides with half-lives less than three months will have substantially decayed before they could cause any dose impact to the public.*

The Lillyhall ESC assessments included radionuclides with half-lives between 3 months and 1 year.

ENRMF approach to accepting waste

The maximum specific activity of a consignment of low level radioactive waste (LLW) that can be accepted at the ENRMF is 200 Bq g^{-1} under the site development consent order (DCO) and the disposal restriction of 448,000 t of LLW therefore limits the maximum site inventory (to 89.6 TBq). These limits are included in the 2015 ESC.

The current Permit lists the radionuclides that can be accepted for disposal and gives disposal limit totals for each radionuclide. It also includes a limit for "Any other radionuclide" and this limit is based on the most limiting radionuclide listed, which is Pu-239 in the current permit. No half-life cut-off is specified.

The 2015 ESC proposes a 'sum of fractions' approach to limiting the radionuclides that can be accepted for disposal. As part of this approach it includes a radiological capacity for 'Any other radionuclide' and this is based on the most limiting radionuclide which is I-129 in the 2015 ESC. Paragraph 307 of the 2015 ESC defines "Any other radionuclide" as those with a half-life greater than 1 year.

The "Conditions for Acceptance" (CFA) used at the ENRMF state:

> *In the "others" category, radionuclides of less than one year half-life are not normally included. If such nuclides are present in significant quantities (>5 MBq/tonne or a high percentage relative to the overall activity content) this shall be notified to ERNMF for acceptance.*

Thus, the CFA provides ENRMF with knowledge of any short half-life activity between 5 Bq g^{-1} and the site limit of 200 Bq g^{-1}, this notification requirement is also included in the 2015 ESC (Paragraph 323).

Radiological capacity for any other radionuclide

It would not be appropriate or proportionate to use the radiological capacity of the "Any other radionuclide" category set using I-129 (long-lived and mobile) for less mobile short half-life radionuclides in the 'sum of fractions' approach since they will have a significantly larger radiological capacity. This significantly larger radiological capacity, and the correspondingly small contribution that short-lived radionuclides would make to the 'sum of fractions', is discussed below drawing on other related assessments and on specific calculations.

Related assessments

Radionuclides with half-lives of 1 year to 6 years

The ESC assessed 8 radionuclides with half-lives of between 1 and about 6 years (Ru-106, Cs-134, Pm-147, Fe-55, Sb-125, Eu-155, Co-60 and Ra-228). The radionuclide with the shortest half-life was Ru-106 with a half-life of 1.02 years.

The radiological capacity of each of these is shown in Table 12 with the associated limiting scenario. These radiological capacities are all significantly higher than the I-129 radiological capacity of 4.17 10^{-2} TBq which is used for "Any other radionuclide". Thus, using the 'Any other radionuclide' radiological capacity would not be appropriate for these radionuclides.

Table 12 Radionuclides with half-lives 1 to 6 years considered in the ESC

Radionuclide	Radiological Capacity (TBq)	Scenario	Kd (m^3 kg^{-1})
Fe-55	5.83 10^{13}	Excavator (Borehole) 60 years	0.22
Co-60	3.83 10^{5}	Excavator (Borehole) 60 years	0.06
Ru-106	5.29 10^{16}	Recreational (0 years)	0.055
Sb-125	3.33 10^{9}	Excavator (Borehole) 60 years	0.045
Cs-134	1.37 10^{11}	Excavator (Borehole) 60 years	0.27
Pm-147	4.74 10^{13}	Excavator (Borehole) 60 years	0.24
Eu-155	7.91 10^{7}	Excavator (Borehole) 60 years	0.24
Ra-228	1.80 10^{5}	Excavator (Borehole) 60 years	0.49

The radiological capacities calculated for these radionuclides can be used to give an indication of the contribution of short lived radionuclides to the 'sum of fractions'. The ESC assessed radionuclides with half-lives between 1 and about 6 years are not limited by the groundwater pathways. In each case the radiological capacity is very large, well above the maximum inventory (89.6 TBq – see Section 6 of the 2015 ESC) that can be disposed of at the site under the DCO. The most restrictive radiological capacity for these 8 radionuclides applies to Ra-228, and is a factor of 2000 greater than the maximum inventory (1.8×10^5 TBq/89.6 TBq). Thus the maximum contribution of Ra-228 disposals to a sum of fractions would come from disposal of 89.6 TBq Ra-228, and the fraction would be 0.0005 (89.6 TBq/1.8×10^5 TBq). The corresponding dose from each exposure scenario would be a factor of 2000 or more below the dose criterion. The contribution of the other radionuclides to the sum of fractions, even from disposal of the maximum inventory, would be smaller than 0.0005 since their radiological capacities are even larger.

These very low doses and very low contributions to the sum of fractions could in fact support a case for not accounting for radionuclides with half-lives between 1 and about 6 years in the sum of fractions; however, this is not proposed here. We will consider them in the sum of fractions and in assessments against the 200 Bq g^{-1} activity limit, as described in the ESC.

Half-lives less than 1 year

It is expected that radionuclides with shorter half-lives will have larger radiological capacities than the values given in Table 12. This is confirmed by looking at the ratio of radiological capacities obtained in the Lillyhall ESC for radionuclides with half-lives between 3 months and 1 year. The ratio to the capacity for Co-60 is presented in Table 13.

Table 13 Relative radiological capacities for Lillyhall

Radionuclide	Ratio of Lillyhall relevant value for the radionuclide to Lillyhall relevant value of Co-60
Ag-110m	4.19
Ce-144	1.97×10^3
Cm-242	1.29×10^4
Mn-54	1.16×10^1
Po-210	9.35
Zn-65	1.58×10^1

Hence, all these radionuclides would be expected to have radiological capacities greater than that for Co-60; and since the ratio for Ra-228 in the ESC is 2.13, they would also be expected to have radiological capacities greater than that for Ra-228. Hence they would contribute even less to the sum of fractions calculations and use of the radiological capacity for 'other radionuclides' would not be appropriate.

Explicit consideration of half-lives 3 months to 1 year

Radionuclides considered

ICRP (ICRP, 2012) list 40 radionuclides with a half-life of between 3 months and 1 year, 10 of which appear in the UK national waste inventory (Nuclear Decommissioning Authority, 2014) and might therefore appear in waste streams accepted for disposal (see Table 14). This group of 10 radionuclides was therefore selected for further consideration.

Table 14 Radionuclides with half-lives between 3 months and 1 year

Radionuclide	Half-life (y)	UK Waste Inventory (MBq)	Kd ($m^3 kg^{-1}$)	Ingestion dose coefficient ($Sv Bq^{-1}$)
Ag-110m	0.684	2.9×10^3	0.09	2.80×10^{-9}
Ce-144	0.778	1.3×10^5	0.49	5.20×10^{-9}
Cm-242	0.446	1.6×10^3	4	1.20×10^{-8}
Gd-153	0.663	2.1×10^{-10}	0.24[1]	2.70×10^{-10}
Mn-54	0.856	5.6×10^4	0.049	7.10×10^{-10}
Po-210	0.379	1.6×10^4	0.15	1.20×10^{-6}
Sn-119m	0.802	1.4×10^{-7}	0.13	3.40×10^{-10}
Sn-123	0.354	3.5×10^{-8}	0.13	2.10×10^{-9}
Te-127m	0.298	2.2×10^1	0.13	2.30×10^{-9}
Zn-65	0.668	5.9×10^3	0.2	3.90×10^{-9}

1. The Kd for Gd-153 is based on the assumption that it has properties similar to Europium.

The national inventory of any of these radionuclides is well below (by at least a factor of 689) the maximum site inventory of 89.6 TBq. Thus the contribution to the sum of fractions would be less than 0.0015 (1/689) on the basis of the inventory alone. Furthermore, the results for the 1 to 6 year half-life radionuclides given above, and the fact that at Lillyhall shorter lived radionuclides would have larger radiological capacities than these radionuclides, suggest that the actual contribution to the sum of fractions would be much lower when radioactive decay in the different exposure scenarios was taken into account (e.g. leading to a sum of fractions contribution of 1/689*0.0005 or less). This is explored further below.

Note that the CFA requirement to notify the presence of short-lived radionuclides at >5 Bq g^{-1} could mean that in 448,000 tonnes of waste a maximum of 2.24×10^6 MBq could be un-notified, a value that is at least a factor of 10 greater than the national inventory of any of these radionuclides, and substantially greater for many of these radionuclides. This is also addressed below.

Dose results for short-lived radionuclides

The ESC explicitly considers 8 radiological assessment scenarios for the period of authorisation (see Section E3 of the ESC) and 4 scenarios relating to the period after authorisation (see Sections E4 and E5). These scenarios are listed in Table 14. The bathtubbing scenario is assumed to occur 450 years after closure and therefore the short-half radionuclides will have decayed to insignificant levels by this time: hence this scenario was not considered further. A similar argument was made for the two human intrusion scenarios which are assumed to occur 150 or 200 years after closure, respectively. Although a similar argument could also be made for the recreational scenario and borehole scenario since they occur 60 years after closure of the site, they were explicitly considered just to illustrate the level of dose. Furthermore, the recreational scenario was conservatively assumed to occur immediately at site closure rather than 60 years after closure.

The recreational use scenario, borehole scenario and the scenarios relating to the period of authorisation were explicitly considered for the 10 short half-life radionuclides listed in Table 14 and a summary of the dose results is given in Table 15. The values are based on disposal of the entire UK inventory of each of these short lived radionuclides in the ENRMF. Further details of the calculations and arguments supporting these results are given in Appendix A.

It can be seen from Table 15 that the dose results for these radionuclides are extremely low, even assuming that the entire inventory of these radionuclides is disposed of at the ENRMF. Since, this is extremely unlikely, these radionuclides with half-lives of 3 months to a year will make an insignificant contribution to the overall dose from the LLW disposed of at the site.

Table 15 Dose results for radionuclides with half-lives between 3 months and 1 year

Scenario	Exposed group	Results for radionuclides with a half-life <1 year
Period of Authorisation – expected to occur		
Direct exposure (external exposure)	Worker	Limited by dose rate criterion not radiological capacity
	Member of public	
Leachate processing off-site	Treatment worker	Dose from UK inventory <0.015 µSv y^{-1}
	Farming family/Angler	
Release to atmosphere	Member of public	Not considered as will not be released in a gaseous form
Release to groundwater	Member of public	Dose from UK inventory <1 x 10^{-10} µSv y^{-1}
Period of Authorisation – not certain to occur		
Leachate spillage	Farming family	Dose from UK inventory <0.006 µSv
Dropped load	Worker	Highly conservative assumptions give dose 1.2 x 10^{-3} µSv or less
Aircraft impact	Member of public	Doses from impact disturbing UK inventory <10^{-2} mSv
Wound exposure	Worker	Unlikely to exceed criterion based on the arguments used for all other radionuclides (i.e. low inventory)
After the period of Authorisation		
Recreational use	Member of public	Dose at time of closure from UK inventory <1.7 x 10^{-15} µSv y^{-1}
Groundwater abstraction	Member of public	Dose at 60 y from UK inventory <1.7 x 10^{-28} µSv y^{-1}
Bathtubbing	Member of public	Modelled to occur 450 years post closure by which time short half-life radionuclides will have decayed to insignificant levels.
Human intrusion		Earliest time of intrusion is 60, 150 or 200 years post closure by which time short half-life radionuclides will have decayed to insignificant levels. Dose from UK inventory of Cm-242 (inc ingrowth) <3.4 x 10^{-3} µSv y^{-1}

Radiological capacity

The direct exposure, leachate processing, leachate spillage, dropped load, aircraft impact, bathtubbing and wound exposure scenarios are not relevant to the calculation of radionuclide capacity since they consider individual loads and/or fractions of the inventory in the site. The scenarios that determine the radiological capacity are release to groundwater, recreational use, groundwater abstraction and human intrusion.

The calculations indicate that the radiological capacity determined from the recreational scenario occurring immediately after closure would be 7×10^{19} MBq or greater, and that for the groundwater scenarios would be 2×10^{16} MBq or greater. The dose from disposal of the maximum inventory for each of the short lived radionuclides would be less than 1×10^{-10} $\mu Sv\ y^{-1}$ and the maximum contribution to the sum of fractions is 6×10^{-12}. For the borehole scenario, the radiological capacities for all radionuclides except Cm-242 are greater than 1×10^{29} MBq, and are greater than those for the groundwater scenario. The radiological capacity for Cm-242 (including ingrowth of Pu-238) is 5×10^{8} MBq but since the national inventory is 2 MBq, the contribution to the sum of fractions would be negligible.

Hence it is appropriate and proportionate to not consider these short lived radionuclides in the sum of fractions calculations.

Conclusion

Radioactive decay reduces the inventory of short half-life radionuclides to insignificant levels by the end of the period of authorisation. The doses estimated for the short half-life radionuclides during the period of authorisation are all significantly below the relevant dose criteria, even assuming that the entire UK inventory of that radionuclide is disposed of at the ENRMF. The very low doses from the entire UK inventory mean that the radiological capacity for these radionuclides is very large and hence these radionuclides would make a negligible contribution to the sum of fractions i.e. including them would have an insignificant impact as the sum of fractions would not change. Hence, the main regulatory mechanism is not sensitive to the consideration of short half-life radionuclides. In reality the sum of fractions is also not sensitive to the inclusion or omission of radionuclides with a half-life of between 1 and 6 years in the sum of fractions calculations, though these radionuclides will continue to be accounted for and explicitly included in the sum of fractions.

The assessments discussed here show that the proposed 1 year cut-off for “Any other radionuclides” in the sum of fractions calculations does not have any impact on the dose to members of the public. In addition, the dose rate criterion protects workers and members of the public when the LLW is received and disposed of, and applies irrespective of the half-life of the radionuclides in the waste.

These results provide a strong case for not including radionuclides with half-lives less than 1 year in the sum of fractions calculations. The low national waste inventory of the short half-life radionuclides suggests that they will not be presented for disposal at the ENRMF in any great amount and the precaution of reporting them if they are present in the LLW at greater than 5 Bq g^{-1} allows their occurrence to be monitored in a proportionate manner.

Appendix A to Annex B

Introduction

Calculations were performed for each radionuclide and then scaled for the UK inventory of that radionuclide, or the hypothetical maximum 'unreported inventory' of 2.24 x 10^6 MBq. The Cm-242 dose conversion factors used in the calculations assumed complete ingrowth of Pu-238.

Direct exposure

The dose to workers from waste handling operations was assessed by HPA (now PHE) and reported in the 2015 ESC (Appendix H). The doses to workers or to a member of the public at the site fence are constrained by the limit that is applied to the external dose rate from the LLW package (10 µSv h^{-1}) which applies to a waste package irrespective of the half-life. Hence, the radiological capacity is not relevant to this scenario.

Leachate processing off-site

The EA initial assessment methodology considers Ag-110m, Ce-144, Cm-242, Mn-54, Po-210 and Zn-65. The results for the short-lived radionuclides follow the same pattern observed for those with a half-life greater than 1 year, giving very low doses to a Farming family (adult) or Fisherman (adult) and slightly larger doses to the sewage treatment facility worker.

The highest doses arise from processing leachate containing Ag-110m or Mn-54, and disposal of the complete UK inventory would produce a dose of about 0.003 µSv y^{-1} and 0.015 µSv y^{-1}, respectively.

Disposal of a hypothetical maximum unreported inventory of 2.24 x 10^6 MBq would result in a dose to the sewage treatment worker of about 1.8 µSv y^{-1} for Ag-110m and 0.6 µSv y^{-1} for Mn-54, and doses to the farming family or fisherman would be <0.005 µSv y^{-1}. Note that this hypothetical inventory is a factor of 700 and 40 higher than the national inventory of these radionuclides, respectively.

The short half-life radionuclides will therefore have no impact on doses from the leachate treatment scenario.

Release to groundwater

The activity concentration at the groundwater extraction point depends on the inventory, the dilution in the groundwater and the time taken for the radionuclide to travel to that point. For radionuclides with very short half-lives, a significant amount of radioactive decay will have occurred during that time. For a given inventory, water flow rate and Kd value, the shorter the half-life, the earlier and smaller the peak activity concentration at a given extraction point. The peak activity concentration at the site boundary for H-3 occurs at 40 years, that for Sr-90 at 138 years and that for I-129 at 2100 years.

The transit time from a waste cell to a well at the site boundary or the existing extraction point is influenced by the radionuclide Kd, the rate at which water moves from the waste cell to the water table beneath the site and the distance to the extraction point through groundwater flow. If the transit time from the base of the landfill to the extraction point is long compared with the half-life then any radioactivity entering groundwater will decay before it reaches a member of the public.

There is Kd information for all short half-life radionuclides except Gd-153, for which Europium was used, and these values are given in Table 14.

The Kd values in Table 14 are all greater than that for Sr-90 (0.013) and hence the 10 short lived radionuclides would therefore be expected to travel slower over the groundwater pathway than Sr-90, leading to longer transit times to the site boundary. Hence Sr-90 transport can be used as a bounding surrogate for these 10 short-lived radionuclides.

GoldSim output shows that the Sr-90 concentration in groundwater at the boundary well increases from breakthrough at about 10 years, peaking at 138 years. At 25 years, the concentration is greater than that at 10 years but remains a small fraction (0.14) of the peak concentration. With a half-life of 1 year there would be 0.1% of an initial inventory remaining after 10 years and the activity concentration would be expected to decrease at longer times, rather than increase.

A scoping calculation was performed based on the assumption that the peak activity concentration for the short lived radionuclides occurred at the breakthrough time (10 years) and that the breakthrough curve followed the same shape as Sr-90 but was modified by radioactive decay. It was also conservatively assumed that the activity concentration at 10 years for Sr-90 was 0.14 of the peak activity concentration, whereas it would be lower. The doses for the short lived radionuclides were then obtained from the Sr-90 dose by applying radioactive decay and scaling by the ratio of the dose coefficients. The dose coefficients for all the short lived radionuclides except Po-210 are lower than that for Sr-90 ($3.07\ 10^{-8}$ Sv Bq^{-1}) so the dose from a 'short half-life Sr-90' can be used as a surrogate for all except Po-210.

Taking the peak groundwater dose for Sr-90 (1.53×10^{-10} µSv MBq^{-1}), and applying the fraction of the peak dose at 25 years (0.14), the dose from Sr-90 at 25 years is 2.1×10^{-11} µSv MBq^{-1}. Correcting for radioactive decay of Sr-90 at 25 years, the dose from Sr-90 at 25 years (assuming no decay) would be 3.8×10^{-11} µSv MBq^{-1}. As discussed above, the Sr-90 dose at 10 years would be less than this value but this is used here as a conservative estimate. The dose from all the short lived radionuclides would be <0.1% of this value based on radioactive decay alone, i.e. $<3.8 \times 10^{-14}$ µSv MBq^{-1} (the results range from 3.2×10^{-21} µSv MBq^{-1} for Te-127m to 1.2×10^{-14} µSv MBq^{-1} for Mn-54). Scaling by the ratio of the dose coefficients gives the estimated doses given in Table 16.

Table 16 Results for the groundwater calculations for a breakthrough time of 10 years

Radionuclide	UK Waste Inventory (MBq)	Estimated dose per unit disposal ($\mu Sv\ y^{-1}\ MBq^{-1}$)	Estimated capacity from groundwater migration (MBq)	Ratio UK national inventory/capacity
Ag-110m	2.9E+03	1.4E-16	1.4E+17	2E-14
Ce-144	1.3E+05	8.9E-16	2.2E+16	6E-12
Cm-242	1.6E+03	2.7E-18	7.5E+18	2E-16
Gd-153	2.1E-10	9.8E-18	2.0E+18	1E-28
Mn-54	5.6E+04	2.7E-16	7.4E+16	8E-13
Po-210	1.6E+04	1.7E-17	1.2E+18	1E-14
Sn-119m	1.4E-07	7.6E-17	2.6E+17	5E-25
Sn-123	3.5E-08	8.2E-21	2.4E+21	1E-29
Te-127m	2.2E+01	2.4E-22	8.4E+22	3E-22
Zn-65	5.9E+03	1.5E-16	1.3E+17	5E-14

Disposal of the complete UK national waste inventory of Po-210 at the site will result in a dose of 3×10^{-13} $\mu Sv\ y^{-1}$ (with a contribution to the sum of fractions of 1×10^{-14}). The corresponding dose from disposal of the maximum inventory for each of the short lived radionuclides would be less than 1×10^{-10} $\mu Sv\ y^{-1}$ and the maximum contribution to the sum of fractions is 6×10^{-12}. Assuming a breakthrough time of 5 years and the same Sr-90 dose as at 25 years, the maximum contribution to the sum of fractions is 5×10^{-10}.

The contribution of short half-life radionuclides to dose will therefore be negligible and they will have no impact on the sum of fractions for the groundwater pathway capacity.

Leachate spillage

Spilt leachate that enters water resources would become diluted and effective mitigation measures would be difficult to achieve. The assessment of leachate spillage therefore focusses on pathways related to the use of water resources (drinking, irrigation, livestock and angling). The leachate activity concentration used in the calculations is the maximum that could occur based on the same approach used in the GoldSim model and assuming no radioactive decay.

The highest dose arises from spillage of leachate containing Po-210, the dose arising from disposal of the complete UK inventory would be about 0.006 μSv. Disposal of 2.24×10^{6} MBq would result in a dose of about 0.9 μSv, though this is a factor of 140 greater than the national inventory.

The short half-life radionuclides will have no impact through the leachate spillage scenario.

Dropped load and Aircraft impact

Assessments have been undertaken for dropped waste containers and an aircraft crash at the site. The maximum doses arising from a dropped container and an aircraft impact depend on the specific activity of waste (assumed to be 5 $Bq\ g^{-1}$). The dropped load 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. In the case of an aircraft impact 300 m^3 of waste are assumed to be displaced and the dose to a member of the public and a worker is assumed to be the same in the early stages of the response to the accident. 300 m^3 at 5 Bq g^{-1} implies a released inventory that is greater than the national inventory for 5 of the short half-life radionuclides.

The dropped load dose assessment for these radionuclides meets the site criterion for workers for all radionuclides: max is for Cm-242 (+Pu-238) with 1 x 10^{-2} mSv) (Cm-242 without Pu-238 ingrowth gives 5 x 10^{-4} mSv). All doses to the public are significantly below 20 µSv: max for Cm-242 (+Pu-238) is 3.2 x 10^{-2} µSv (value for Po-210 is 1.2 x 10^{-3} µSv). Even at 200 Bq/g (which would never be the case) the doses from the short lived radionuclides would meet the dose criteria.

For these short half-life radionuclides, the largest calculated dose following an aircraft impact on the site (approximately 1.3 x 10^{-2} mSv) arises from inhalation of dust containing Cm-242 (+Pu-238); inhalation of Po-210 gives about 5 x 10^{-4} mSv and the remaining radionuclides give much lower doses. Even at 200 Bq/g (which would never be the case) the doses from the short lived radionuclides would meet the dose criteria. As shown above the national inventory of LLW (Nuclear Decommissioning Authority, 2013) includes relatively small amounts of these radionuclides, and hence the doses from this scenario would be much lower than these values. Assuming that the impacted area contains the complete national inventory, the doses for Cm-242(+Pu-238) would be 1.7 x 10^{-2} mSv and 3.5 x 10^{-3} mSv for Po-210.

The assessment has not taken into account the depth of daily cover, has used a high resuspension factor and assumed that a large proportion of a waste package is very powdery. This calculation is therefore conservative and the complexity of an aircraft crash means that this calculation can only be considered as a scoping calculation. Nevertheless, the scoping calculations indicate that the 3 to 20 mSv dose guidance level for human intrusion events would not be exceeded by this very low probability event.

The short half-life radionuclides will have no impact through the dropped load or aircraft impact scenarios.

Wound exposure

An assessment of exposure resulting from a wound in the ESC (Section E.3.2.3) concluded that internal doses from a contaminated wound would be very unlikely to exceed 1 mSv in practice. The highest dose from incorporation of 0.1g of material at 200 Bq/g was calculated in the ESC to be 3 mSv from Ac-227. As this radionuclide is most unlikely to predominate, it was concluded that internal doses from a contaminated wound would be very unlikely to exceed 1 mSv in practice. The short-lived radionuclide with the highest dose per unit intake by ingestion (which is relevant to incorporation in a wound) is Po-210, with a dose coefficient very similar to Ac-227 (1.2E-6 Sv/Bq instead of 1.21E-6 Sv/Bq). Po-210 is also unlikely to predominate in the waste and therefore internal doses from a contaminated wound from short lived radionuclides would be very unlikely to exceed 1 mSv in practice. The short half-life radionuclides in the LLW will be handled in the same way as other radionuclides in the LLW.

Recreational user

The intended end use of the site includes woodland and grassland with paths and a view point. An assessment is therefore made of the doses to a member of the public who spends time walking over the restored site for about 2 h d^{-1} and receives external exposure from buried waste packages. The results are calculated at the time of closure (assuming no radioactive decay). The results are shown in Table 17.

Table 17 Results for the recreational scenario

Radionuclide	UK Waste Inventory (MBq)	Estimated dose per unit disposal (μSv y^{-1} MBq^{-1})	Estimated capacity from recreational scenario (MBq)	Ratio UK national inventory/capacity
Ag-110m	2.9E+03	2.3E-19	8.7E+19	3E-17
Ce-144	1.3E+05	4.3E-38	4.7E+38	3E-34
Cm-242	1.6E+03	4.4E-45	4.6E+45	3E-43
Gd-153	2.1E-10	1.5E-46	1.3E+47	2E-57
Mn-54	5.6E+04	2.9E-20	6.9E+20	8E-17
Po-210	1.6E+04	1.4E-25	1.4E+26	1E-22
Sn-119m	1.4E-07	0.0E+00	unlimited	0E+00
Sn-123	3.5E-08	6.3E-22	3.2E+22	1E-30
Te-127m	2.2E+01	2.3E-42	8.8E+42	3E-42
Zn-65	5.9E+03	2.8E-19	7.2E+19	8E-17

The doses at the time of closure are all very low, and those after 60 years would be even lower. The highest dose per unit inventory is from Zn-65. Disposal of 2.24 x 10^6 MBq of Zn-65 would result in a dose of about 6 x 10^{-13} µSv y^{-1}. The radiological capacities of the radionuclides are 7 10^{19} MBq or greater. Disposal of the complete UK inventory would produce a dose of about 1.7 x 10^{-15} µSv y^{-1} or less from each radionuclide (highest dose is from Zn-65).

The short half-life radionuclides will have no impact on the dose through the recreational user scenario, nor on the sum of fractions.

Groundwater extraction

The doses from groundwater extraction after the period of authorisation (i.e. after 60 years) will be much lower than those from groundwater during the period of authorisation since radioactive decay will have reduced the activity concentrations. The minimum radiological capacity is 7 x 10^{33} MBq and the maximum dose from the national inventory is 1.7 x 10^{-28} µSv y^{-1}. Hence the doses will be negligible and the contribution to the sum of fractions will be negligible.

Borehole

The doses from the borehole scenario at 60 years are very low. The maximum dose per unit disposal results from Cm-242, due to ingrowth of Pu-238. The dose from the complete UK inventory for

Cm-242 (including ingrowth of Pu-238) is 3.4×10^{-3} μSv y^{-1}, and the associated radiological capacity assuming a 1 mSv dose criterion is 5×10^{2} TBq (5×10^{8} MBq). For all other radionuclides the associated radiological capacity is greater than 1×10^{23} TBq (1×10^{29} MBq) and the dose from the national inventory is less than 3×10^{-22} μSv y^{-1}. Hence the doses will be negligible and the contribution to the sum of fractions will be negligible.

Annex C. Time-dependent concentrations in groundwater

Figure 1. Time-dependent concentrations of H-3 in groundwater

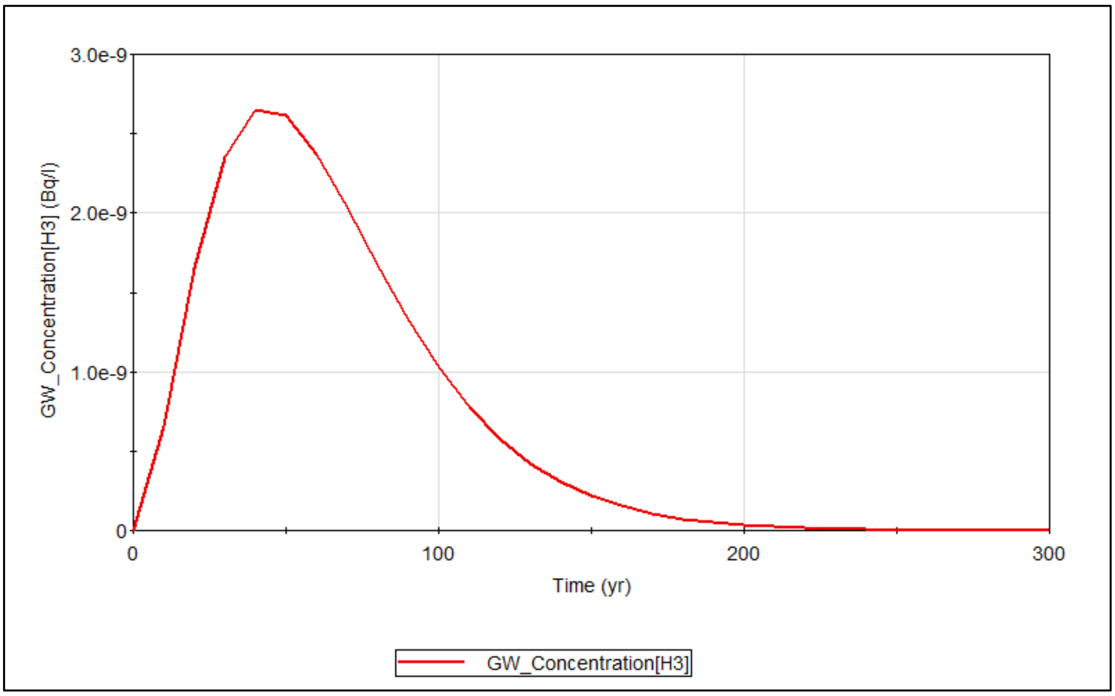


Figure 2. Time-dependent concentrations of C-14 in groundwater

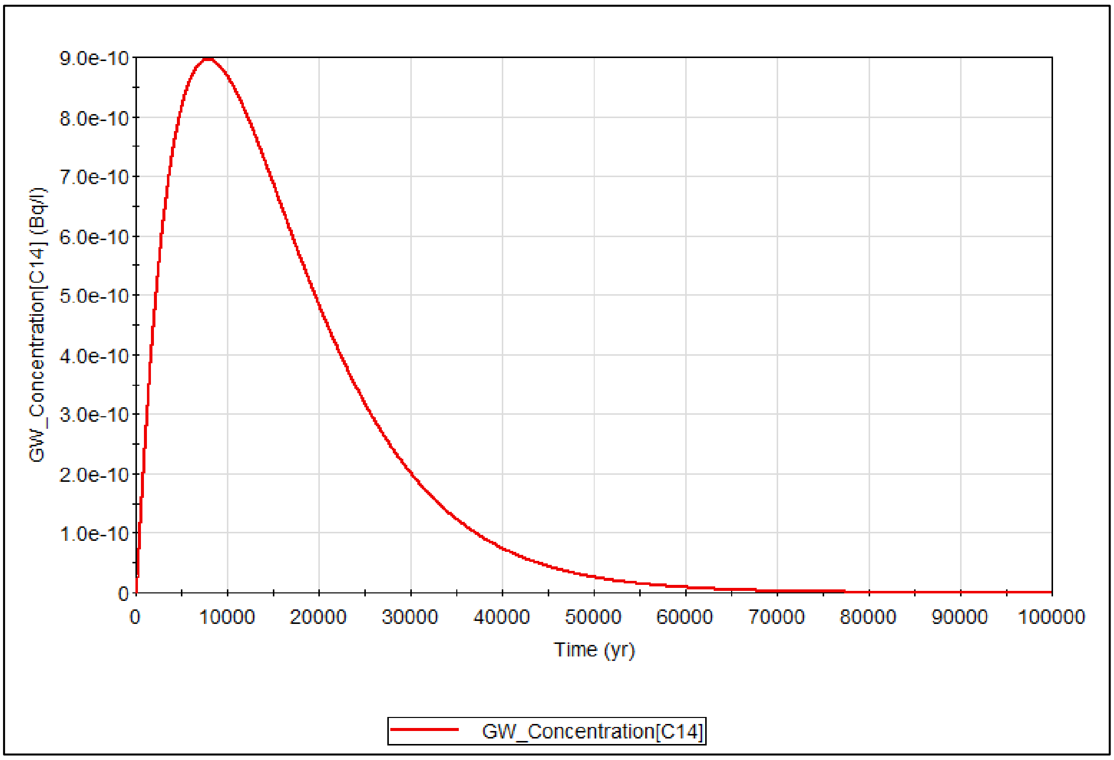


Figure 3. Time-dependent concentrations of Cl-36 in groundwater

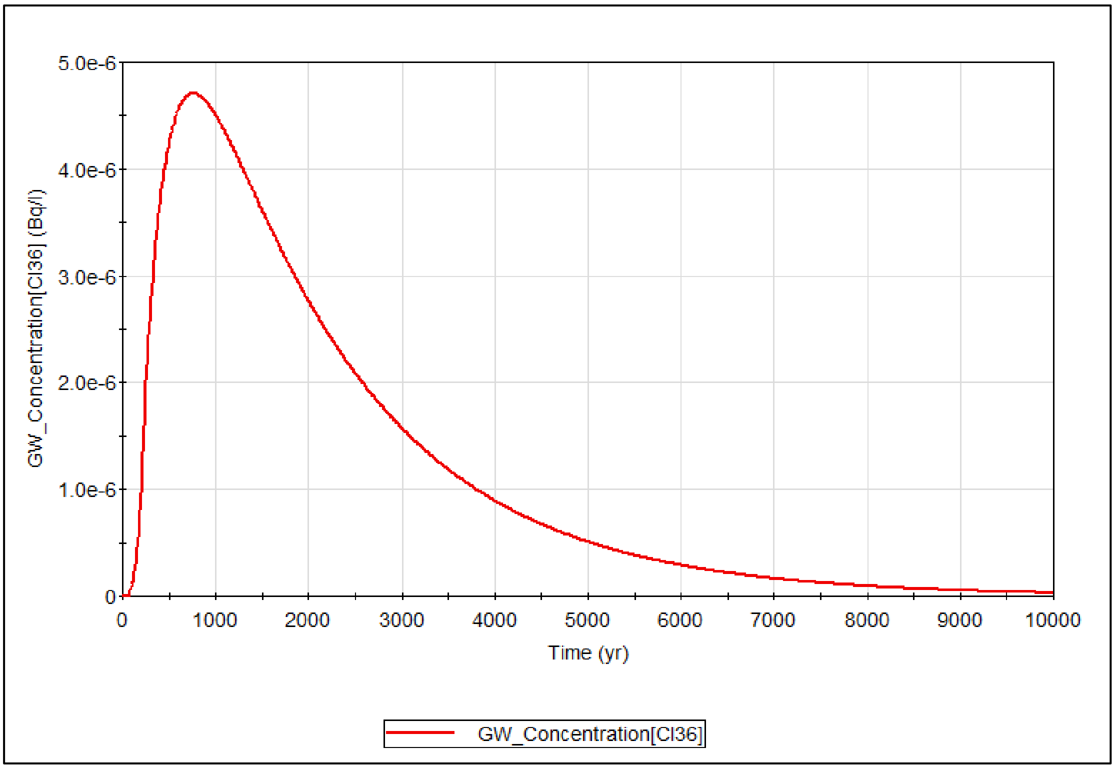


Figure 4. Time-dependent concentrations of Tc-99 in groundwater

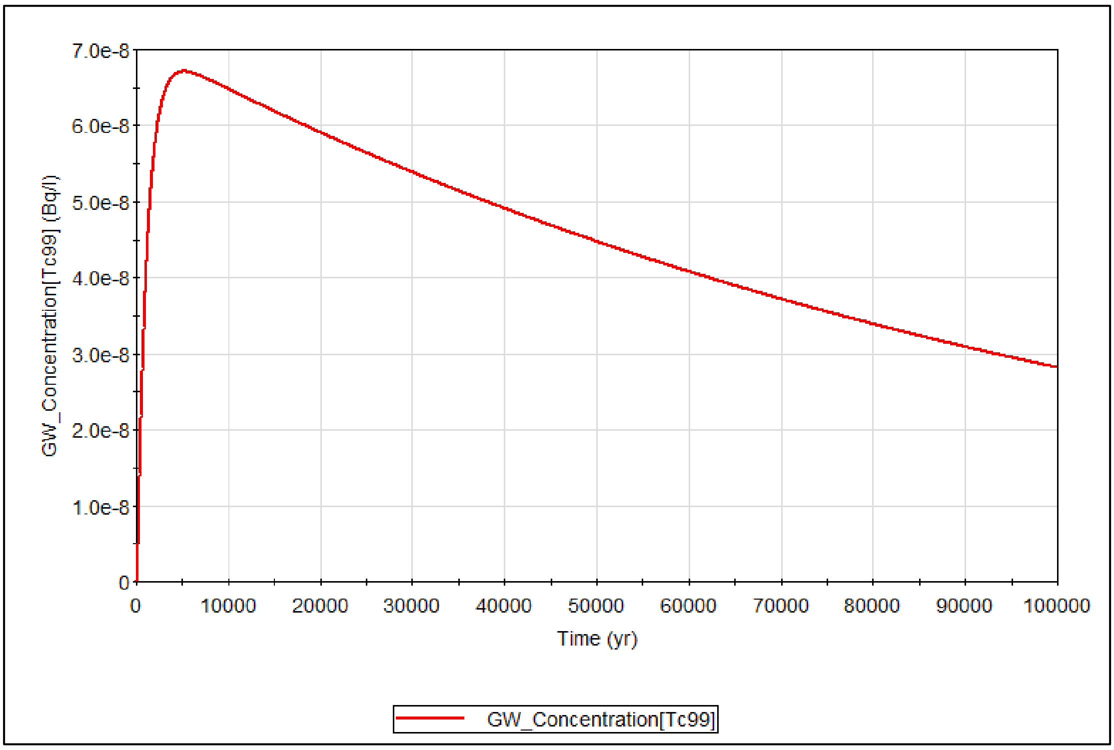


Figure 5. Time-dependent concentrations of Sn-126 in groundwater

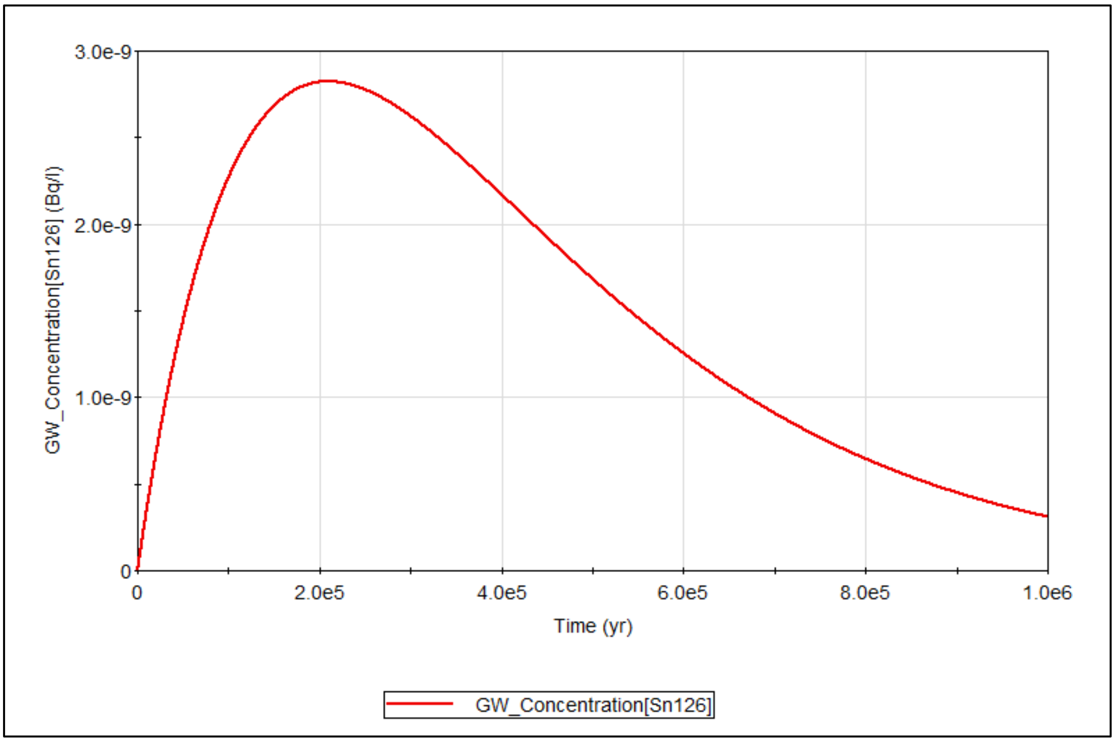


Figure 6. Time-dependent concentrations of I-129 in groundwater

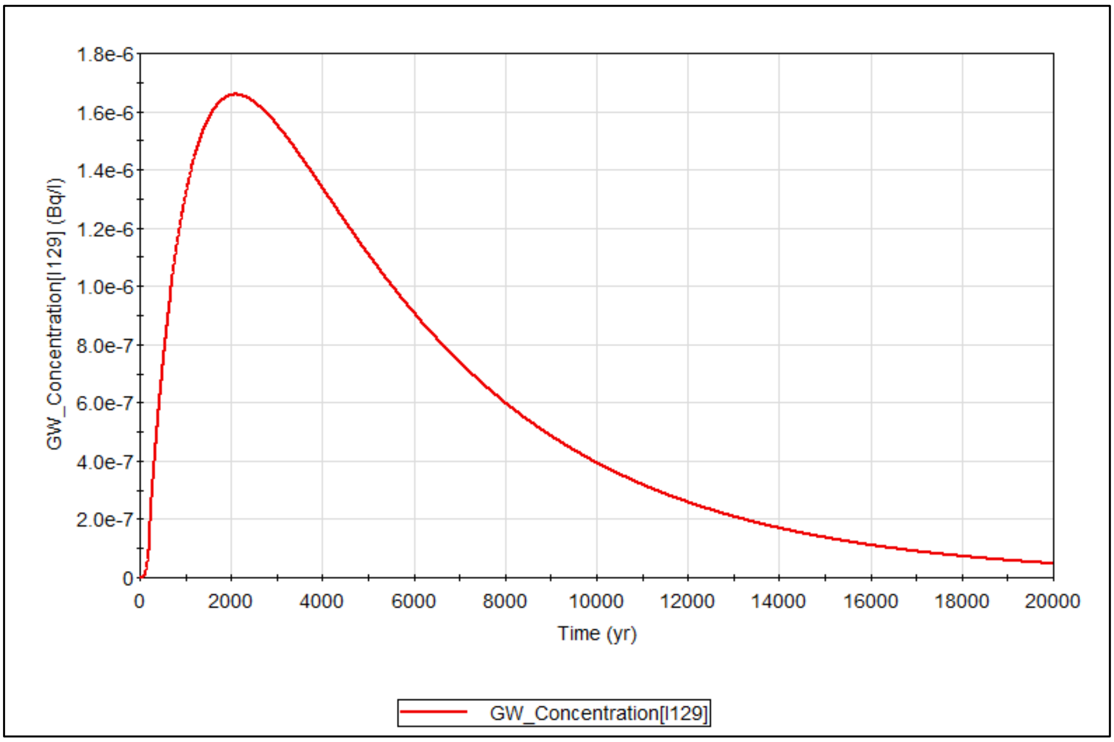


Figure 7. Time-dependent concentrations of Ra-226 in groundwater

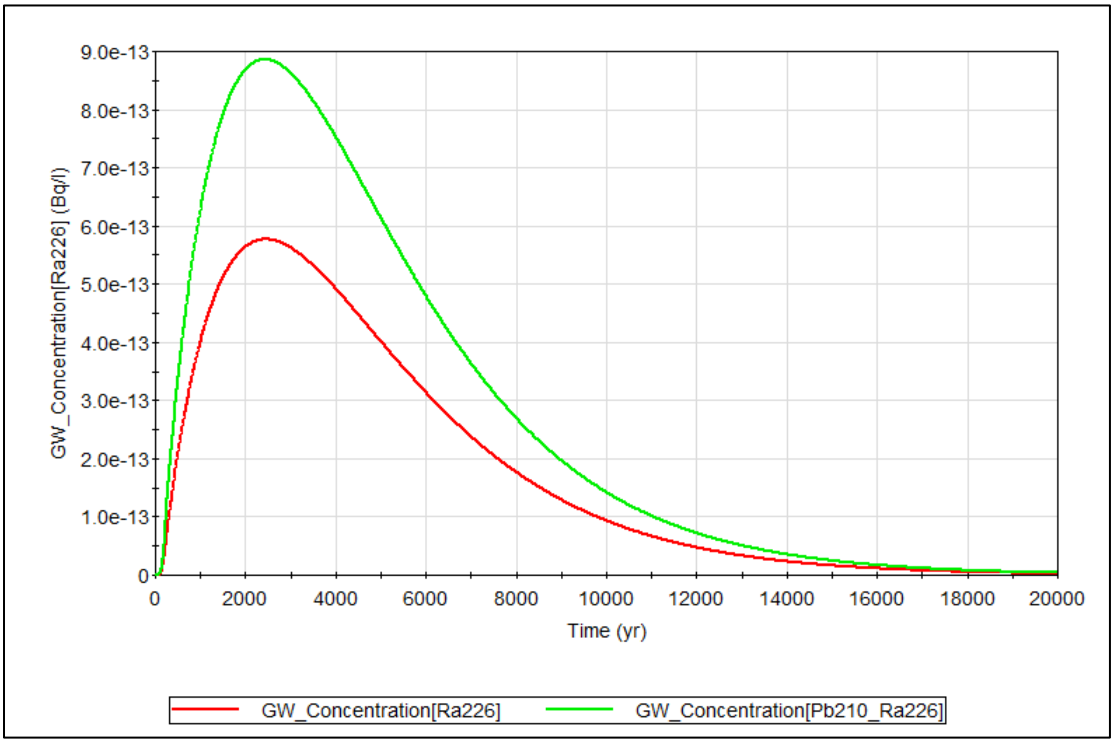


Figure 8. Time-dependent concentrations of U-234 in groundwater

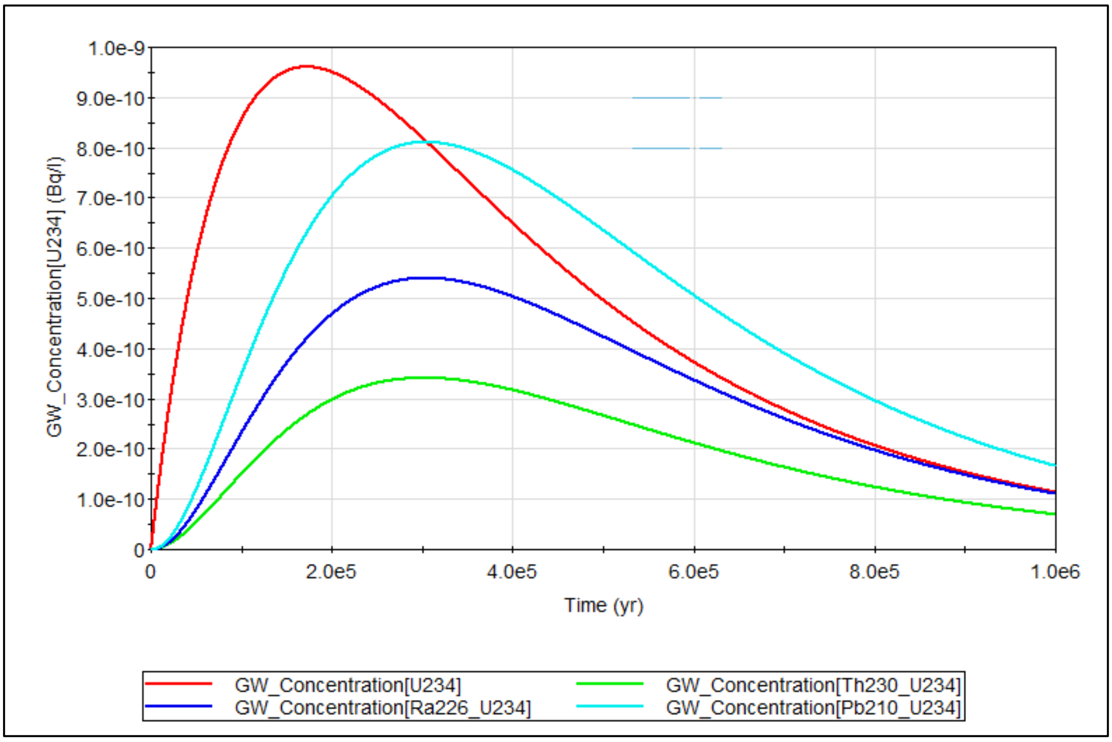


Figure 9. Time-dependent concentrations of U-235 in groundwater

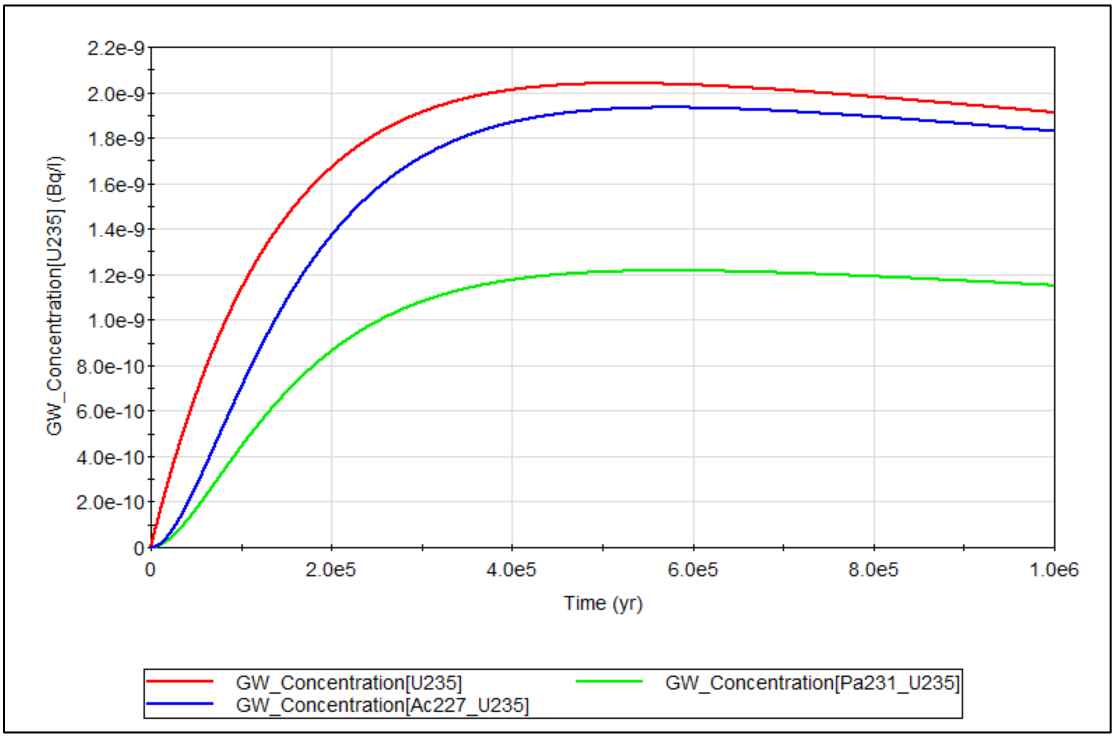


Figure 10. Time-dependent concentrations of U-238 in groundwater

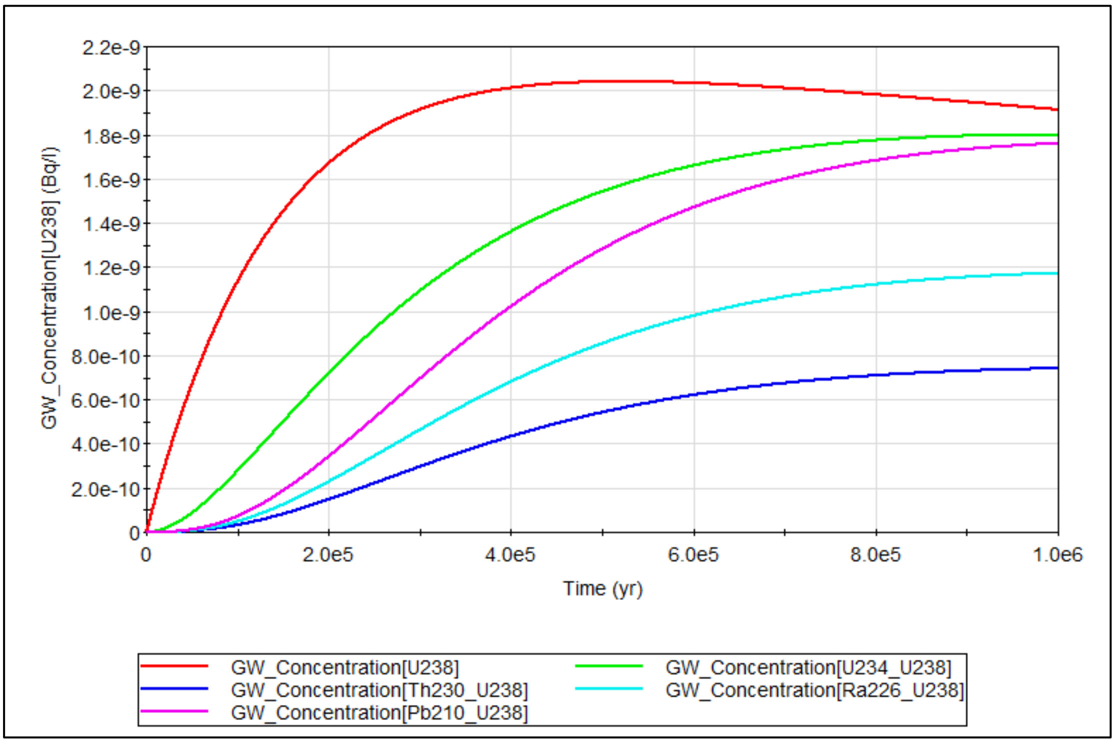


Figure 11. Time-dependent concentrations of Np-237 in groundwater

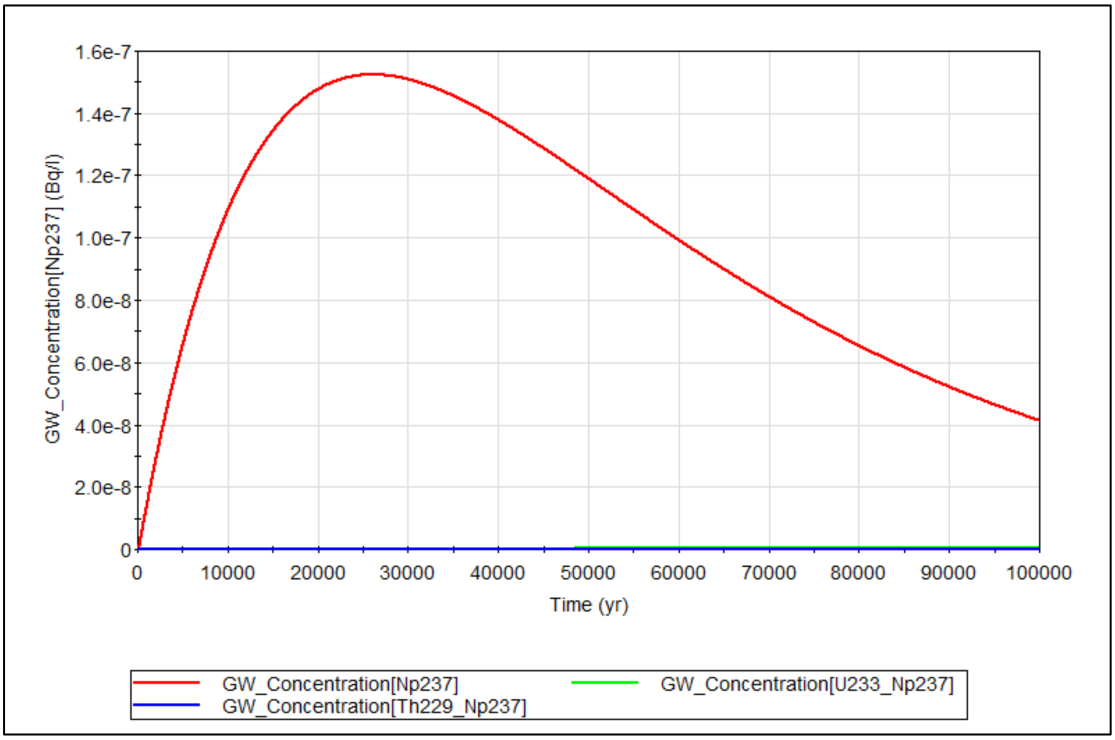


Figure 12. Time-dependent concentrations of Pu 240 in groundwater

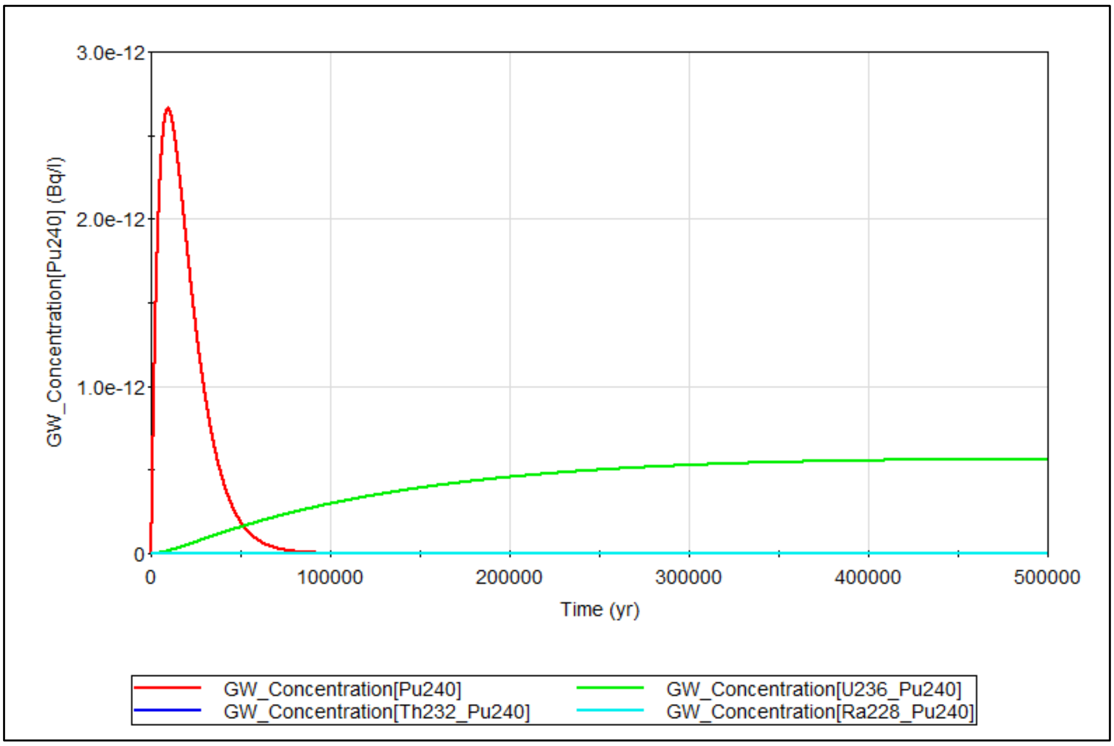
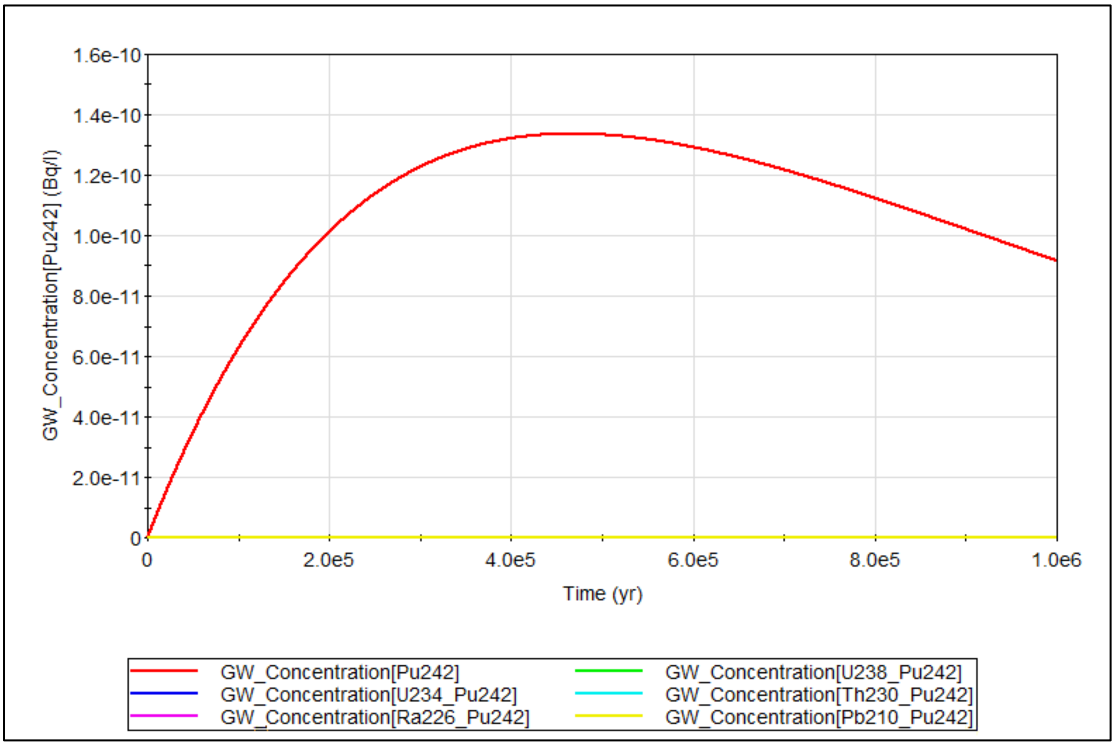


Figure 13. Time-dependent concentrations of Pu-242 in groundwater



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