Site Restoration

Winfrith Site: End State Radiological Inventory

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See table on page 5.

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Winfrith Site:
End State Radiological Inventory

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Report History

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Version	Date	Author	Amendments / Change
Version 1 Draft 1	10/07/2019	T.D. Baldwin, J. Graham, E.K. Phipps and W. Ludlam	Draft for Magnox review.
Version 1	18/12/2019	T.D. Baldwin, J. Graham, E.K. Phipps and W. Ludlam	Revised in response to comments from Magnox Ltd (Matt Izzo, Harry Miller and David Wickenden).
Version 2 Draft 1	13/03/2023	C. Eldridge, J.C. Farrow, E.K. Swain-Phipps, T. Harrison, T.D. Baldwin, J. Graham	Revised to address review comments from the Environment Agency and incorporate additional characterisation data. Uses void and backfill volumes from January 2023 Conceptual Site Model report, clarifies demolition level for each reactor, includes new inventories for the Dragon Purge Gas Pre-Cooler spill, SGHWR mortuary tubes, and SGHWR bulk structure tritium contamination, plus additional sample results. SGHWR components are identified in more detail. New sensitivity analysis sub-sections have been added to every inventory feature discussion, and a new section has been added to qualitatively consider confidence in the characterisation data, inventory derivation, and overall confidence in the estimate. Alternate inventories are presented for every feature.
Version 2 Draft 2	20/02/2024	C. Eldridge, J.C. Farrow, E.K. Swain-Phipps, T. Harrison, T.D. Baldwin	Revised to include Dragon B78 and A59 inventories, reflect new decision to demolish Dragon to ground level instead of 1 m below ground level, address review comments on SGHWR from James Graham (NNL) and on inventory-related content in the Conceptual Site Model from the EA and Quintessa, ensure consistent terminology, and make graphical and editorial improvements.
Version 2 Draft 3	11/10/2024	J.C. Farrow, C. Eldridge, C.M. Herbert, T.D. Baldwin	Revised in response to NRS End State (Joe Roberts) and Intelligent Customer (Alan Fisher and Kenneth McKee) comments; to include new inventory for the Dragon Primary Mortuary Hole Structure as derived by NRS based on 2023 sampling campaign; and to reflect the latest plans for the A59 area.
Version 2	17/12/2024	J.C. Farrow, C. Eldridge, T.D. Baldwin	Revised to implement minor corrections resulting from changes in Version 2 Draft 3 that had not been universally applied, and to respond to further NRS review comments. Further minor modifications made in response to NRS approver review.

Executive Summary

E1 The Winfrith nuclear site, a former nuclear power research and development site which housed nine unique experimental reactors, is being decommissioned. The site operator, Nuclear Restoration Services (NRS), is developing a proposal for the site decommissioning that includes on-site disposal of radioactive waste. This involves a combination of disposal in-situ of radioactive below-ground structures, disposal of radioactive waste (mainly blocks of concrete and broken concrete from demolition of the above-ground building structures), and deposit of non-radioactive waste (blocks of concrete, broken concrete and brick) for the purpose of infilling unwanted below-ground voids as part of land restoration. Following remediation and landscaping, the planned end state for the site is heathland open to the public for recreational purposes.

E2 NRS is developing the Waste Management Plan (WMP) and Site-Wide Environmental Safety Case (SWESC) for the Winfrith site, which will be submitted as part of the application for a variation to the site's radioactive substances regulation (RSR) permit granted by the Environment Agency under the Environmental Permitting Regulations to allow on-site disposal of radioactive wastes. The principal regulatory guidance of relevance to this application is the environment agencies' *Guidance on Release of Nuclear Sites from Radioactive Substances Regulation* (GRR).

E3 This radiological inventory report informs a number of assessments and safety arguments to be made for on-site disposal of radioactive waste at Winfrith. The overall objective of this report is to develop an estimate of the radiological inventory for the features that are being considered for on-site disposal using the best available information at the time of writing. Subject to further optimisation and characterisation, the radiological features proposed for on-site disposal as part of the Winfrith end state are:

- The Steam Generating Heavy Water Reactor (SGHWR) complex.
- The Dragon Reactor complex, including the B78 Dragon Fuel Store building and the spent fuel Mortuary Hole Structure it contains.

E4 The two reactor complexes will be demolished to ground level or below, accessible recyclable materials removed (e.g. wood, metal), and the above-ground concrete and rubble used to fill the below-ground voids. Non-radiological materials on the site, such as demolition materials from buildings and excavations (e.g. soil, concrete, brick), and existing material stockpiles and soil mounds, may be used for void filling, capping and landscaping on the site.

E5 A third radiological feature presented is the area of historically remediated ground following the removal of the A59 Active Handling and Decontamination building. This area is the subject of a separate inventory report and so only a summary is included in this report. Following options assessment, which the dedicated A59 inventory report supported, NRS intends to further remediate the radiological contamination in this area sufficient to demonstrate that the remaining ground is out-of-scope (OoS) of RSR. Thus, the A59 area does not form part of the RSR permit application. However, inventory information for the A59 area is needed to support site decommissioning activities and the radiological risk assessment. Therefore, remediated A59 feature inventory estimates are included here.

E6 In this report, a feature refers to a discrete contaminated structure or area (e.g. SGHWR bioshield, ponds and primary containment), composed of one or more components. Separate inventory estimates have been made for components of the in-situ features that are distinctly different in radiological fingerprint, or amount, or spatial extent of contamination or activation. These estimates support the radiological performance assessments and provide inputs to the optimisation process.

E7 The inventory estimates have been developed for the identified end state features and components through meetings with facility staff to understand key components and processes, the range of data available, and ongoing decommissioning plans. Inventory estimates compiled from the available characterisation data were compared to facility plans to identify any missing components, and the facility use histories were reviewed to assess if the fingerprint and inventory estimates were appropriate. The estimates were discussed with NRS staff to identify any inappropriate assumptions, gaps and inconsistencies, and if additional data were available.

E8 Assumptions have been made where there is limited information for some components and access limitations prevent sampling and additional characterisation at this time. Where specific sampling or characterisation data are not currently available or are insufficient, other experience at Winfrith or elsewhere has been considered as appropriate. Any calculations and/or assumptions used to develop the inventory estimates are applied in a realistically conservative manner and are stated in the report. However, whilst conservative assumptions are made, the inventory estimates must still be credible (i.e. not overly conservative), otherwise appropriate optimisation assessments cannot be made. The uncertainties and assumptions discussed in this report for the two reactor complexes (the A59 inventory report has its own table of uncertainties) are summarised in the appended Uncertainty Management Plan table and are categorised as follows:

- Potential for additional plant, structures and any contaminated land associated with SGHWR and Dragon to be included in the inventory scope.
- Uncertainties associated with comprehensiveness, scope, and applicability of waste fingerprints.
- Use of generic material compositions and densities due to lack of site-specific data.
- Adequateness and statistical robustness of the available characterisation data.
- Impact of changes to current outline demolition and backfill plans.

E9 A summary of the potential radioactive waste inventory that may remain on the site at the end state, in terms of total activity and the activity concentrations associated with each of the features of the SGHWR and Dragon Reactor complexes, and the A59 area, is presented in Table ES.1 for a reference date of 1 January 2027. The inventory is based on the current understanding of the SGHWR, Dragon and A59 features and components, drawing on the characterisation, decontamination and decommissioning carried out to date. Table ES.1 presents a cautious but credible estimate of the inventory that could be left on the Winfrith site at the end state and clearly indicates the dominance of the SGHWR inventory (98% of the total radioactivity) over that of the Dragon Reactor complex and A59 (around 1% each). The most significant features of the SGHWR inventory are the bioshield (59% of the SGHWR total), and then the secondary (11%) and primary (10%) containments. The inventory is presented in terms of total activity, and in terms of average and maximum activity concentrations. Note that the maximum concentration has been derived from the maximum activity concentration measured for each radionuclide across all samples obtained for a given feature, not from the sample with the maximum total concentration; therefore, it is very unlikely that such a conservative maximum would occur in any one future sample.

E10 The identified gaps, uncertainties and assumptions have been used in this report to support a qualitative assessment of the confidence in the inventory estimates for each component and in derivation of alternative, more conservative, inventory estimates, also presented in Table ES.1. The qualitative assessment reflects the comprehensiveness of the characterisation data supporting the inventory, the confidence in the inventory derivation approach, and the overall confidence in the inventory derived for each component (and the significance of this) as a function of the total feature group inventory. The alternative inventory estimates assume the maximum, rather than average, characterisation data by default, but alternative assumptions have also been made where there are other sources of uncertainty. The alternative inventories explore the impact of uncertainties, but are not considered to be realistic estimates.

E11 The identified gaps and uncertainties are also used by NRS to inform the need for additional characterisation. NRS will undertake further characterisation as demolition activities proceed and additional parts of the facilities are safely accessible, in an approach set out in the Staged Inventory Management Plan. Additional characterisation will be undertaken as needed during the EA's determination period, and during implementation of the end state. Emplacement Acceptance Criteria will also be applied to control material that is emplaced in the reactor voids. NRS recognises that it carries a risk in undertaking characterisation after the permit application has been submitted. If material is discovered beyond that indicated in the permit application then, following an options assessment, it is acknowledged that the material will need to be removed for off-site disposal or a delay in the permit application/final release incurred as revised assessment documentation is submitted. However, the reference inventory estimates presented here are considered to be a credible but cautious estimate of the end state activity that are characterised proportionately to the hazard presented. Sensitivity to alternative inventory assumptions has also been considered. In practice, the inventory estimates are expected to reduce as decommissioning proceeds and further characterisation information becomes available. The end state radiological inventory report will be revised as necessary as additional characterisation and sampling data become available.

Table ES.1: Reference and alternative inventories of radioactive waste for features potentially remaining on the Winfrith site at the end state, with the estimated activity concentration and total inventory associated with each feature of SGHWR, the Dragon Reactor and A59 area presented for a reference date of 1 January 2027. *(This page is set to print on A3.)*

Feature		Rationale for Inventory Estimate	Contaminated Mass [kg]	Contaminated Volume [m^3]	REFERENCE INVENTORY						ALTERNATIVE INVENTORY					
					Feature Total Activity		Activity Concentration [Bq/g]	Maximum Activity Concentration [Bq/g]	Reactor Complex / Area Activity		Feature Total Activity			Activity Concentration [Bq/g]	Reactor Complex / Area Activity	
					[MBq]	Feature %			[MBq]	Feature %	[MBq]	Feature %	Increased by factor		[MBq]	Feature %
SGHWR	Bioshield	Based on characterisation data from 2 concrete cores and neutron activation modelling of the concrete and rebar in the bioshield.	7.65E+05	3.14E+02	3.58E+05	58.6%	4.69E+02	8.92E+03	6.12E+05	98.0%	5.22E+06	88.3%	14.6	6.83E+03	5.91E+06	99.4%
	Mortuary Tubes	Preliminary, high-level approach in the absence of characterisation data which adopts the sum of activities of primary circuit pipework, moderator circuit pipework, ponds liners, activated rebar, and activated reactor components for mortuary tubes liners. Further characterisation expected.	2.75E+03	3.50E-01	8.11E+03	1.3%	2.95E+03	9.37E+03			2.56E+04	0.4%	3.2	9.30E+03		
	Primary	Based on available characterisation data from Room 111, and deeper concrete intervals in the primary containment, assumed depths of penetration of contamination into the building fabric, SGHWR primary external contamination fingerprint (FP-028) and proportion more than 1 m below ground level.	4.96E+06	2.07E+03	6.05E+04	9.9%	1.22E+01	1.55E+03			2.55E+05	4.3%	4.2	5.15E+01		
	Secondary	Based on available characterisation data from the structure, 7 secondary containment fingerprints, assumed depths of penetration, and the proportion more than 1 m below ground level (assume comprises Levels 1-3). Some areas assumed to be inactive.	4.21E+06	1.75E+03	6.97E+04	11.4%	1.65E+01	5.62E+03			1.35E+05	2.3%	1.9	3.20E+01		
	Ponds	Based on 2016 ponds characterisation programme comprising 17 cores from pond floor areas and 126 wall cores.	1.17E+06	4.87E+02	1.09E+04	1.8%	9.32E+00	7.01E+03			2.01E+04	0.3%	1.9	1.73E+01		
	Ancillary Areas	Based on characterisation data where available and applicable fingerprints (including FP-003, FP-016 and FP-026). Some areas assumed to be inactive.	1.89E+06	7.89E+02	3.33E+03	0.5%	1.76E+00	7.81E+01			1.66E+04	0.3%	5.0	8.77E+00		
	Bulk structure	To account for tritium contamination of the bulk concrete. Based on the mass of accounted for structure and the median tritium activity for components with an inventory.	2.86E+07	1.19E+04	1.86E+04	3.0%	6.50E-01	6.50E-01			3.54E+04	0.6%	1.9	1.24E+00		
	Backfill	Rubble mounds assumed to be at out-of-scope (OoS) of RSR (to be confirmed in future characterisation). Incorporates inventory from demolished Levels 4-10.	6.12E+07	2.97E+04	8.23E+04	13.5%	1.35E+00	2.74E+03			2.04E+05	3.4%	2.5	3.33E+00		
Dragon	Below cutline Bioshield	Based on characterisation data from 6 cores, fingerprints for Dragon Upper Support Ring concrete blocks and the mild steel baseplate, and by analogy with SGHWR neutron activation modelling.	2.57E+05	9.25E+01	1.51E+03	20.9%	5.86E+00	2.84E+01	7.23E+03	1.2%	6.41E+03	25.2%	4.2	2.49E+01	2.55E+04	0.4%
	Below	Fingerprint derived from characterisation data for 10 datasets at various locations in Dragon and	4.58E+06	1.91E+03	8.12E+02	11.2%	1.52E+01	1.61E+02			6.30E+03	24.7%	7.8	1.55E+01		

Feature		Rationale for Inventory Estimate	Contaminated Mass [kg]	Contaminated Volume [m^3]	REFERENCE INVENTORY						ALTERNATIVE INVENTORY					
					Feature Total Activity		Activity Concentration [Bq/g]	Maximum Activity Concentration [Bq/g]	Reactor Complex / Area Activity		Feature Total Activity			Activity Concentration [Bq/g]	Reactor Complex / Area Activity	
					[MBq]	Feature %			[MBq]	Feature %	[MBq]	Feature %	Increased by factor		[MBq]	Feature %
	cutline B70 Building Contamination	scaled using most active hotspot measured in the 2018 in-situ sampling campaign; assumes 5% of the building structure is surface contaminated. ^{3}H ingress 30 cm into building structure. Betalite store area inventory (included in this row) calculated separately using a dedicated fingerprint.														
	PGPC Spill	Preliminary estimate based on Dragon primary coolant fingerprint (shown to closely correlate with Purge Gas Pre-Cooler (PGPC) contamination), together with an estimate for the total activity remaining in the contaminated floor region derived from MicroShield dose modelling. 95.5% of contamination is assumed to be removed. Further characterisation expected.	7.92E+01	3.30E-02	9.50E+02	13.1%	1.20E+04	1.20E+04			9.50E+02	3.7%	1.0	1.20E+04		
	Backfill	Assumed to comprise the above-ground portions of the bioshield, B70 building and B78 building, emplaced as concrete blocks and/or rubble, together with some material from the existing rubble stockpiles. Based on demolition to ground level and known dimensions, the inventory is expected to contain 51% of total bioshield activity and 65% of the building surface contamination activity and ^{3}H ingress into structure.	1.29E+07	6.54E+03	3.88E+03	53.7%	3.02E-01	1.61E+02			1.16E+04	45.4%	3.0	8.98E-01		
	Primary Mortuary Hole Structure	Estimate for mortuary holes and cross vents based on systematic 2023 survey and sampling campaign; estimate for main ventilation ducts and sump based on smear from ventilation outlet stack (2016 inventory).	2.51E+03	3.20E-01	3.37E+01	0.5%	5.18E-01	1.34E+01			4.76E+01	0.2%	1.4	7.33E-01		
	B78 Floor Slab	Fingerprint, contamination level and % of contamination present are assumed to be the same as for B70 general building contamination. ^{3}H ingress 30 cm into building structure.	2.56E+05	1.07E+02	4.01E+01	0.6%	1.52E+01	4.29E+01			2.20E+02	0.9%	5.5	1.57E+01		
A59	PSA / Pit 3 APC	Remediated OoS end state inventory estimate derived for (i) the historical remediation works excavation surface at the cessation of soil removal; and (ii) infill material used to create the post-remediation ground surface. Primarily based on remediation dataset including verification gamma monitoring, radiochemical sampling and analysis prior to backfilling; and post-backfill monitoring and sampling, supplemented by subsequent analysis.	2.20E+06	1.10E+03	3.41E+02	6.2%	1.10E-01	2.42E+00	5.49E+03	0.9%	9.05E+02	7.0%	2.7	2.86E-01	1.30E+04	0.2%
	A591 / HVA APC		6.95E+05	3.47E+02	1.29E+03	23.5%	1.83E+00	2.19E+01			1.60E+03	12.3%	1.2	2.32E+00		
	Other A59 Areas		1.61E+07	8.07E+03	3.86E+03	70.3%	1.94E-01	2.25E+00			1.05E+04	80.7%	2.7	8.66E-01		

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Appendix A Uncertainties, Assumptions and Gaps 300

Glossary and Acronyms

ACL	Above Cutline
ACW	Ancillary Cooling Water
ALARP	As Low As Reasonably Practicable
ALES	Active Liquid Effluent System
APC	Area of Potential Concern
AOD	Above Ordnance Datum
ATS	Active Tools Store
CAD	Computer Aided Design
CCR	Climate Control Room
Component	A part of a **feature** for which a separate inventory is derived, corresponding to each row of the tables in Section 4.
cps	counts per second
CSM	Conceptual Site Model
Demolition cutline	The level above which structures will be demolished; at or below approximately ground floor slab level.
DQO	Data Quality Objectives. An approach to developing sampling plans for effective data collection activities that support decision making.
EA	Environment Agency
EAC	Emplacement Acceptance Criteria
EAST	External Active Sludge Tanks
ECW	Emergency Cooling Water
End State	The condition of an **NDA site** or a part thereof once decommissioning and clean-up activities have ceased.
EPR16	Environmental Permitting Regulations 2016 (as amended)
ESC	Environmental Safety Case - The collection of arguments, provided by the developer or operator of a disposal facility that seeks to demonstrate that the required standard of environmental safety is achieved (also see **SWESC**).
FCD	Failed Can Detection
Feature	Discrete contaminated structure or area, composed of one or more **components**. For the SGHWR **reactor complex**, features include the bioshield, **mortuary tubes**, primary containment, secondary containment, ponds and ancillary areas. For the Dragon **reactor complex**, features include the bioshield, reactor building and

	Primary **Mortuary Hole** Structure. For the A59 area, features include two **APC**s and the rest of the A59 area.
FP	Fingerprint. Percentage distribution of radionuclides contaminating a structure (valid for a specific date). Fingerprints are generally assigned to features and allow the whole radioactive inventory to be specified based on characterisation of a single radionuclide (e.g. ^{137}Cs).
GRR	A guidance document produced by the UK's environment agencies, with the full title "Management of radioactive waste from decommissioning of nuclear sites: Guidance on Requirements for Release from **RSR**".
GTLD	Gas Tritium Luminescent Devices
HCUP	Helium Clean Up Plant
HRGS	High Resolution Gamma Spectrometer
HTR	High Temperature Reactors
HVA	Heavy Vehicle Airlock
Inadvertent human intrusion	Any inadvertent human action that accesses the waste or that damages a barrier providing an environmental safety function after the release from **RSR**.
IAEA	International Atomic Energy Agency
ISOCS	In-Situ Object Counting System
IEP	Interim End Point. The point in time at which the Winfrith IES is achieved.
IES	Interim End State. The condition of the Winfrith **site** following completion of all physical decommissioning and clean-up activities required to make the land suitable for the next planned use of the site (but an environmental permit or other restrictions remain in force).
ILW	Intermediate Level Waste
In-situ disposal(s)	(Of redundant below-ground radioactive structures) On-site disposal of solid radioactive waste, such as a buried structure, by leaving it permanently in position, together with any necessary preparatory works.
IWS	Integrated Waste Strategy
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LOD	Limit of Detection
LSD	Liquid Shut Down
m agl / bgl	metres above / below ground level

MH	**Mortuary Hole**
Mortuary tubes / mortuary holes	Structures within the reactor complexes used for storing a variety of radioactive items. Within this report, the terms “mortuary tubes” and “mortuary holes” are used to refer to the structures associated with **SGHWR** and Dragon Reactor respectively. This usage reflects both the terminology most commonly found in plant documentation and their differing geometry (the SGHWR mortuary tubes have a smaller diameter than the Dragon mortuary holes and are open-ended, whereas the Dragon mortuary holes are closed at the bottom).
MSP	Main Shield Plug
NDA	Nuclear Decommissioning Authority
NIST	U.S. National Institute for Standards and Technology
NORM	Naturally Occurring Radioactive Material
OECD	Organisation of Economic Co-operation and Development
ONR	Office for Nuclear Regulation
Out of Scope / OoS	Material or waste with a level of radioactivity such that it is deemed to be non-radioactive for the purposes of legislation and is not subject to any regulatory requirement under **RSR**.
PA	Performance Assessment
PGPC	Purge Gas Pre-Cooler
PIE	Post Irradiation Examination
PSA	Pressurised Suit Area
QA	Quality Assurance
Radioactive waste	Radioactive material that is no longer of use.
Radioactive material	Material in which the concentrations of radionuclides are greater than the values specified in **RSR**. Excludes material lawfully disposed of as waste or contaminated ground that remains where it was contaminated.
Reactor complex	The group of buildings and other structures associated with each reactor remaining on the Winfrith **site** (**SGHWR** and Dragon). Each reactor complex consists of several **features**.
Room	In the **SGHWR reactor complex**, a numbered “room” (numbers originate directly from site) that may correspond to a conventional room with four walls and a roof, or an open platform or sub-area of a space. One or more rooms may make up a **component**. In the Dragon **reactor complex**, room is used in a more general sense.
RPV	Reactor Pressure Vessel
RSA 93	Radioactive Substances Act 1993

RSR	Radioactive Substances Regulation. A generic term used by the environment agencies to cover the different regulations in force in the four different countries of the United Kingdom. In England, radioactive substances regulation refers to **EPR16**.
RT	Radionuclide Transport
SGHWR	Steam Generating Heavy Water Reactor
SGS	Segmented Gamma Scanner
SIMP	Staged Inventory Management Plan
Site	Term used both in a general sense to refer to the land occupied by the Winfrith research reactors and their surroundings, and specifically to refer to the land delineated by the environmental permit as constituting the authorised premises. Note that the boundaries of the land owned by the NDA, the land delineated by the environmental permit as constituting the authorised premises, and the land delineated by the nuclear site licence as constituting the licensed premises are all different at Winfrith.
SoLA	Substances of Low Activity Exemption Order
SWESC	Site-Wide Environmental Safety Case. A documented set of claims to demonstrate achievement by the **site** as a whole of the required standard of environmental safety.
System	In the **SGHWR reactor complex**, an interconnected network or circuit (such as the moderator circuit or ventilation system) that may be associated with different **components** and/or **features**, and may also cut across them.
USR	Upper Support Ring
WACM	Winfrith Abrasive Cleaning Machine
WMP	Waste Management Plan
WSCP	Winfrith **Site** Closure Programme

Winfrith Site: End State Radiological Inventory

1 Introduction

1.1 Background

1 The Winfrith nuclear site, located in Dorset, is a former nuclear power research and development site, which hosted nine research and prototype reactors as well as laboratories. The site is owned by the Nuclear Decommissioning Authority (NDA) and operated by Nuclear Restoration Services (NRS)[1]. Site construction commenced in 1957 [1, p.4] and, following decades of reactor research and development, NESTOR and DIMPLE were the final site reactors to cease operations in 1995 [1, p.10]. Site decommissioning works have been on-going since the 1990s. Following remediation and landscaping, the planned end state for the site is heathland open to the public for recreational purposes.

2 Activities involving radioactive substances at the Winfrith site are regulated by the Environment Agency (EA) under the Environmental Permitting (England and Wales) Regulations 2016 (EPR16) [2] and as amended in 2018 [3; 4], 2019 [5] and 2023 [6]. Release from radioactive substances regulation[2] (RSR) cannot take place until the EA is satisfied that all activities involving radioactive substances and any disposals of radioactive waste (solid, liquid or gaseous) on or from the site have ceased, and that the site is in a state that will ensure a satisfactory standard of protection for people and the environment. Regulatory guidance was published in July 2018 in the *Management of radioactive waste from decommissioning of nuclear sites: Guidance on Requirements for Release from Radioactive Substances Regulation* (referred to here as the GRR) [7].

3 The GRR requires operators to assess different options for the disposal of radioactive waste arising from decommissioning, including on-site disposal options. Following options analysis, on-site disposal was identified by NRS as a credible option for some materials remaining at the Winfrith site [8]. Therefore, NRS is developing a proposal that entails on-site disposal of radioactive waste and deposit of recovered non-radioactive waste. A suite of documents, headed by a Site-Wide Environmental Safety Case (SWESC) and a Waste Management Plan (WMP), and supported by a series of underpinning topic reports (Figure **1.1**), is being produced by NRS to support the regulatory applications required to permit on-site disposal. The end state radiological inventory report presented here has been produced to support the application to vary the site environmental permit.

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1 Established by the Atomic Energy Authority (UKAEA), site ownership was transferred to the NDA in 2005. The site was originally operated directly by UKAEA and then by a variety of subsidiaries, including Research Sites Restoration Ltd (RSRL). Magnox Ltd, which managed the site from 2015, transitioned to NRS on 1 April 2024.

2 Radioactive substances regulation is a generic term used by the environment agencies to cover the different regulations in force in the four different countries of the United Kingdom.

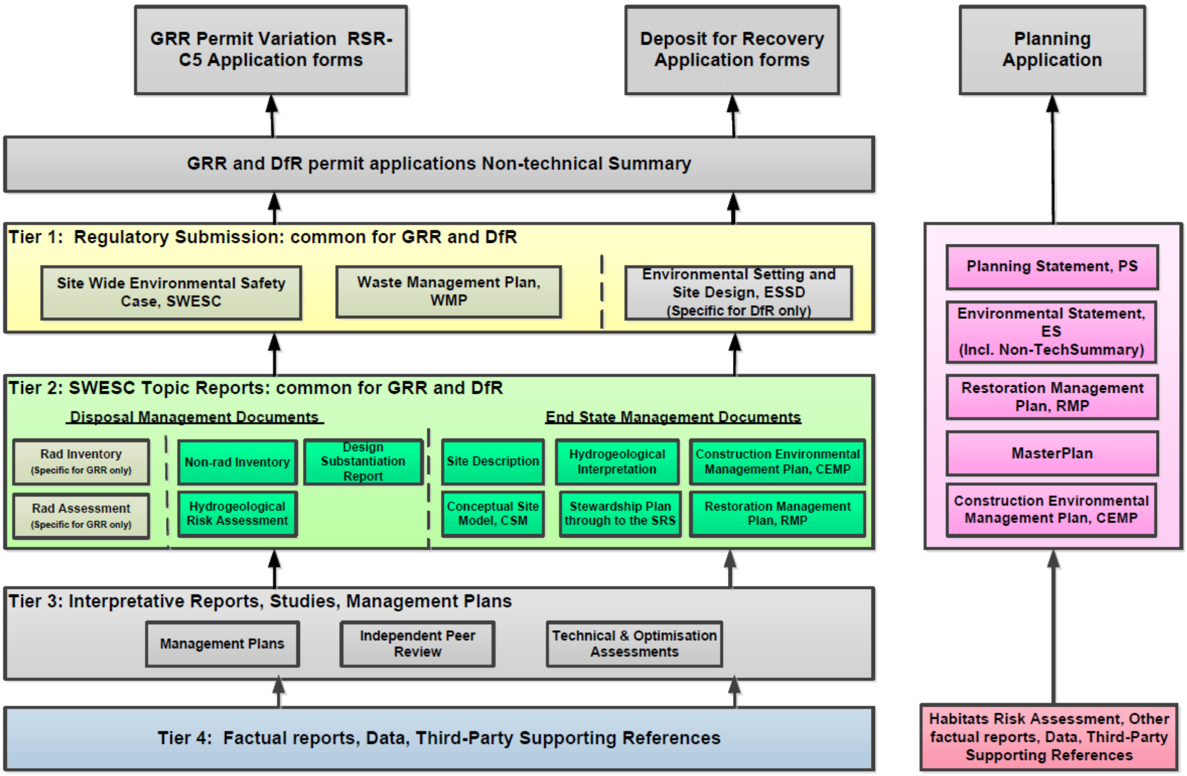

Figure 1.1: Winfrith end state GRR permit variation and deposit for recovery application documentation hierarchy.

1.2 Objectives

4 Radiological performance assessments for the site end state will need to include consideration of aqueous releases into the environment, potential activities taking place on the site during the period of RSR and any subsequent land use (e.g. residency, public access or farming), and hypothetical inadvertent human intrusion into radioactive features on the site following the period of RSR. These assessments will require knowledge of the radioactive inventory, in terms of both total activity (becquerels) and concentration (becquerels per unit mass). Therefore, the objectives of this radiological inventory report are to:

- Collate and develop a traceable radioactive inventory for the features that are anticipated to remain on site at the end state, based on the best available information at the time of writing.
- Estimate, for each feature or component[3] proposed for on-site disposal, the concentration of individual radionuclides and any spatial variation.

[3] A feature is a discrete contaminated structure or area, composed of one or more components. A separate inventory is derived for each component.

1.3 Scope

5 This report considers the radiological features that are planned to remain on site at the end state, as this is the scope of the required input to the radiological performance (or risk) assessments. The Winfrith Starting Case [9] defined the features (structures and areas of contamination) that could potentially remain on the site at the end of active decommissioning and form part of the end state. Subsequent optimisation assessments and NDA/NRS strategic decision-making[4] means that, subject to further optimisation and characterisation, the radiological features proposed for on-site disposal as part of the Winfrith end state are part of the following (Figure **1.2**):

- The Steam Generating Heavy Water Reactor (SGHWR) complex.
- The Dragon Reactor complex, including the B78 Dragon Fuel Store building and the spent fuel Mortuary Hole Structure it contains.

6 This report describes in detail the inventory derivation for the two reactor complexes (SGHWR and Dragon).

7 A third radiological feature presented in this report is the area of historically remediated ground following the removal of the A59 Active Handling and Decontamination building. This area is the subject of a separate inventory derivation report [10]. Following options assessment, which the dedicated A59 inventory report supported, NRS intends to further remediate the radiological contamination in this area sufficient to demonstrate that the remaining ground is out-of-scope (OoS) of RSR [11]. Thus, the A59 area does not form part of the RSR permit application. However, inventory information for the A59 area is needed to support site decommissioning activities and the radiological performance assessment, and to inform site monitoring expectations. Therefore, a summary of the A59 area inventory is included here and an estimate for the remediated OoS end state inventory presented.

8 The proposal for on-site disposal of the reactors involves a combination of disposal in-situ of radioactive below-ground structures, disposal of radioactive waste (mainly blocks of concrete and broken concrete from demolition of the above-ground building structures), and deposit of non-radioactive waste (blocks of concrete, broken concrete and brick) for the purpose of infilling unwanted below-ground voids as part of land restoration. Thus, the two reactor complexes will be demolished to or below ground level, accessible recyclable materials removed (e.g. wood, metal), and the above-ground concrete and rubble used to fill the below-ground voids. Non-radiological (i.e. OoS of RSR) materials on the site, such as OoS demolition materials from buildings and excavations (e.g. soil, concrete, brick), and existing OoS material stockpiles and soil mounds, may also be used for void filling, capping and landscaping on the site.

9 The Interim End Point (IEP), the point when decommissioning of the site and physical operations involving radioactive waste are complete, is not yet fully constrained. Therefore, this report presents an estimate of the potential radiological inventory remaining on the Winfrith site on 1 January 2027. It is currently anticipated that the site IEP will be reached in the 2030s and so the inventory presented here will be subject

[4] All radioactive features and wastes on the Winfrith site are listed in the site WMP. The WMP identifies the anticipated management route for each feature/waste stream and the optimisation assessments undertaken to support decision-making.

to an additional few years of radioactive decay and ingrowth, which is accounted for in the radiological performance assessments undertaken.

10 The scope of this report is limited to consideration of the radiological inventory. A parallel report assessing the non-radiological inventory has also been developed [12].

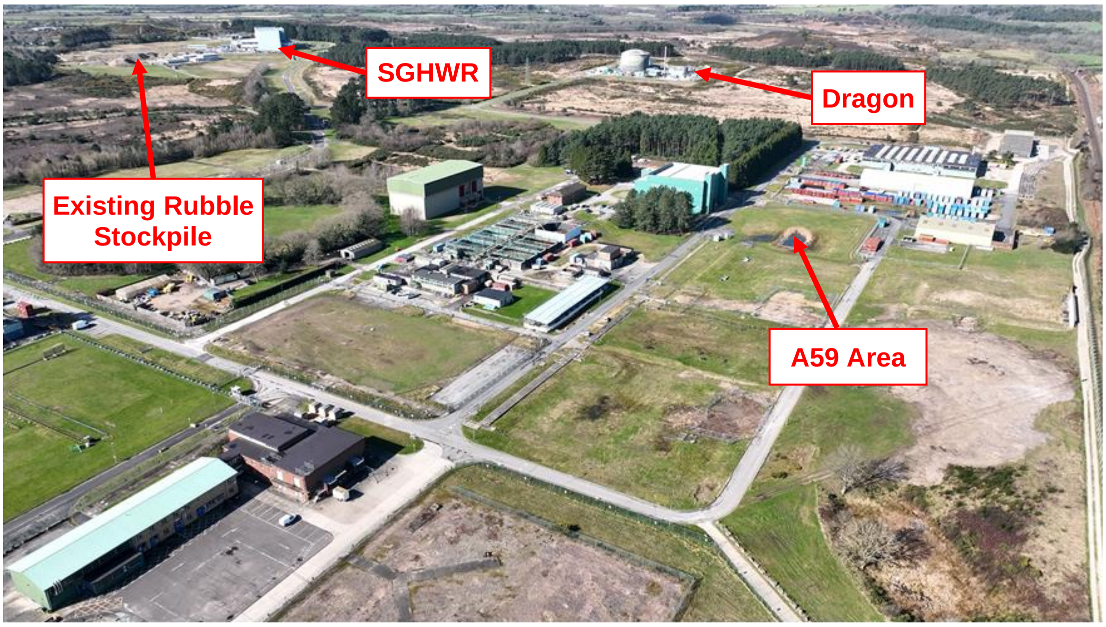

Figure 1.2: Aerial view of the Winfrith site from the east with the two reactor complexes, the A59 area and existing rubble stockpiles marked.

1.4 Approach to Inventory Development and Uncertainty Management

1.4.1 Motivation and Focus

11 As noted in Section 1.2, the primary role of this inventory is as an input to the radiological performance assessment (PA) of the Winfrith site end state. The inventory and PA are part of a risk-driven iterative loop whereby the results from PA of a given inventory estimate are used to identify which features make the greatest contributions to dose, and hence can be used to guide future cleaning and/or characterisation. Such risk-informed work will result in a revised inventory estimate which will form the input to a subsequent PA. This process is discussed further in the Staged Inventory Management Plan, SIMP [13].

12 In the PA, the inventory will be used to assess the potential dose resulting from releases due to the natural evolution of the in-situ disposals, and inadvertent human intrusion into such features. The inventory will also provide inputs to the optimisation process. Both uses require separate consideration of parts of the on-site disposal features. Therefore, in deriving the inventory, separate estimates have been made for components of the in-situ features that are distinctly different in radiological fingerprint, or amount, or spatial extent of contamination or activation.

13 The focus of this report is deliberately on the fingerprint and total activity of the different features and components[5]. It is acknowledged that some radionuclides will have greater impact on human and/or environmental health than others and this is primarily addressed in the PA, although this aspect does feed into discussion throughout the report in relation to uncertainties (particularly in fingerprints), sensitivity to different assumptions and the derivation of alternative inventories, and overall confidence in and significance of the inventory estimates for certain features.

1.4.2 Methodology

14 The following general approach was applied to develop reference inventory estimates for the identified end state features and components of the reactor complexes[6]:

- Facility staff were interviewed to understand key components and processes, the range of data available, and ongoing decommissioning plans.
- The available characterisation data were compiled and inventory estimates for each component calculated in a series of underpinning spreadsheets that accompany this report ([14; 15] and supporting spreadsheets). The estimates developed consider, so far as is practicable, the appropriate mechanisms by which structures may have become contaminated (i.e. neutron activation and/or radiological contamination).
- The estimates were compared to facility plans to identify any missing components, and the use histories of the various parts of the features were checked to assess if the fingerprint and inventory estimates were appropriate.
- The estimates were discussed with NRS staff to identify any inappropriate assumptions, gaps and inconsistencies, and if additional data were available.
- Confidence in the inventory estimates developed for each component was qualitatively assessed and then used, along with known uncertainty data, to undertake sensitivity analysis and develop alternative inventories for use in the radiological PA. The alternative inventory estimates assume the maximum, rather than average, characterisation data by default, but alternative assumptions have also been made where there are other sources of uncertainty. The alternative inventories explore the impact of uncertainties, but are not considered to be realistic estimates.

15 As indicated above, the reference inventory estimates have been developed based on the available characterisation information (which may have been obtained to support operator safety during decommissioning, rather than on-site disposal). Assumptions have been made where there is limited information for some components and access limitations (radiological and conventional safety constraints during decommissioning)

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[5] Maximum specific activities are also presented and discussed to provide a comprehensive picture of the inventory. Where appropriate, these are used in the derivation of alternative inventories, including alternative activity concentrations for use in inadvertent human intrusion and site occupancy assessments as part of the PA.

[6] The approach to inventory development and uncertainty management relating to the A59 area is discussed in the separate inventory report [10] and not repeated here; the rest of this section relates only to the two reactor complexes (SGHWR and Dragon). However, new A59-related uncertainties identified during development of this report are included in Appendix A.

prevent sampling and additional characterisation at this time. Where specific sampling or characterisation data are not currently available or are insufficient, other experience at Winfrith or elsewhere has been considered as appropriate. Any calculations and/or assumptions used to develop the inventory estimates are applied in a realistically conservative manner and are stated in the report. However, whilst conservative assumptions are made, the inventory estimates must still be credible (i.e. not overly conservative), otherwise appropriate optimisation assessments cannot be made.

1.4.3 Management of Uncertainties and Assumptions

16 Gaps in available information and uncertainties and assumptions associated with the radiological inventory are recorded in this report for use in the Uncertainties Management Plan [16]. These uncertainties and assumptions are noted within the report using an identifier of the form “INV-####-000”, which is an index to an entry in the Uncertainties, Assumptions and Gaps table in Appendix A. For example, an identifier of the form “INV-SGHWR-001” indicates an uncertainty, assumption or gap associated with the SGHWR inventory estimate.

17 The identified gaps, uncertainties and assumptions have been used in this report to support the qualitative assessment of the confidence in the inventory estimates for each component and in derivation of alternative, more conservative, inventory estimates. The qualitative assessment presented in Section 4 reflects the comprehensiveness of the characterisation data supporting the inventory, the confidence in the inventory derivation approach, and the overall confidence in the inventory derived for each component (and the significance of this) as a function of the total SGHWR or Dragon complex inventory.

18 The identified gaps and uncertainties are also used by NRS, alongside PA results, to inform the need for additional characterisation. NRS will undertake further characterisation as demolition activities proceed and additional parts of the facilities are safely accessible (the approach to this is set out in the SIMP [13]). Additional characterisation will be undertaken as needed during the EA’s determination period and during implementation of the end state. Emplacement Acceptance Criteria (EAC) [17], covering radiological, biological, chemical and physical properties, will also be applied to materials that are emplaced in the reactor voids. NRS recognises that it carries a risk in undertaking characterisation after the permit application has been submitted. If material is discovered beyond that indicated in the permit application then, following an options assessment, it is acknowledged that the material will need to be removed for off-site disposal or a delay in the permit application/final release incurred as revised assessment documentation is submitted. However, the reference inventory estimates presented here are considered to be a credible but cautious estimate of the end state activity, that are, in the main, characterised proportionately to the hazard presented. Sensitivity to alternative inventory assumptions has also been considered. In practice, the inventory estimates are expected to reduce as decommissioning proceeds and further characterisation information becomes available. The radiological inventory report will be revised as necessary as additional characterisation and sampling data become available.

1.5 Acknowledgements

19 The invaluable help and support from SGHWR and Dragon staff in gathering and interpreting the information required to produce this inventory report is gratefully acknowledged.

1.6 Report Structure

20 The radiological inventory assessment for the two reactor complexes is presented in two parts, this report and a set of accompanying spreadsheets for SGHWR and Dragon Reactor inventory data [14; 15]. The separate A59 inventory report is also accompanied by a spreadsheet [10; 18].

21 This report is structured as follows:

- Section 2 presents an estimate for the inventory that could potentially be left in the footprint of the SGHWR, considering each of the key features (e.g. the bioshield, primary and secondary containments, the ponds and ancillary areas) and the backfill material, and identifying key components of these. Each feature and component, its use and contamination history are described, along with the data sources used to develop the inventory estimate. A sensitivity analysis considering alternative assumptions is also presented for each component, along with an alternative inventory, and uncertainties/gaps that would benefit from additional characterisation are highlighted. The section concludes with a summary of the inventory estimate for the entire SGHWR reactor complex at the envisaged end state, and its alternative inventory estimate.
- Section 3 presents an estimate for the inventory that could potentially be left in the footprint of the Dragon Reactor (B70) and Dragon Fuel Store (B78) buildings. The section takes the same structure as Section 2, with the inventory (and alternative inventories) for each feature and component considered individually, and then presented for the entire Dragon Reactor complex at the envisaged end state.
- Section 4 discusses overall confidence in the inventory estimate for the two reactor complexes, including an assessment of relative confidence for the different components. This includes tables that summarise, for SGHWR and Dragon respectively, the available characterisation data, inventory derivation approaches used, uncertainties and overall confidence in the inventory (and significance of this) for each component considered.
- Section 5 presents summary tables for the estimated OoS inventory (and alternative inventory) that could potentially be left in-situ in the A59 area, noting that the derivation of the reference inventory is described in detail in the dedicated report [10].
- Section 6 presents conclusions of the study, including a summary of the total end state inventory.
- Section 7 presents a list of references that underpin the report.
- Appendix A contains a summary table of the gaps and uncertainties associated with the radiological inventory estimates identified during the development of this report.

2 Inventory Associated with the SGHWR Complex

2.1 Background

22 The SGHWR was built as a prototype power-producing heavy-water moderated reactor to demonstrate the viability of such systems; the SGHWR was the only water-cooled and heavy-water moderated reactor ever built in the UK [19]. Constructed between 1963 and 1967, the reactor reached its full power of 100 MW in January 1968. The SGHWR used slightly enriched uranium fuel, was moderated with heavy water, and used ordinary (light) water as a coolant. The reactor core consisted of 104 zirconium alloy pressure tubes, which passed through vertical aluminium tubes into a tank (calandria) of heavy water. The fuel elements in the pressure tubes comprised bundles of rods of uranium oxide pellets contained in zirconium alloy. Light water was pumped over the fuel elements and boiled in the core. The resulting steam was passed directly to the turbine, with condensate returned to the reactor to be mixed with the re-circulating water. After operating successfully for 23 years as a research reactor and supplying electricity to the national grid (Figure **2.1**), the reactor was shut down in October 1990.

Figure 2.1: The SGHWR building (D60), cooling towers and electricity sub-station during its operational period.

23 The SGHWR reactor building comprises ten main floors, or levels, with Levels 1 to 3 (L1-L3) forming extensive below-ground basements. Levels 4 to 10 (L4-L10) are above ground level. The main regions and levels of the building are shown in Figure **2.2** and Figure **2.3**.

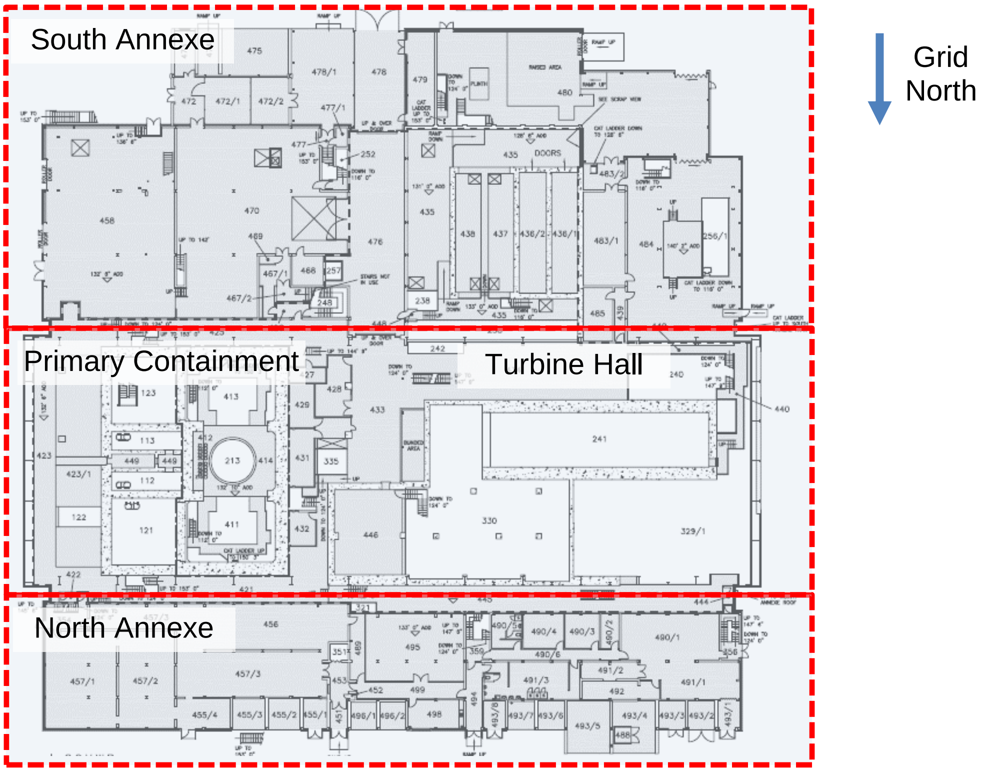

Figure 2.2: Plan view of SGHWR Level 4 (136' 6" level) showing general regions within the D60 building (edited from [20]). The building extent varies according to the level viewed.

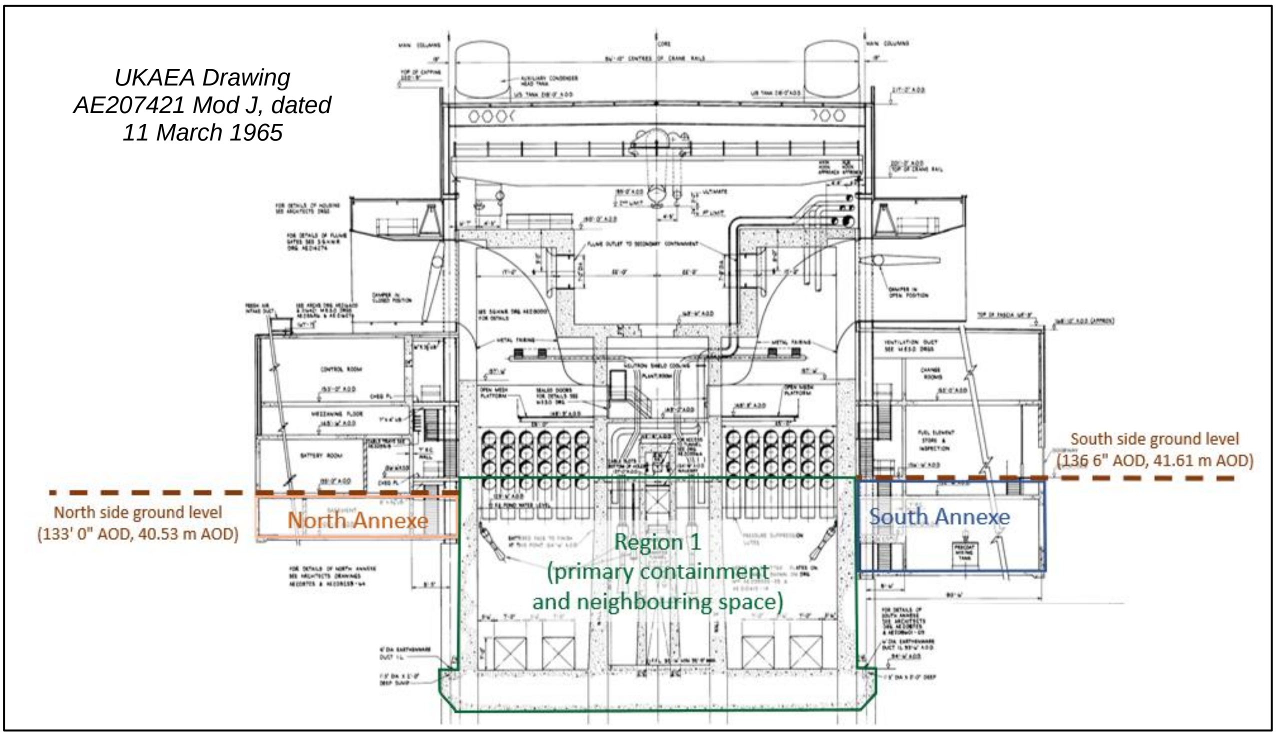

Figure 2.3: Cross-section thorough the SGHWR building with ground level indicated [21, Fig.606/5].

24 An extensive programme to decommission the SGHWR began in 1991. Stage 1 decommissioning began with the defueling and transfer of the 279 fuel elements to Sellafield, the flushing of the heavy-water moderator circuits, and the emptying and decontaminating of the fuel ponds [19]. The cooling towers were demolished, along with the diesel house, the round house (office accommodation) and the stack. Stage 2 decommissioning began in 2005 with a focus on the decommissioning of major items of plant and equipment in the secondary containment building and has included the removal of many large pieces of redundant equipment from more than 250 rooms and areas.

25 Plant and equipment removal is ongoing, with the focus now on the safe decommissioning, segmentation and disposal of the primary containment, reactor core and, finally, the building and all remaining facilities.

2.2 Proposed End State

26 The current plan is that the SGHWR reactor and plant building (D60) will be cleared to 1 m below ground level, with plant and accessible metal removed. The above-ground structure will be demolished and the resulting concrete blocks/rubble used to backfill the below-ground void spaces, along with spoil from existing rubble mounds elsewhere on site. Fixed traces of asbestos (painted and cast in concrete) will remain following bulk asbestos removal, and the ~3-mm-thick fibreglass pond liners will also remain.

27 Two inventory sets are therefore needed to model the SGHWR, one for the below-ground SGHWR structure potentially remaining in-situ and one for material used as the backfill.

2.3 Sources of the SGHWR Radioactive Inventory

28 A number of sources contributing to the SGHWR radioactive inventory remaining at the point of final demolition have been identified. These include activity derived from neutron activation of the reactor bioshield during reactor operation and activity resulting from contamination from contact with liquids or the atmosphere.

29 There were three main sources of contamination in SGHWR [22]:

- The primary circuit was directly in contact with the fuel and was the primary heat transfer medium. The primary circuit was contaminated due to activation (^{60}Co and ^{63}Ni form 98% of the total fingerprint [22, p.8]) and corrosion of the metal core components and transport through the circuit. The primary circuit also held a significant inventory of ^{137}Cs and tritium from fission products and activation of the light water.
- The moderator circuit contained deuterated water (D_2O) during operation. Exposure to high neutron fluxes led to significant tritium (93%) and ^{14}C (5%) activities [22, p.8] in the circuit during operations. Operational tritium levels were known to be in the region of 4 TBq/l.
- The ponds and fuel route had greater contact with spent fuel and therefore elevated alpha and ^{137}Cs contamination levels compared to other areas of the facility.

All contamination within the reactor facility originates from these areas. The contamination of any particular room in SGHWR will be dependent on the relative influence of the three contaminant sources and the processes undertaken.

30 No inventory associated with any external areas of the SGHWR is captured in this report. It is assumed that any such contamination, if present, will be removed or confirmed as OoS (see INV-SGHWR-001).

31 For the purposes of this assessment, the rooms and components associated with the SGHWR disposal inventory are grouped into the following features:

- Bioshield.
- Mortuary tubes.
- Primary containment.
- Secondary containment.
- Ponds.
- Ancillary areas.
- SGHWR bulk structure.
- Backfill.

2.4 SGHWR Characterisation Activities

32 The characterisation data contributing to the SGHWR inventory span almost two decades and were taken for a variety of purposes. The duration over which the data were taken and the diversity of objectives means that there is no consistent sampling methodology supporting the inventory for each component.

33 The majority of the characterisation campaigns supporting the inventory followed a Data Quality Objective (DQO) methodology. A key exception to this is the D60 endpoint characterisation study in 2005 which instead relied upon expert judgement and plant knowledge to select core locations without a pre-defined methodology. The 2005 campaign underpins the inventory for a number of the SGHWR components.

34 It should also be noted that in some campaigns, inventory derivation was not considered in the methodology design. This includes the oil spill characterisation campaign and the cofferdam sediment sampling. Where these campaigns provide relevant data, they are nonetheless included in the inventory derivation.

35 Sampling strategies for SGHWR characterisation fall into four broad approaches:

- Probabilistic sampling in which sampling locations are randomly chosen.
- Probabilistic sampling in which sampling locations are systematically chosen (e.g. cores at fixed intervals).
- Judgemental sampling targeting locations of expected contamination identified based on radiological data (such as hand-held probe or large area High Resolution Gamma Spectrometer (HRGS) surface surveys).
- Judgemental sampling targeting locations of expected contamination identified based on expert judgement and knowledge of process history.

36 SGHWR characterisation studies often adopted a mix of both probabilistic and judgemental sampling to ensure that samples collected were representative of both

hotspots and diffuse contamination. In some cases, sampling locations were guided by survey measurements of particular radionuclides such as ^{137}Cs (informed by process history and contamination provenance); it is acknowledged that in such surveys less easy-to-detect radionuclides may have been present but not targeted by the sampling. Sampling strategy and its likely effect on the representativeness of the resulting characterisation dataset is a key factor in the assessment of overall confidence in and significance of the inventory of different features and components (Section 4).

37 A timeline of the major SGHWR characterisation studies is presented in Table **2.1** and illustrated graphically in Figure **2.4**. The list is not exhaustive and there are additional activities targeting individual rooms or components that are not captured.

Table 2.1: Major SGHWR characterisation studies underpinning the inventory, their respective aims and adopted approach.

Date	Sampling study, aim and approach
2005	**D60 endpoint characterisation study** **Aim:** To scope the level contamination throughout the SGHWR structure, to establish potential remediation requirements and to facilitate waste estimates. **Approach:** Judgemental sampling based on plant history and expert advice. Cores targeted areas of expected contamination as well as areas expected to be clean across the entirety of the SGHWR structure.
2012	**Active Workshops** **Aim:** To demonstrate that the active workshops would meet delicensing criteria. **Approach:** Initial HRGS area survey of floor and walls used to develop a contamination map of the workshops which was then used to identify preferred core locations. A proportion of cores targeted identified areas of higher activity and the remainder were taken randomly.
2013	**Cofferdams (sediment/water)** **Aim:** To enable disposal of the groundwater in the cofferdams to the Active Liquid Effluent System (ALES). **Approach:** Most contaminated cofferdams (based on prior sampling) targeted, plus an additional cofferdam to ensure samples were spatially well distributed.
	Off-gas beds **Aim:** To determine a fingerprint, determine the amount of decontamination required and demonstrate that the room will be capable of being delicensed. **Approach:** Mix of systematic sampling of walls and floors and targeted sampling based on knowledge of process history.
	Pond clean-up areas **Aim:** To determine a fingerprint, determine the amount of decontamination required and demonstrate that the room will be capable of being delicensed. **Approach:** Judgemental sampling approach targeting areas of both high and low activity identified by an area survey.

Date	Sampling study, aim and approach
2016	**Maintenance and decontamination pit** **Aim:** To determine the activity in the pit and its distribution, and support development of a fingerprint and risk assessment for the pit. **Approach:** An initial area survey was undertaken to identify areas of elevated contamination. A proportion of cores targeted identified areas of higher activity and the remainder were taken randomly.
	Ponds **Aim:** Develop an inventory for the ponds and assess the depth of penetration of contamination through the walls. **Approach:** Sampling of the walls based on a uniform mesh. Sampling of the floor based on a mix of random and judgemental sampling based on prior survey data.
	Effluent delay and sludge tanks **Aim:** Develop a fingerprint, estimate the contamination and provide sufficient analytical data to determine/underpin Best Available Technique (BAT) for risk assessment and remediation. **Approach:** Areas from floor selected based on process knowledge and/or visual observations, focusing on areas with a higher risk of contamination.
2018	**D630 Rubble stockpiles** **Aim:** Preliminary programme of characterisation undertaken in support of on-going technical and optioneering programmes. **Approach:** Mix of random and judgemental sampling based on a large area surface survey.
2019	**Primary containment** **Aim:** Assess the remaining radiological inventory and likely activity concentrations. **Approach:** Judgemental sampling based on process history for the floor and uniform sampling of the walls.
2023	**Oil spills** **Aim:** To characterise oil spills across the SGHWR building and to identify the associated radiological and chemical contamination. **Approach:** Judgemental sampling based on location of visibly identified oil spills as well as one sample taken from a clean (no oil) area as a control.

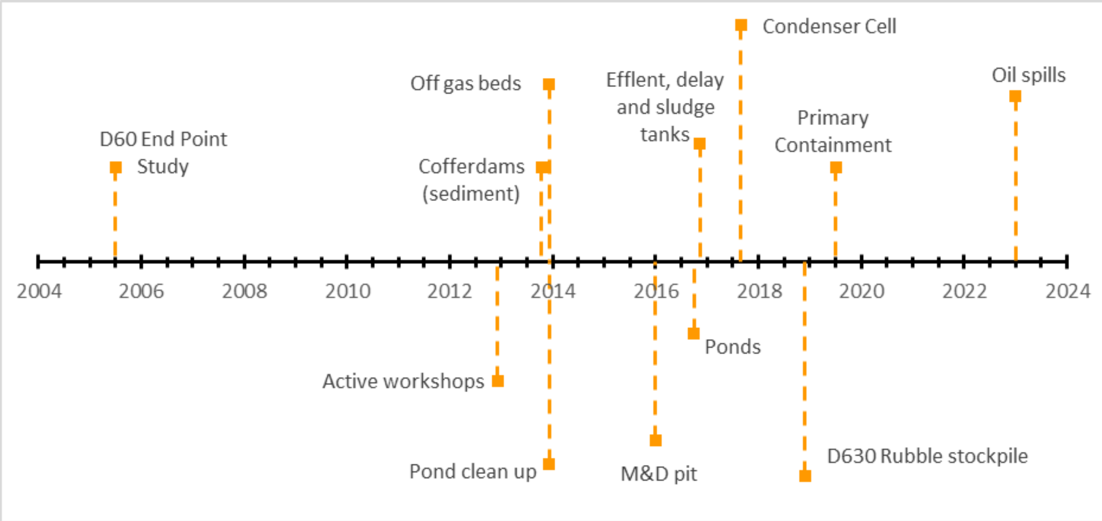

Figure 2.4: Partial timeline of SGHWR characterisation activities, indicating the reporting dates for characterisation data supporting the SGHWR end state inventory.

2.4.1 SGHWR Characterisation Data Presentation and Inventory Calculation

38 As noted in Section 1.4, the data presented in this report are underpinned by a series of spreadsheets. For SGHWR, there is a set of underlying spreadsheets covering specific features, components or levels within the reactor complex (referenced within the relevant sections), and an overarching main SGHWR inventory spreadsheet [14]. The split of information and calculations between the underlying spreadsheets and the main SGHWR spreadsheet is as follows:

- Raw and manipulated characterisation data supplied by NRS, including activity concentrations, and key dimensional data (generally surface area and contamination depth) for individual rooms and areas can be found in the underlying spreadsheets.
- Each underlying spreadsheet includes a "Characterisation Data Summary" sheet which collates (via links to other sheets in the file) all information to be imported into the main SGHWR spreadsheet in a common format.
- Data are copied and pasted directly from each of these summary sheets into the "Measured Activity Data" sheet of the main SGHWR spreadsheet, which therefore acts as a compilation of key data from all underlying spreadsheets. No further data manipulation is carried out on this sheet, maintaining traceability between the main and underlying spreadsheets.
- Further sheets in the main spreadsheet are used to estimate the activity concentrations of missing or hard-to-measure radionuclides, and decay all activity concentrations to the inventory date of 01/01/2027.
- The main spreadsheet also uses universal density assumptions for different materials (see Section 2.9) to calculate masses of contaminated material (at room and feature level) and total activities at feature level, split into below-ground components that will be left in-situ and above-ground components that will form part of the void backfill.

In the feature-specific sections below (Sections 2.10 to 2.17), characterisation data and figures are generally taken from the underlying spreadsheets, while the feature-level inventory tables are taken from the main SGHWR inventory spreadsheet.

2.5 SGHWR Fingerprints

39 Many radionuclide fingerprints [23] have been developed for the SGHWR, primarily for the purposes of waste disposal and assessment. These fingerprints reflect the varying radiation and contamination sources, the environments, and the proportion and type of materials with which the sources interact. Typically, a fingerprint for an area will be developed or verified during a sampling and analysis campaign. Fingerprint derivation is required to be documented in a Note for the Record and reviewed and approved by a Radioactive Waste Advisor (RWA)[7] for solid waste [24].

40 These area fingerprints typically reflect averaged radionuclide species found across the waste types in the investigation; however, at a more local scale significant heterogeneity in species may be encountered. For instance, experience indicates that significant radionuclide ratio differences may occur between surface contaminated material and contamination that has penetrated deeper into the material (usually carried in the liquid phase). Generally, some radionuclides including tritium and ^{90}Sr penetrate further into concrete than other radionuclides such as ^{60}Co and alpha emitters, which are retained in the surface layers and generally penetrate no further than the paint layer. At a sample scale (e.g. a core sample taken through a wall), this commonly leads to two distinct contaminant profiles, one containing a majority of low mobility radionuclides in the surface layer, and the other containing more mobile radionuclides in the concrete below the surface.

41 In this assessment, SGHWR waste fingerprints have been used to fill gaps that may be present in the analytical dataset (e.g. where radionuclides were not included in the requested analysis suite or were reported at the limit of detection (LOD)). Due to the size of the underlying dataset, this adjustment is made at a room scale, rather than sample scale. The uncertainty introduced by applying the various SGHWR fingerprints can vary in significance. For example, if there is a substantial characterisation dataset and the fingerprint is only used to fill the gap for a few minor contributors to the inventory and/or if the fingerprint applied is for the same material subject to the same operational history, then the additional uncertainty would be insignificant. Alternatively, as an example, if partial decontamination of a room was undertaken by removal of some areas of significantly contaminated paint, then application of a fingerprint based on paint/concrete core samples taken prior to decontamination would bias the correction towards a paint composition[8]. Similarly, a fingerprint used in end state inventory derivation should ideally be fully representative of the substrate it is to be applied to. However, in some instances, due to limitations in the dataset, the fingerprints applied in this assessment may have been derived for waste that has since

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[7] RWA designations have now been replaced with Appointed Suitably Qualified Experienced Persons (Monitoring) (ASQEP(M)), equivalent to Providers of Radioactive Substances Legislation Advice.

[8] Where possible, the underlying fingerprint paint and concrete data components have been separated out and/or the paint fraction in the fingerprint reduced to better reflect the material being considered in this assessment (e.g. see discussion for FP-016* in Table **2.2**).

been removed rather than the waste that remains. Differences in substrate materials and relative radionuclide migration mean that application of a fingerprint based on removed material will have increased uncertainty in the inventory estimate. The SGHWR fingerprints were reviewed at each use to apply the most suitable of those available and, where it is considered that this has notably increased uncertainty, this has been reflected in the inventory confidence assessment presented. Additionally, where there is a risk that key radionuclides (particularly those likely to drive significant impacts in the PA, such as actinides) may not be captured, alternative fingerprint assumptions are considered in the derivation of alternative inventories.

42 The approach adopted for applying fingerprints to a dataset is by ratio against a selected determinand common to both the fingerprint and analytical data, typically from the gamma spectrometry dataset, which is available for nearly all samples. If a fingerprint was entirely representative of the waste then any common determinand could be selected. In practice this is not the case and more accurate predictions can be made by selection on the basis of radionuclide relationships established during assessments of SGHWR decommissioning wastes, such as described in [25]. The approach generally adopted in the current assessment is as follows:

- Calculation of ^{55}Fe and ^{63}Ni based on ^{60}Co.
- Calculation of uranium isotopes based on ^{235}U.
- Calculation of plutonium and curium isotopes based on ^{241}Am, preferentially adopting alpha spectroscopy data over gamma spectroscopy data when both are available.
- Commonly using ^{137}Cs for other radionuclides.

43 For calculations of the bioshield using activation model data, ^{60}Co was used for all radionuclides. The uncertainties associated with the use of fingerprints are captured in INV-SGHWR-002.

44 The source and derivation approach for fingerprints used in the SGHWR inventory are summarised in Table **2.2**. Radionuclide activities and reference dates for the fingerprints are captured in Table **2.3** and Table **2.4**.

Table 2.2: Source and description of SGHWR fingerprints.

FP #	Descriptor	Derivation Approach	Source
FP-003	D60 Secondary Containment General Area	FP-003 removes ^{226}Ra from the original fingerprint FP-156, as it is naturally-occurring radioactive material (NORM) and was only detected at near the LOD in a small fraction of contributing samples.	[26]
FP-004	General A59 Fingerprint	FP-004 is a combined fingerprint for waste arising from the A59 building (which has now been demolished, with rubble contributing to the site material stockpile). The fingerprint was calculated from the combination of five other fingerprints which were derived from the total activity of A59 waste bags. The contributing fingerprints are: • FP-146 – Pressurised suit area • FP-147 – All other SB wastes • FP-148 – North Cave Line • FP-149 – Cave 9	[27]

FP #	Descriptor	Derivation Approach	Source
		• FP-150 – Remaining bags The contributing fingerprint activities for each radionuclide were added together to derive the combined fingerprint activities.	
FP-016*	SGHWR Active Workshop -	Fingerprints were derived separately for concrete and paint samples taken from across the active workshops and associated areas prior to decontamination. FP-016* is a combined fingerprint which is calculated from a weighted average of the concrete and paint fingerprints, assuming paint accounts for 0.1% of the waste activity and that concrete accounts for the remaining 99.9%. The original FP-016 was derived for waste sentencing and assumed a 90% paint to 10% concrete ratio which is considered inappropriate for inventory derivation for material left in-situ. The following assumptions were made in the development of the fingerprints: • All LOD values were reported as half LOD value for fingerprint development where the radionuclide was identified in any sample. • ^{63}Ni and ^{241}Am activities were derived from their respective ratios to ^{60}Co and ^{241}Pu in FP-003. • ^{90}Sr values were assigned based on the gross beta values (excluding ^{137}Cs and ^{60}Co values).	[84, §4; 14]
FP-018	SGHWR Pond Clean-Up	Fingerprints were derived separately for concrete and paint samples from the Pond Clean Up Area and Associated Rooms (Rooms 222, 223, 224, 225 and 228). Radionuclides at LOD or attributable to NORM were omitted from the fingerprints. The combined fingerprint FP-018 was calculated from a weighted average of the concrete and paint fingerprints, assuming paint accounts for 0.1% of the total waste volume (which provides a pessimistic activity calculation) and concrete accounts for 99.9% of the total volume.	[66]
FP-026	SGHWR Off-Gas Beds	Nuclide ratios were derived from a total of 1 smear sample, 9 paint samples and 37 concrete/brick samples which were collected from the floor and walls of the Off-Gas Bed rooms (rooms 236, 237, 253, 254, 258 and 324). Fingerprints were derived separately for concrete and paint and the combined fingerprint was calculated from a weighted average of the concrete and paint fingerprints, assuming paint accounts for 0.1% of the total waste volume (which provides a pessimistic activity calculation) and concrete accounts for 99.9% of the total volume.	[28]
FP-028	SGHWR Primary External Contamination	FP-028 was derived from samples taken from across the primary containment environment (smears from loose contamination and paint from general surfaces), as well as from metal samples from individual ancillary circuits. The fingerprint was developed based on relative contributions from the different sources: • Smears (surface contamination): 50% • Paint (fixed contamination): 20% • Metal (bound contamination): 30% ^{55}Fe, ^{63}Ni, uranium and plutonium activities were extrapolated from the expected ratio to ^{60}Co where not measured. ^{90}Sr activity	[25, §8]

FP #	Descriptor	Derivation Approach	Source
		was conservatively extrapolated by assuming it is present in its highest measured ratio to ^{137}Cs in the primary containment.	
FP-030	SGHWR Moderator Circuit	The moderator circuit fingerprint was produced based on a single metal sample (D60-P/14/D20/MET/009) as this was judged to be the most representative of the pipework as a whole and also minimised the contribution of scale within the pipe which has a disproportionate concentration of ^{60}Co. Alpha radionuclides are excluded from the fingerprint on the basis that there is no history, provenance or positively identified analytical data to suggest the presence of alpha radionuclides. ^{55}Fe and ^{63}Ni activities were extrapolated from the expected ratio to ^{60}Co where not measured. ^{90}Sr activity was conservatively extrapolated by assuming it is present in its highest measured ratio to ^{137}Cs in the primary containment.	[25, §7]
FP-034	SGHWR Fuel Ponds	The fingerprint was derived from a set of 19 floor cores from the SGHWR ponds. Radionuclides attributable to NORM or which were at LOD in all samples were omitted from the fingerprint. The ^{239}Pu/^{240}Pu and ^{90}Sr activities were extrapolated from the gross alpha and beta results respectively.	[82]
FP-038	SGHWR Maintenance and Decontamination Pit	The fingerprint was derived from a set of 8 cores (4 each from the floor and walls) from the maintenance and decontamination pit. Radionuclides with results that were all at LOD were removed from the FP calculation. To ensure that the amount of ^{90}Sr within the fingerprint is representative, ^{90}Sr activities were extrapolated from detectable gross beta activities where direct laboratory analysis for this radionuclide did not occur (it was assumed that all gross beta activity not attributable to ^{137}Cs was due to ^{90}Sr and its daughter ^{90}Y). Fingerprints were derived separately for concrete and paint and the combined fingerprint was calculated from a weighted average of the concrete and paint fingerprints, assuming paint accounts for 1% of the total waste volume (which provides a pessimistic activity calculation) and concrete accounts for 99% of the total volume.	[29]
FP-046	SGHWR Condenser Cell	The fingerprint was derived from a total of ten concrete samples (5 cores and 5 chipping samples). For the cores, only data for the top 20 mm of material was included as the activities at deeper intervals were mostly at LOD.	[30]
-	SGHWR Bioshield Concrete / Rebar Activation	Derived from Monte Carlo N-Particle® (MCNP) neutron transport and EASY-2003 activation modelling of the SGHWR reactor core and bioshield. The radionuclides in the activation model have been screened to eliminate those that have a half-life less than one year or that account for less than 0.01% of the total modelled activity. The screened-out radionuclides account for a total of 0.03% of the activity for the concrete fingerprint and 0.02% for the rebar fingerprint. The activities were renormalised following screening such that the total activity is conserved.	[44; 51]

Table 2.3: Screened SGHWR bioshield activation fingerprints.

Radionuclide	Bioshield concrete (01/01/2006)		Bioshield rebar (01/01/2006)	
	Proportion of activity	**Activity (Bq/g)**	**Proportion of activity**	**Activity (Bq/g)**
^{39}Ar	3.16E-04	28.0	-	-
^{133}Ba	2.81E-03	249.8	-	-
^{14}C	7.11E-04	63.1	-	-
^{41}Ca	1.45E-03	129.0	-	-
^{60}Co	4.27E-03	378.7	2.51E-03	160.1
^{134}Cs	3.89E-04	34.6	-	-
^{152}Eu	7.72E-02	6855.9	-	-
^{154}Eu	5.79E-03	513.8	-	-
^{155}Eu	1.24E-04	11.0	-	-
^{55}Fe	5.10E-03	452.5	9.95E-01	63588.6
^{3}H	9.00E-01	79903.0	1.45E-04	9.3
^{63}Ni	1.42E-03	125.9	2.88E-04	18.4
^{151}Sm	6.51E-04	57.8	-	-
^{204}Tl	-	-	1.74E-03	111.1
^{113m}Cd	-	-	1.05E-04	6.7
Total		**88,803.1**		**63,894.2**

Table 2.4: Radionuclide activity proportions for SGHWR fingerprints.

Nuclide	FP-003: D60 Secondary Containment General Area (01/06/2005)	FP-004: General A59 Fingerprint (24/10/2018)	FP-016*: SGHWR Active Workshop Combined (01/12/2012)	FP-018: SGHWR Pond Clean-Up (13/12/2012)	FP-026: SGHWR Off-Gas Beds (01/06/2005)	FP-028: SGHWR Primary External Contam (01/01/2006)	FP-030: SGHWR Moderator Circuit (01/06/2005)	FP-034: SGHWR Fuel Ponds (01/09/2016)	FP-038: SGHWR Maintenance & Decon Pit (01/01/2016)	FP-046: SGHWR Condenser Cell (01/08/2017)
^{3}H	1.01E-01	-	7.81E-01	7.25E-01	5.17E-01	6.24E-02	6.93E-01	3.92E-04	3.32E-01	1.92E-01
^{14}C	1.02E-02	-	5.45E-03	5.38E-03	2.60E-01	8.86E-04	2.28E-02	1.88E-04	1.36E-02	2.21E-02
^{137}Cs	3.79E-01	4.76E-03	1.04E-01	2.52E-01	1.47E-01	8.52E-01	2.18E-03	3.36E-01	5.76E-01	7.47E-01
^{57}Co	-	-	-	-	-	-	2.70E-01	-	-	-
^{60}Co	2.68E-01	3.51E-05	5.26E-03	9.69E-04	3.15E-03	4.95E-02	4.54E-03	5.63E-03	2.34E-03	6.08E-03
^{241}Am	9.77E-04	1.07E-04	8.78E-05	5.72E-03	2.14E-04	2.30E-05	-	1.39E-02	4.97E-04	-
^{94}Nb	-	-	-	-	-	-	2.29E-04	9.32E-05	-	-
^{125}Sb	-	6.92E-08	-	-	-	-	-	-	-	-
^{152}Eu	-	-	-	-	-	-	3.13E-03	-	-	-
^{154}Eu	-	-	-	2.90E-07	-	-	8.25E-04	1.24E-03	-	-
^{155}Eu	-	-	-	-	-	-	3.21E-04	1.68E-04	-	-
^{55}Fe	1.15E-01	-	-	2.13E-07	3.55E-02	1.22E-02	1.21E-03	5.33E-04	8.91E-03	8.00E-03
^{63}Ni	9.13E-02	3.03E-04	3.79E-03	2.22E-03	8.27E-03	2.31E-02	2.04E-03	7.09E-03	1.53E-02	1.14E-02
^{90}Sr	2.19E-02	1.73E-03	9.22E-02	8.04E-05	9.60E-03	1.38E-04	1.13E-04	5.13E-01	5.16E-02	1.38E-02
^{241}Pu	1.09E-02	2.96E-04	6.47E-04	-	1.48E-02	1.99E-04	-	7.80E-02	-	-
^{234}U	2.89E-04	1.73E-08	1.38E-03	1.56E-03	9.72E-04	-	-	4.84E-04	-	-
^{235}U	2.22E-05	8.63E-09	1.08E-04	1.09E-04	5.85E-05	-	-	2.05E-05	-	-
^{236}U	-	-	-	-	-	-	-	2.05E-05	-	-
^{238}U	2.37E-04	1.04E-07	1.14E-03	1.27E-03	1.36E-03	-	-	1.54E-04	-	1.12E-04
^{238}Pu	3.73E-04	1.08E-05	1.68E-03	1.94E-03	2.91E-04	9.16E-06	-	5.56E-04	-	-

Nuclide	FP-003: D60 Secondary Containment General Area (01/06/2005)	FP-004: General A59 Fingerprint (24/10/2018)	FP-016*: SGHWR Active Workshop Combined (01/12/2012)	FP-018: SGHWR Pond Clean-Up (13/12/2012)	FP-026: SGHWR Off-Gas Beds (01/06/2005)	FP-028: SGHWR Primary External Contam (01/01/2006)	FP-030: SGHWR Moderator Circuit (01/06/2005)	FP-034: SGHWR Fuel Ponds (01/09/2016)	FP-038: SGHWR Maintenance & Decon Pit (01/01/2016)	FP-046: SGHWR Condenser Cell (01/08/2017)
^{239}Pu	3.61E-04	5.04E-05	1.72E-03	1.96E-03	8.75E-04	1.56E-06	-	2.31E-02	-	-
^{240}Pu	2.95E-04	7.01E-05	1.40E-03	1.60E-03	8.75E-04	1.56E-06	-	1.89E-02	-	-
^{243}Cm	6.77E-07	-	-	3.22E-06	-	-	-	1.09E-06	-	-
^{244}Cm	4.56E-05	4.78E-05	1.59E-04	2.03E-04	-	-	-	4.75E-05	-	-

2.6 Naturally-occurring Radioactivity in SGHWR

45 All building materials contain various amounts of naturally-occurring radioactive nuclides. Materials derived from rock and soil contain mainly natural radionuclides of the uranium (^{238}U) and thorium (^{232}Th) series, and the radioactive isotope of potassium (^{40}K). In the uranium series, the decay chain segment starting from radium (^{226}Ra) is radiologically the most important and, therefore, reference is often made to radium instead of uranium [31]. The world-wide average concentrations of radium, thorium and potassium in the Earth's crust are about 0.04 Bq/g, 0.04 Bq/g and 0.4 Bq/g, respectively [31].

46 The bulk of the material in the SGHWR and demolished buildings now forming the rubble mounds comprises concrete and brick. Summary statistics of UK and EU concrete and brick ^{226}Ra, ^{232}Th and ^{40}K activity concentrations are presented in Table **2.5**.

Table 2.5: Summary of compiled UK and EU ^{226}Ra, ^{232}Th and ^{40}K activity concentrations in concrete and brick in [32].

Material	Area	^{226}Ra (Bq/g)			^{232}Th (Bq/g)			^{40}K (Bq/g)		
		Min	**Mean**	**Max**	**Min**	**Mean**	**Max**	**Min**	**Mean**	**Max**
Concrete	UK	0.018	0.060	0.089	0.013	0.032	0.043	0.370	0.492	0.650
	EU	0.003	0.060	1.300	0.003	0.034	0.556	0.014	0.341	1.190
Brick	UK	0.002	0.049	0.094	0.003	0.034	0.081	0.012	0.560	1.000
	EU	0.0001	0.051	0.281	0.002	0.049	0.233	0.012	0.563	1.098

47 In the SGHWR and demolished buildings now forming the rubble mounds, ^{238}U and other uranium isotopes may also be present from nuclear operations. However, due to the long half-lives of these radionuclides, insufficient time has elapsed since their generation to allow significant ingrowth of daughter products from the decay of anthropogenic uranium inventories. No records of operations using NORM have been identified on site that could have imparted elevated levels of these radionuclides from natural sources.

48 For the purposes of deriving an inventory, potentially naturally-occurring radionuclides have been treated as follows:

- Those that have been identified in SGHWR or rubble mounds fingerprints (^{234}U, ^{235}U and ^{238}U) are included in the inventory, as an anthropogenic source cannot be ruled out. A significant, but patchy, analytical dataset has been accumulated for naturally-occurring uranium isotopes, comprising gamma scan (^{235}U) and/or alpha spectrometry (^{234}U and ^{238}U); however, for some components/rooms no data exist. The approach taken is that where data exists for ^{234}U, ^{235}U or ^{238}U, no attempt has been made to separate and remove a contribution from a natural source. Where a gap in the uranium dataset exists (i.e. one or more of the isotopes are missing for a sample), but uranium is present in the waste fingerprint, then the waste fingerprint has been used to derive the missing values.

- Those that have half-lives of less than one year are excluded from the inventory on the basis that they will not be significant over timescales relevant for the end state.
- Those that are not identified in the SGHWR or rubble mounds fingerprints and have half-lives of more than one year (such as ^{40}K, ^{226}Ra and ^{210}Pb), are considered on a case-by-case basis. The general approach is to conservatively include them in the inventory if the relevant dataset includes both some above-LOD results and some (LOD or above-LOD) results that are higher than the range expected in UK concrete as set out in Table **2.5**. They are therefore usually excluded (i.e. a natural origin assumed) if either all results are LOD or all results are within the expected ranges, although they may be included for consistency if the majority of other rooms in the same area meet the criteria for inclusion. In the few datasets where ^{210}Pb is reported, the values are compared with those of ^{226}Ra to decide whether a case can be made for a natural origin. All such justifications are clearly set out in the relevant underlying spreadsheets for SGHWR features.

2.7 Future Contamination

49 There are a number of activities still occurring in the SGHWR or planned to occur that may contribute to the overall disposal inventory. The most consequential of these activities is expected to be the segmentation of the reactor core, which will involve activities in a number of areas spanning the primary and secondary containments as well as parts of the ancillary areas. Historical and ongoing decommissioning activities also mean that rooms have been re-purposed, walls removed and new structures added.

50 Under existing plans, the reactor core segmentation process cell itself will be situated on the Level 4 platforms in the primary containment (Rooms 413 and 414). The maintenance cell and semi-remote operations area will extend from the North Side Lagging Box Platform (421) into the North Corridor (411), through the current secondary containment boundary wall into the Guaranteed Supplies Switch Room (456) and Document Store (457). A number of other areas will be involved in supporting the segmentation activities as storage or import/export areas, including the mortuary tube area (412), the deuterising plant room (427), the south transducer room (429), the Failed Can Detection (FCD) plant room (431), the neutron shield plant room (516), and the cluster loop rooms (612 and 628) [33]. In addition, the extract pump pit (328) and the Emergency Cooling Water (ECW) tank room/active scaffold store (446) have the potential to be reused during segmentation. The activities undertaken and rooms involved are subject to change, and will be accounted for in future updates of this report as core segmentation proceeds.

51 Following the core segmentation there will be a post-operational clean-out of the area. This will include the removal of Segmentation Cell Plant followed by remediation/decontamination of the affected areas.

52 Furthermore, there are a number of areas within the SGHWR building that were being used for ongoing waste operations until recently. These include the Winfrith Abrasive Cleaning Machine (WACM) facility (Room 480) for cleaning contaminated metals, the new fuel room (now a drum store) (458), and the Segmented Gamma Scanner (SGS) area (470). However, contamination levels in the waste processing areas were assessed

during routine operational surveys and are known to be very low [22; 33]. All waste handling plant has now been removed and the rooms decontaminated.

53 The uncertainty in the disposal inventory due to ongoing and future sources of contamination is captured in uncertainty INV-SGHWR-009.

2.8 Presence of Barytes Concrete in the SGHWR

54 Barytes concrete uses a barium aggregate to substantially enhance the shielding properties of the concrete. Within the SGHWR barytes concrete is known to be present in the walls, roof and floor of the D_2O plant room, the two element loop room and the cluster loop room [34; 35]. An excerpt of one of the SGHWR sections indicating the presence of barytes concrete is presented in Figure **2.5**. Barytes concrete is not believed to be present in the bioshield.

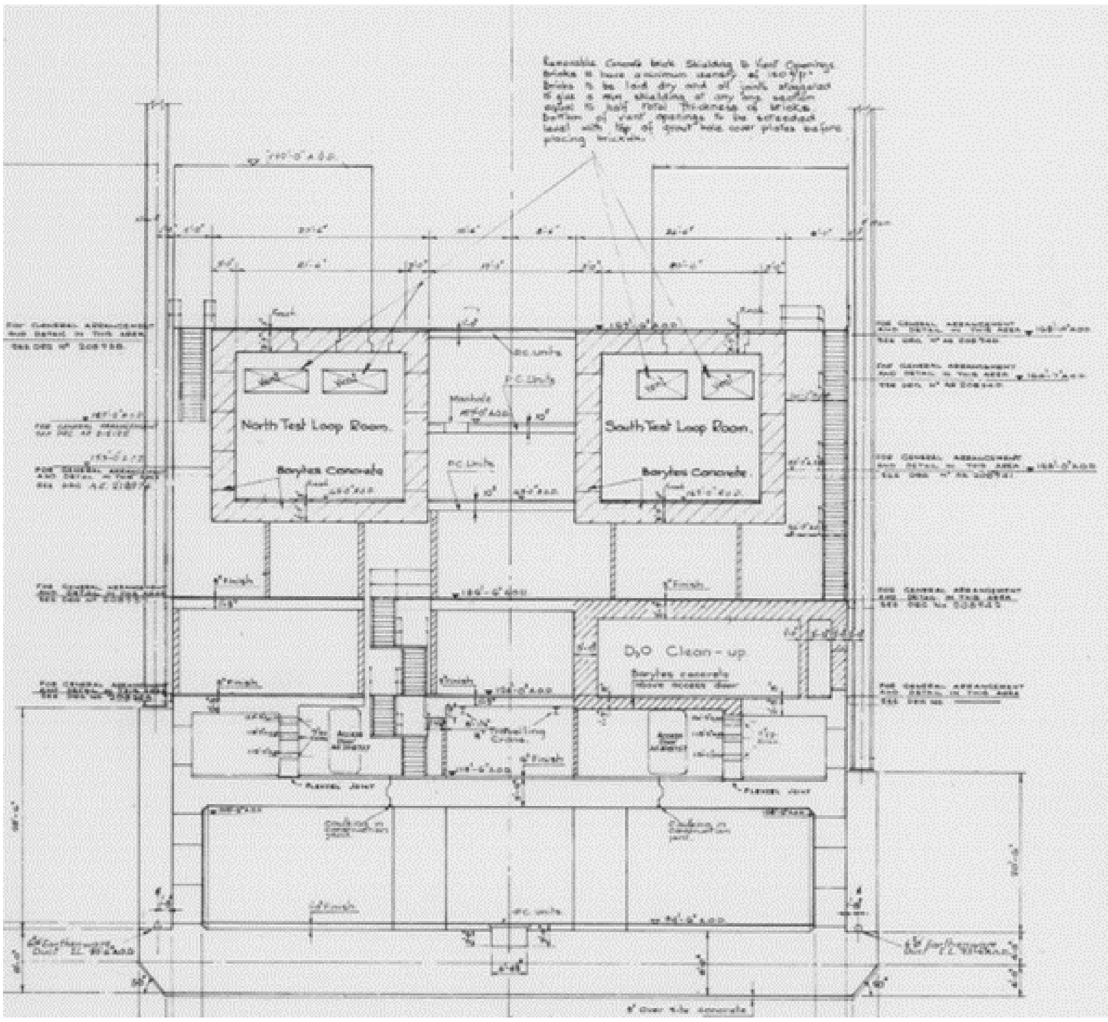

Figure 2.5: Section of the SGHWR structure showing the location of barytes concrete (hatched areas) [35]. Barytes concrete comprises the walls, roof and floor of the two element loop room (labelled north test loop room in diagram), cluster loop room (south test loop room) and the D_2O plant room (labelled D_2O clean-up).

55 The locations where barytes concrete has been identified are not in areas of substantial irradiation. Barytes concrete is typically used where additional shielding is required and/or there is insufficient space to use ordinary concrete; Wilson [36] states that barytes concrete was used in SGHWR where it was not possible to increase the wall thickness to provide an effective neutron shield. It is speculated that barytes concrete was used in the identified areas because both the two element loop and cluster loop were expected to be used as experimental circuits and substantial shielding may have been needed for anticipated experiments (which never eventualised).

56 None of the barytes concrete in the SGHWR is expected to be activated, meaning that only activity due to surface contamination needs to be considered. It is assumed that where equivalent contamination of barytes and normal concrete occurs, the activity density (Bq/m^3) will be the same. Barytes concrete is higher density than normal concrete so the activity per unit mass of barytes concrete (in Bq/g) will be lower than the equivalent specific activity of normal concrete. This assumption allows sample data for barytes concrete to be applied to normal concrete and vice versa, with the specific activity scaled by the ratio between the densities of the concretes.

2.9 SGHWR Common Physical Parameters

57 The SGHWR contains several hundred rooms. In derivation of the inventory a number of common physical parameters have been used based on data available in existing Winfrith waste calculations [14; 21] and available literature. Common density values are listed in Table **2.6**. Room or component-specific parameters such as surface areas and depths of contamination are discussed in the following sections or in the underlying references.

Table 2.6: Adopted SGHWR infrastructure material densities.

Material	Density [kg/m^3]	Source
Concrete (in-situ)	2,400	Density of structural concrete used in the Conceptual Site Model (CSM) [21].
Broken concrete	1,714	Assume in-situ concrete density with a bulking factor of 1.4 [37].
Compacted broken concrete	1,967	Assume the density of broken concrete rubble compacts by a factor of 13% on emplacement (equivalent to a bulking factor of 1.22 compared to in-situ volume) [37].
Barytes concrete	3,650	Wilson [36] states the planned barytes density was 218 lbs/cu. ft. (~3,492 kg/m^3). The slightly higher density of barytes concrete used in the Dragon bioshield has been adopted [131].
Paint	1,500	Assume density of waterborne wall paint from [38], adopted in conjunction with an assumed paint thickness of 1 mm (see discussion in text below).
Steel	7,860	Density of plain carbon steel from CRC Handbook of Chemistry and Physics [39, p.12-204].
Brick	2,200	Density of brick selected from high end of range (1.4-2.2 g/cm^3) given in table “Density of Various Solids”, in CRC Handbook of Chemistry and Physics [39].

Material	Density [kg/m^3]	Source
Fibreglass	1,880	Density of type E epoxy resin fibreglass given in ASM engineered materials handbook [40, Tab.37].

58 The density of paint provides a challenge as there is a large spectrum of wet paint densities depending on the specific paint type [41]. The bulk density can also increase or decrease during drying depending on the formulation and application of the paint, which results in an even larger spread of potential densities[9]. A further issue is that the mass of paint in an area is derived from the surface coverage rather than volume, which requires coupled assumptions about the dry density and dry film thickness of the paint.

59 In order to derive a surface density (i.e. kg/m^2) for paint in the inventory, the drying of paint is neglected. This is pessimistic as the surface density will always decrease as solvent evaporates (as opposed to volume density which can increase due to the film becoming thinner). This also enables the use of density and coverage values for wet paint, which are more readily available than equivalent values for dry paint. The following density and coverage assumptions are made:

- A wet paint density of 1,500 kg/m^3 is adopted, equivalent to the highest density paint in [38] (waterborne wall paint). This is also consistent with the density of standard wall and ceiling paint from 3M [41].
- A surface coverage of 6 m^2/l per coat is assumed, equivalent to the lowest surface coverage quoted in [38] (lowest surface coverage selected to be pessimistic as surface coverage is inversely proportional to surface density).
- Six coats of paint are arbitrarily assumed; plant staff at SGHWR have expressed that this is expected to be a conservative assumption (more paint is more conservative because the highest activity concentrations are generally found in the paint layer).

60 The above values and assumptions result in an overall film thickness of 1 mm and a resulting surface density of 1.5 kg/m^2.

61 The uncertainties associated with use of the adopted physical parameters are captured in INV-SGHWR-003.

2.10 Bioshield

2.10.1 Feature Description

62 The SGHWR bioshield is a reinforced concrete structure located at the centre of the primary containment, which enclosed the reactor core during reactor operation. This is shown schematically in Figure **2.6**. The bioshield is located on Level 1 to Level 3 of the SGHWR building (Figure **2.7**).

[9] The bulk density of a paint depends on both the dry particle density of the solids in the paint as well as the amount of porosity (void space) in the dry paint layer. For a solvent-based paint, the porosity will mostly be introduced as the solvent evaporates during drying; the density behaviour of the paint during drying will be linked to the extent to which the solids contract to fill the introduced porosity. Epoxy-based paints may not have any component that evaporates and the density will not change as the paint dries.

63 The walls of the bioshield vary in thickness. Adjacent to the fuel storage ponds the thickness is as low as 1.22 m (4’); however, elsewhere the wall is 1.5 – 1.6 m thick. Adjacent to the Liquid Shut Down (LSD) plant room, the bioshield is attached to the primary containment and in combination forms a wall 2.82 m (9’3”)[10] thick (see Table **2.7**).

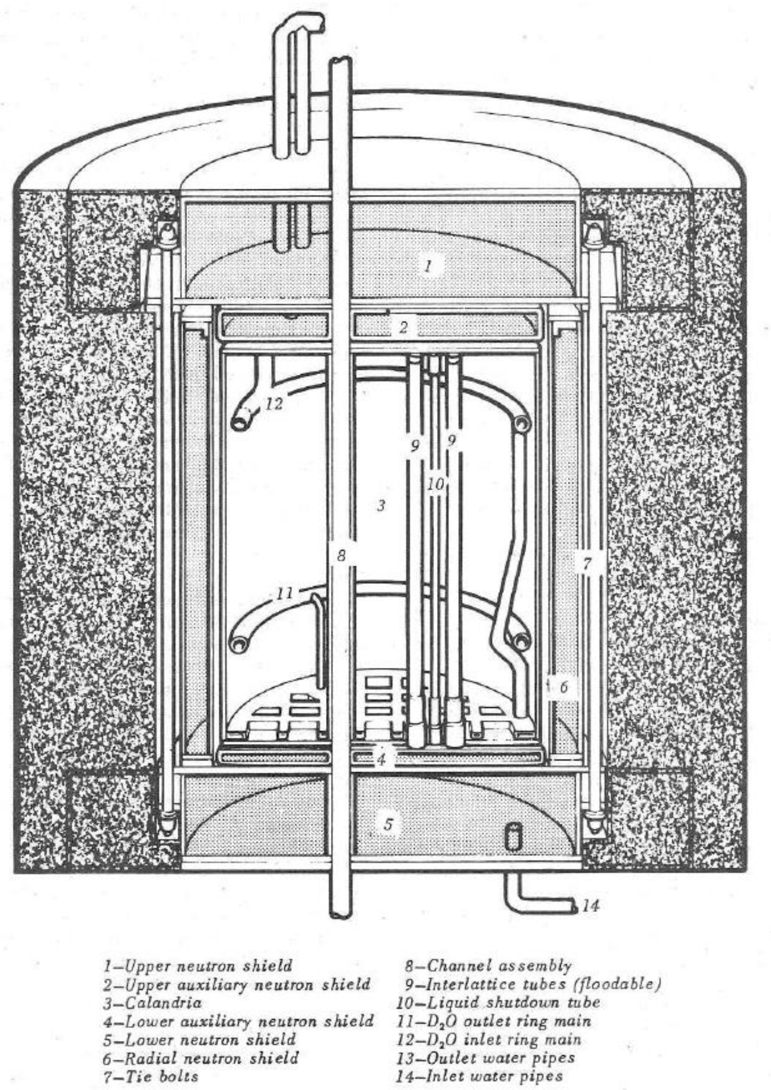

Figure 2.6: Schematic diagram of the SGHWR core and surrounding concrete bioshield [44].

[10] This incorporates aspects of the primary containment structure.

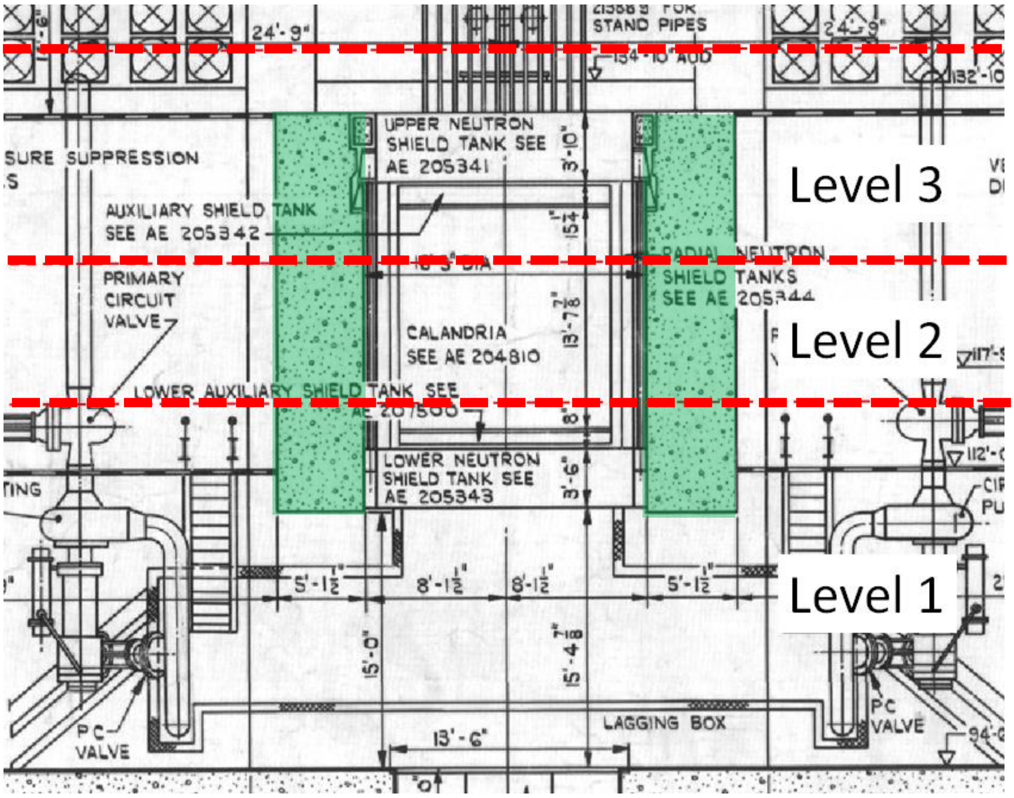

Figure 2.7: SGHWR bioshield (highlighted green) viewed from the NE end; floor levels are indicated (edited from [42]).

Table 2.7: Dimensions of the SGHWR bioshield [51].

Parameter	Value [m]
Wall thickness (Fuel Storage Pond)	1.22
Wall thickness (N, S)	1.56
Wall thickness (LSD plant room)	1.549 (2.82 total)
Height	6.99

2.10.2 Origin and Constraints on Radiological Inventory

64 The SGHWR bioshield inventory is based on two main sources:

- Radiological characterisation data of two cores taken through the bioshield [43], comprising concrete and limited rebar samples.
- Modelling of neutron activation of the concrete and rebar in the bioshield [44].

Characterisation Data

65 Current characterisation data for the bioshield comprises data from two cores analysed in 2005. One core was taken from Room 245, the Active Tools Store (ATS), and is near the mid-height of the reactor [43] and the other core was taken from the LSD plant room (Room 334) near the top of the reactor [45]. The cores were taken in different campaigns and have differing analytical suites. The ATS core comprises concrete and paint as well as three rebar samples, while the LSD core is limited to only concrete and

paint data. Data for the ATS core is summarised in Table **2.8** and Figure **2.8**, and for the LSD core in Table **2.9** and Figure **2.9**.

66 It was originally considered that the ATS core at the midpoint of the reactor would represent the position of highest neutron flux in the bioshield, although the subsequent neutron transport calculations undertaken in support of the activation modelling showed that there is in fact a higher neutron flux into the bioshield associated with the ion chamber basket penetrations and at the upper rim of the reactor due to neutron leakage between the radial and upper neutron shield tanks [44]. The LSD core samples the region of the bioshield impacted by the neutron leakage at the upper rim of the reactor and the data correspondingly reflects higher activation in the LSD core than the ATS core. The section of the bioshield near the ion chamber basket penetrations has not been characterised and the lack of data on the activation associated with these and other radial bioshield penetrations is an outstanding uncertainty in the bioshield inventory.

67 The data for some radionuclides, particularly ^{60}Co, ^{14}C, ^{41}Ca, ^{152}Eu and ^{154}Eu, indicate a clear decreasing trend in activity in concrete away from the inside surface of the bioshield to greater depths than typically expected by ingress of contamination from the surface. This is consistent with a source from neutron activation. Tritium also follows a decreasing trend, albeit with a differing slope. However, ^{3}H is known to be mobile in concrete and to have deeply penetrated the SGHWR structure from contamination sources and so its presence may not therefore be solely attributable to in-situ activation.

68 A number of other radionuclides, for example ^{40}K and ^{235}U, also show decreasing trends away from the reactor core but are reported at LOD. It is uncertain whether these are simply analytical artefacts or represent an increase in the concentration of radionuclides that are expected to be present naturally in concrete (Section 2.6). However, it is noted in the ATS core that low (0.51 – 2.4 Bq/g), but above LOD, results for Gross Alpha are reported for the innermost three concrete samples, covering 200 mm of core (samples from deeper depths are all LOD), suggesting that some elevated alpha-emitting radionuclides may be present.

69 The highest measured activity in the two cores is for ^{3}H in the painted surface of Room 213 (bioshield inner surface) with an activity of 27,000 Bq/g. Tritium in the concrete below this sample was lower (8,500 Bq/g), but still the second highest activity measured. Caesium-137 was also highly elevated in the inner painted surface of the LSD core at 850 Bq/g; however, it falls off rapidly to 14 Bq/g in the underlying concrete sample, consistent with a source from contamination.

70 After around 1,000 mm depth (from the inner bioshield surface) a clear activation trend in concrete becomes difficult to discern. In the LSD core this reflects an absence of data; however, in the ATS core activity levels reach the reported analytical limits of detection. Further into the core at just over 1,500 mm, a rise in activity occurs for some determinands that would not be expected due to activation. This appears to correspond to the location of a 1” ‘flexcell’ joint, denoted on Figure **2.8** and Figure **2.9**, which essentially separates the bioshield from the surrounding primary containment structure. In the ATS core a fibrous layer was found which is likely to represent the flexcell joint; this was removed and analysed separately by gamma spectrometry. This sample contained 120 (±10) Bq/g of ^{137}Cs and 3.3 (±0.2) Bq/g of ^{60}Co. It is likely that this joint acts as a pathway for low-level activity transport into the wall, hence disrupting the

activation trend. The elevated ^{137}Cs could potentially have arisen from liquid contamination leaks from the pipework at the top of the reactor, or from spills occurring during refuelling; alternatively, the result could be erroneous. As there is no further evidence for any mechanism, the lack of full understanding of the results from this layer is noted as an uncertainty in the bioshield characterisation data (INV-SGHWR-006).

71 Beyond the flexcell joint, ^{137}Cs, ^{60}Co and ^{3}H activity concentrations can be seen to increase again in the LSD core, but remain lower than the levels within the bioshield. This is likely to relate to ingress of contamination from the secondary containment, but the mechanism behind it is also noted as an uncertainty (INV-SGHWR-006).

72 To avoid interpretation of data below the limit of detection, a working assumption is that a circle with a diameter extending to the flexcell joint denotes the limit of activation (Figure **2.10**). Vertically the full bioshield height is assumed to be activated, although in practice the top and bottom parts of reactor bioshields typically have significant shielding and somewhat lower activities may be expected in these areas (Figure **2.7**; INV-SGHWR-004).

73 For the rebar samples little activity is reported (Table **2.8**). Tritium was detected in all three samples in a narrow range (0.72-1.2 Bq/g). This is not a contaminant expected to form in steel and could be associated with adsorption to the rebar surface. ^{60}Co (0.072 Bq/g) and ^{63}Ni (0.075 Bq/g) were also detected in the innermost rebar sample. A sole ^{55}Fe analysis was also undertaken for this sample, but was reported as an elevated limit of detection (<1 Bq/g). While the reported rebar activities are low, it is noted that the innermost rebar sample was taken from the concrete core sample from between 740 and 1280 mm from the inner bioshield surface and so it is likely that higher activities will be present in rebar closer to the reactor core.

Table 2.8: Analytical results (Bq/g) of an SGHWR bioshield core sample taken from the ATS and analysed in 2005 [43]. *(This page is set to print on A3.)*

Ref	GAU Ref	Media	Core Interval [mm]	Distance from Inner Surface [mm]	Specific Activity [Bq/g]																		
					Gross Alpha	Gross Beta	^{134}Cs	^{137}Cs	^{57}Co	^{58}Co	^{60}Co	^{241}Am	^{54}Mn	^{65}Zn	^{235}U	^{55}Fe	^{63}Ni	^{3}H	^{14}C	^{41}Ca	^{45}Ca	^{152}Eu	^{154}Eu
866	GAU548/1	Paint	Paint		<0.2	0.46	<0.06	0.5	<.2	<0.05	0.12	<0.06	<0.05	<0.2	<0.2			3.9				<0.8	<0.3
		Concrete	0-50	2655	<0.3	0.57	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			6.6				<0.04	<0.02
			50-150	2580	<0.3	<0.2	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.02	<0.01			6.4				<0.08	<0.03
			150-250	2480	<0.3	<0.3	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			4.4				<0.04	<0.02
866R	GAU548/2	Rebar	n/s		<0.2	<0.5	<0.03	<0.03	<0.06	<0.03	<0.06	<0.02	<0.03	<0.08	<0.06			0.72				<0.4	<0.1
867	GAU548/3	Concrete	250-410	2350	<0.3	<0.2	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.02	<0.01			0.24				<0.08	<0.02
			410-570	2190	<0.3	<0.2	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			0.24				<0.05	<0.02
			570-750	2020	<0.3	<0.2	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.02	<0.01			1.7				<0.08	<0.03
868	GAU548/4	Concrete	750-910	1850	<0.3	<0.3	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			1.1				<0.04	<0.02
			910-1070	1690	<0.3	<0.2	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.02	<0.1			5.2				<0.09	<0.03
			1070-1260	1515	<0.4	8.2	<0.01	6.1	<0.04	<0.01	0.057	<0.04	<0.01	<0.02	<0.04			46	< 0.06			0.045	<0.03
868R	GAU548/5	Rebar	n/s		<0.2	<0.5	<0.03	<0.02	<0.08	<0.02	<0.06	<0.03	<0.02	<0.07	<0.09			1.2				<0.3	<0.1
869	GAU548/6	Concrete	1260-1440	1330	<0.4	0.31	<0.01	0.017	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			57	0.08			<0.04	<0.02
			1440-1620	1150	<0.3	<0.2	<0.01	<0.01	0.003	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01			18				<0.05	<0.02
			1620-1800	970	<0.3	0.28	<0.01	<0.01	<0.01	<0.01	0.017	<0.01	<0.01	<0.03	<0.01			61				0.063	<0.03
			1800-1980	790	<0.3	0.56	<0.01	<0.01	<0.02	<0.01	0.16	<0.01	<0.01	<0.03	<0.02			330		< 0.4	< 0.2	0.3	<0.05
869R	GAU548/7	Rebar	n/s		<0.23	<0.45	<0.03	<0.03	<0.06	<0.03	0.072	<0.02	<0.03	<0.08	<0.06	<1	0.075	0.72				<0.4	<0.1
870	GAU548/8	Concrete	1980-2080	650	<0.4	4.5	<0.01	<0.01	<0.03	<0.01	0.85	<0.01	<0.01	<0.05	<0.04			630	0.85	0.63	< 0.1	1.5	0.1
			2080-2180	550	<0.4	8.6	<0.03	<0.03	<0.07	<0.03	2.1	<0.02	<0.03	<0.1	<0.08			730	0.99	n/a	n/a	3.5	0.27
			2180-2280	450	<0.4	30	<0.09	<0.07	<0.2	0.077	5.7	<0.04	<0.09	<0.4	<0.2			640	0.65	2.1	< 0.1	15	0.99
			2280-2380	350	<0.4	74	<0.1	<0.09	<0.2	0.19	17	<0.08	<0.09	<0.4	<0.3			1100	1.3			32	2.3
			2380-2480	250	0.51	120	0.17	<0.1	<0.3	0.36	27	<0.08	<0.2	<0.8	<0.3			1300	1.6	6.3	< 0.1	82	5.3
			2480-2580	150	0.74	240	0.13	<0.1	<0.4	0.74	63	<0.1	<0.2	<0.7	<0.4			1700	2.8	14	< 0.1	120	7.2
			2580-2680	50	2.4	450	<0.3	<0.3	<0.6	1.5	80	<0.2	<0.3	<2	<0.7			2500	4.3	23	< 0.1	260	19

Note: GAU report 548 states a fibrous layer was found at the top of this core (inferred as sample 869). This was removed and analysed separately by gamma spectrometry. This sample contained 120 (±10) Bq/g of ^{137}Cs and 3.3 (±0.2) Bq/g of ^{60}Co.
n/s – not stated. Results for a number of naturally-occurring radionuclides at LOD levels have been omitted from the table.

Table 2.9: Analytical results (Bq/g) of an SGHWR bioshield core sample taken from the LSD plant room and analysed in 2005 [43]. *(This page is set to print on A3.)*

Magnox Ref WA/SAMPLE/	**GAU Ref**	**Media**	**Core Interval [mm]**	**Distance from Inner Surface [mm]**	**^{134}Cs**	**^{137}Cs**	**^{57}Co**	**^{58}Co**	**^{60}Co**	**^{241}Am**	**^{54}Mn**	**^{65}Zn**	**^{235}U**	**^{3}H**	**^{152}Eu**	**^{154}Eu**
958	GAU626/2	Paint	Paint Initial	2811	<0.090	0.54	<0.14	<0.083	0.33	<0.043	<0.083	<0.26	<0.15	73	0.32	<0.40
		Concrete	0-50	2786	<0.006	0.17	<0.011	<0.005	<0.009	0.005	<0.005	<0.014	<0.011	32	<0.067	<0.024
			50-100	2737	<0.004	0.006	<0.011	<0.003	0.031	<0.004	<0.003	<0.009	<0.012	47	<0.046	<0.019
			100-150	2687	<0.009	0.019	<0.013	<0.008	<0.016	<0.004	<0.008	<0.024	<0.014	45	<0.11	<0.037
			150-350	2562	<0.007	0.007	<0.010	<0.006	0.007	<0.003	<0.006	<0.018	<0.011	32	<0.087	<0.031
			350-550	2362	<0.005	0.006	<0.015	<0.004	0.008	<0.005	<0.004	<0.010	<0.016	5	<0.052	<0.025
			2260-2460	1469	<0.15	<0.12	<0.25	0.19	21	<0.071	<0.14	<0.76	<0.27	840	45	3
			2460-2660	252	<0.34	<0.29	<0.81	1	110	<0.25	<0.32	<1.5	<0.86	2700	180	13
			2660-2710	127	<0.64	<0.49	<1.1	1.9	140	<0.31	<0.61	<3.1	<1.2	4500	370	34
			2710-2760	77	0.24	0.5	<1.0	1.9	140	<0.31	<0.39	<1.8	<1.1	5700	280	28
			2760-2810	27	<0.56	14	<1.4	1.8	170	<0.42	<0.49	<2.4	<1.5	8500	310	29
		Paint	Paint Final	1	<6.4	850	<11	<5.7	170	<3.3	<5.5	<25	<12	27000	200	36

Note: Results for a number of naturally-occurring radionuclides at LOD levels have been omitted from the table.

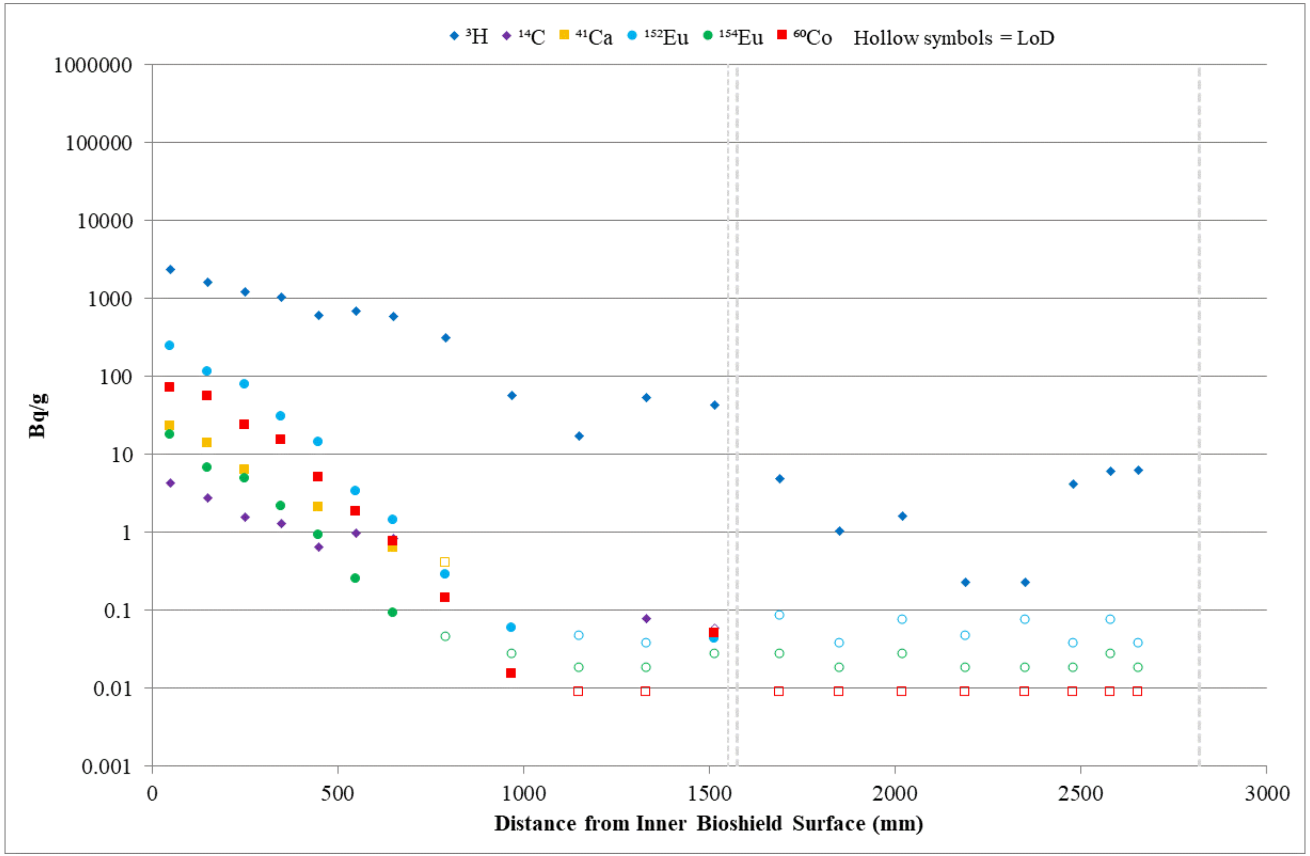

Figure 2.8: Profile of selected 2005 radionuclide data from the ATS concrete core through the SGHWR bioshield and primary containment. Based on structural drawings the two vertical grey lines at ~1,550 mm have been added to represent a 1 inch 'flexcell' joint separating the two structures; similarly, the grey line at 2,819 mm represents the outer wall of the primary containment. However, the reported ATS core length is only 2,680 mm. A fibrous layer believed to be the flexcell joint was found in the ATS core and analysed for ^{60}Co and ^{137}Cs. The ^{60}Co has been added to this plot and is represented by the point at 1,515 mm and ≈ 6 Bq/g; as such, a slight offset is evident. This has been accounted for in calculations.

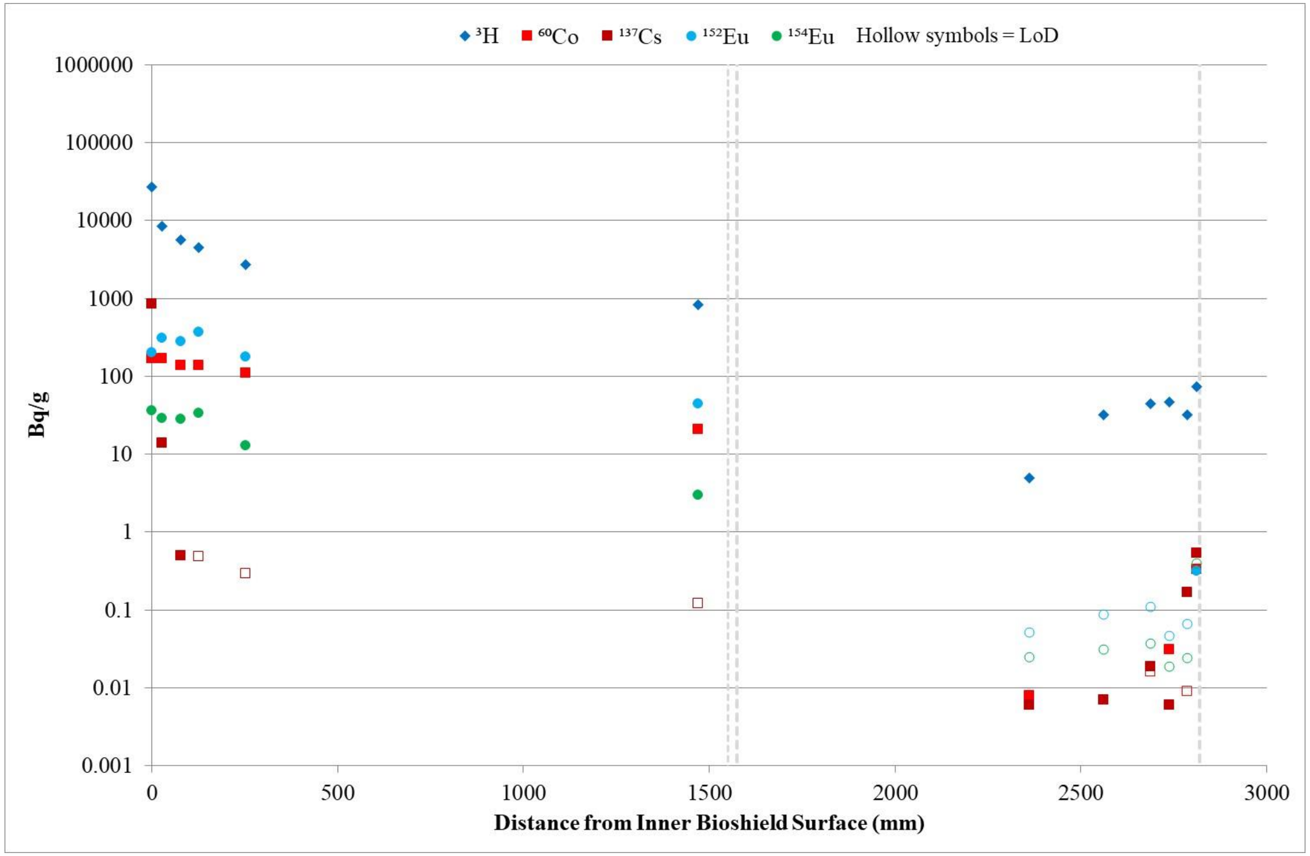

Figure 2.9: Profile of selected 2005 radionuclide data from a concrete core from the LSD plant room through the SGHWR bioshield. Based on structural drawings the two vertical grey lines at ~1,550 mm have been added to represent a 1 inch ‘flexcell’ joint separating the two structures; similarly, the grey line at 2,819 mm represents the outer wall of the primary containment. The reported core length of 2,810 mm is a close fit to the drawings.

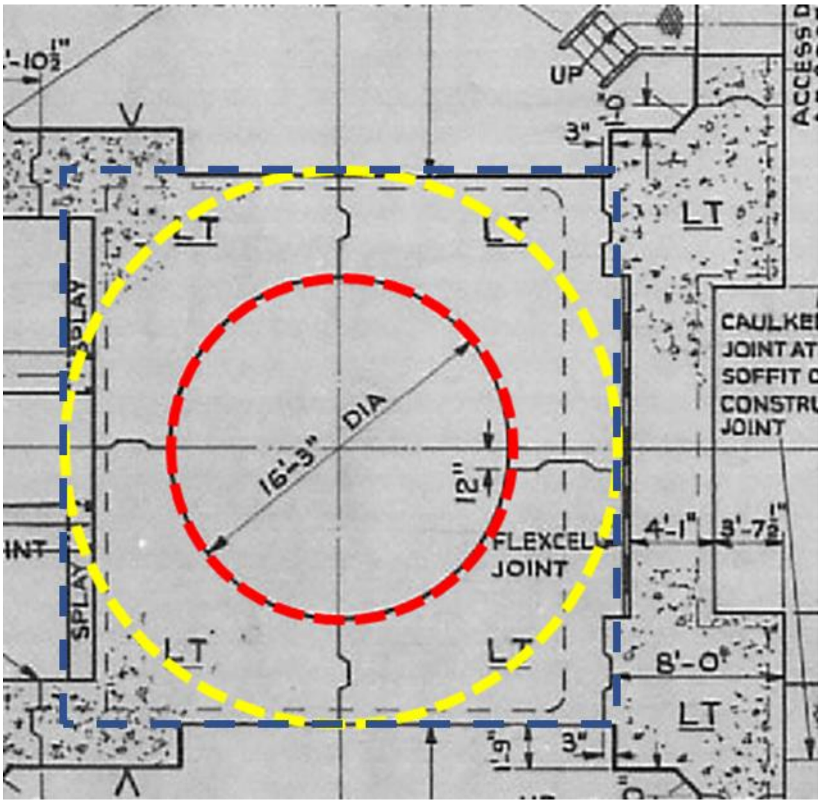

Figure 2.10: Top-down view of the SGHWR bioshield showing inferred zone of bioshield activation between the core (red circle) and 'flexcell' joint (outside of the yellow circle). The blue box indicates the simplified bioshield outer dimensions adopted for contamination estimation. Bioshield plan from [46].

Neutron Activation Modelling for Primary Bioshield

74 Hertel [44] reports the results of 3D neutron activation modelling of the components within, and including, the calandria as well as the primary system shielding surrounding the reactor (including the concrete bioshield and rebar). The calculations were undertaken to support waste classification and quantification decisions. Neutron fluxes were assessed using the MCNP® code [47] and the fluxes were then input into the EASY-2003 activation code package [48], along with estimates of material compositions and the operating history of the SGHWR.

75 In the absence of real SGHWR material compositions, the major element concrete composition used in the MCNP® modelling was taken from the U.S. National Institute for Standards and Technology (NIST) [49]. The bioshield concrete trace element composition was taken from [50], a U.S. Nuclear Regulatory Commission concrete specification based on data from American nuclear plants. The potentially strong geographic variation in concrete composition is highlighted as a significant contributing uncertainty with this activation data (INV-SGHWR-005). For example, it is acknowledged that the Ca content could be higher than assumed in the activation modelling if the Winfrith bioshield concrete used a limestone aggregate, as it would then be present in the aggregate as well as the cement. However, the fact that the

modelled ^{41}Ca activities are higher than those measured in the bioshield cores (Figure **2.11**) indicates that this is unlikely to be the case.

76 The bioshield was split into a number of vertical and radial flux intervals. Activities were derived at four radial intervals of the bioshield based on the flux averaged over the full height of the bioshield and based on the vertical interval with the largest flux. The average and maximum predicted activation product levels in January 2006 are given in [44, Tab.7-29 and Tab.7-30] for the bioshield concrete and in [44, Tab.7-31 and Tab.7-32] for the rebar. A total of 128 isotopes are predicted to be formed in the activation modelling, although the majority contribute insignificant amounts of activity.

77 The uncertainties associated with use of the activation model are summarised in INV-SGHWR-005.

Comparison of Modelling and Characterisation Data

78 Selected results for both analysed and modelled bioshield concrete data are compared in Figure **2.11** as well as data for ^{55}Fe, which is an example of a radionuclide with only modelled data. The results indicate that the modelling overestimates, to variable degrees, the levels of radionuclides present in the concrete core samples except for ^{137}Cs, which underestimates the activities. While the model may predict some ^{137}Cs formation due to activation, as discussed above, the proximity of ^{137}Cs with surfaces or joints is more consistent with a source of ^{137}Cs from ingress of fission product contamination. Caesium-137 is the contaminant which may be most easily identifiable as sourced through surface contamination. To account for the presence of any co-contaminants in the measured dataset, the SGHWR Primary External Contamination fingerprint (FP-028; see Section 2.10.4) was applied (based on the ^{137}Cs activity) to the concrete associated with both the inner and flexcell-related concrete contamination. Conservatively, this may result in a small amount of double counting of activity, where both activation and contamination are co-located.

79 The discrepancy between modelled and analysed results for most radionuclides is commonly several orders of magnitude and as such the direct use of the modelling to supply activity concentrations in the inventory calculations is not justified. The discrepancy is assumed to arise because the activation modelling made assumptions about the material compositions present (e.g. using concrete compositions for US nuclear plants rather than SGHWR), which would lead to different proportions of activation products due to differences in the trace elements present in geographically varying concrete. However, the activation model results do predict the presence of a number of radionuclides not captured in the analytical results (e.g. ^{55}Fe). As such, in order to provide additional indicative values for these radionuclides, the modelled data were used to develop a bioshield concrete activation fingerprint [51].

80 Only limited rebar results are available for the bioshield and, of the three samples, two were taken from outside of the flexcell joint (the assumed maximum activation extent). Measured and modelled values for ^{60}Co and ^{63}Ni are presented in Figure **2.12**. The very limited data suggest, as per the concrete results, that the model over-estimates the activity concentrations. However, as so little rebar data is available, a fingerprint was developed directly from the modelled data ([51]; INV-SGHWR-006).

81 Fingerprints are derived for the bioshield concrete and rebar based on the activation modelling. Isotopes are screened to eliminate those which are stable, have half-lives of

less than one year, or contribute less than 0.01% to the total fingerprint activity. The screening exercise reduces the 128 isotopes predicted in the modelling to a set of 13 radionuclides in the concrete fingerprint and 6 radionuclides in the rebar fingerprint. The eliminated radionuclides account for 0.03% of the total activity for the concrete fingerprint and 0.02% of the total activity for the rebar fingerprint.

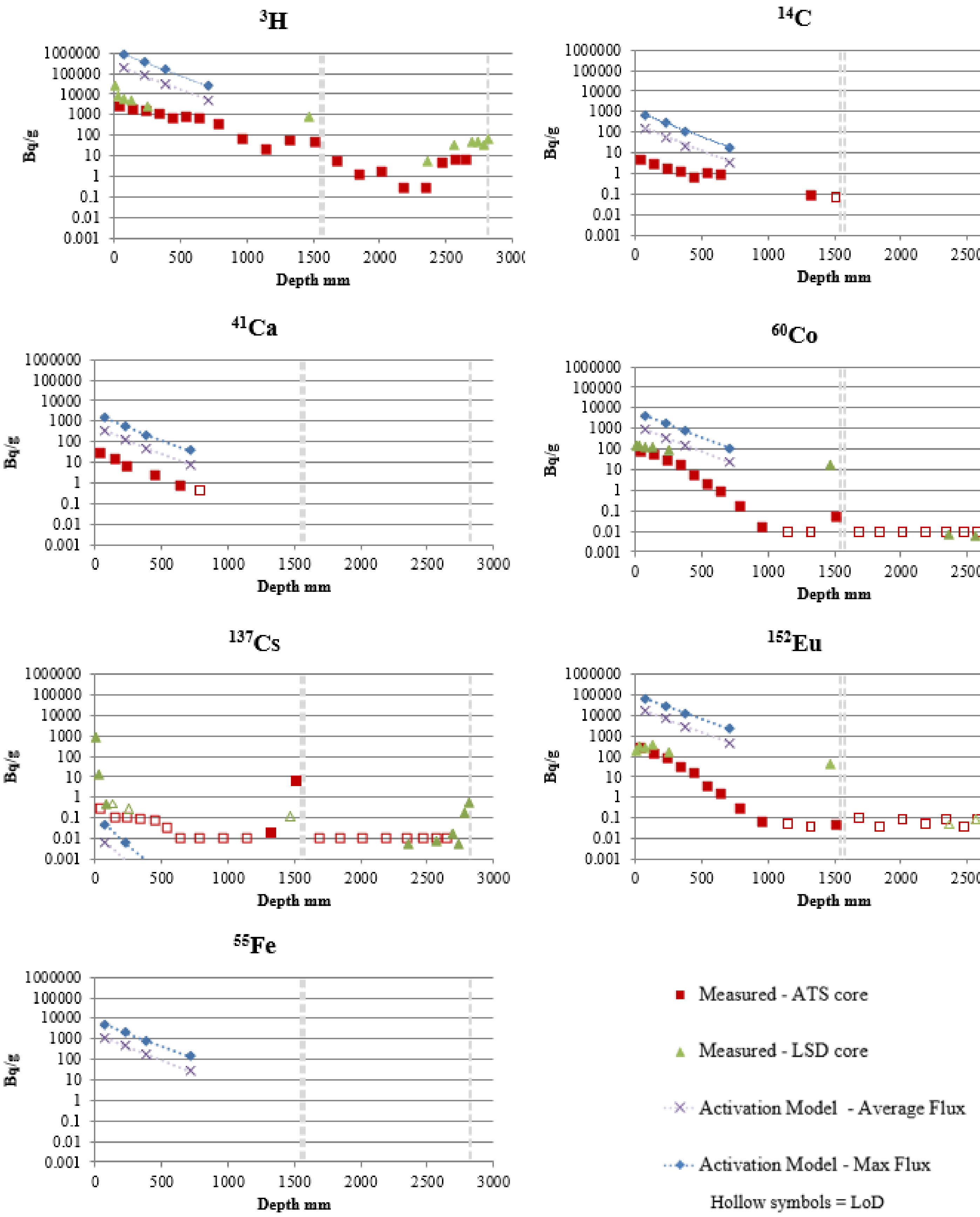

Figure 2.11: Comparison of analysed and modelled bioshield concrete radionuclide activity data. Measured data have been decay corrected to January 2006 in order to be compared to the modelling data.

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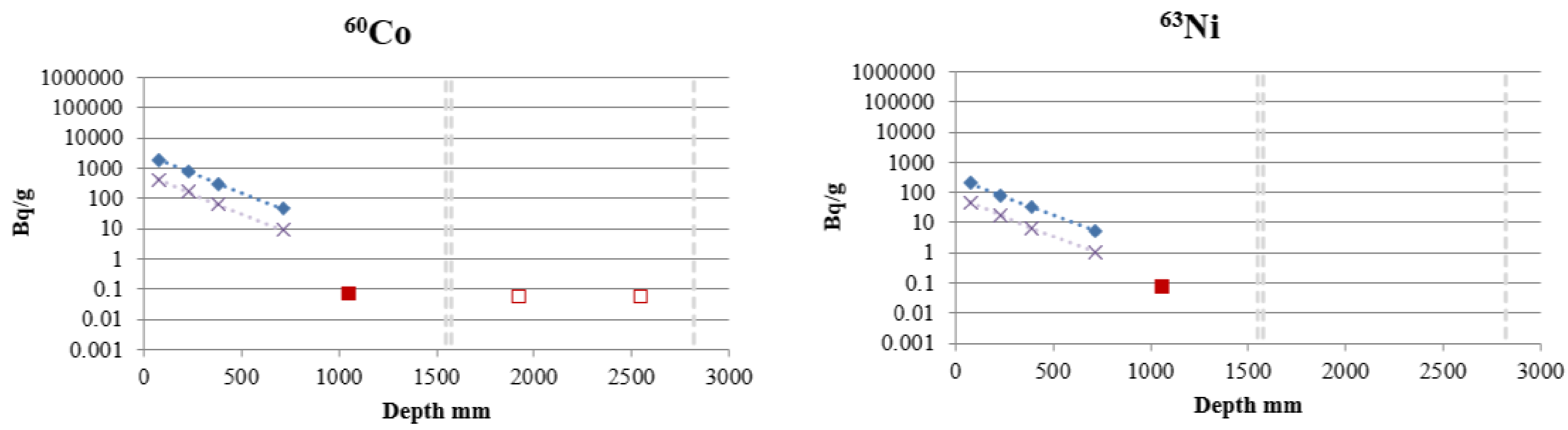

Figure 2.12: Comparison of analysed and modelled bioshield rebar radionuclide activity data. Measured data have been decay corrected to January 2006 in order to be compared to the modelling data. Key as for Figure **2.11**.

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2.10.3 Bioshield Inventory Estimate

83 The bioshield inventory was derived assuming the following four contributing components:

- Activated concrete.
- Activated rebar.
- Paint.
- Contaminated concrete.

84 The volume of activated concrete was derived assuming an annular cylinder of concrete the full height of the bioshield, with an inner surface corresponding to the inner face of the bioshield and outer diameter corresponding to the location of the flex cell joint (resulting in the area between the red and yellow circles in Figure **2.10**). The activation inventory was derived from the maximum and average measured radionuclide activities in segments of the bioshield cores deeper than the flex cell joint. Missing radionuclides were scaled from the activation modelling based on the measured ^{60}Co activity. This is conservative as samples are disproportionately concentrated towards the centre of the bioshield where the activation is highest.

85 The quantity of activated rebar is not accurately known. An assumption of 3% by volume of the activated concrete was assumed consistent with common proportions in structural concrete (the concrete inventory conservatively neglects the volume taken up by rebar). Only a single rebar sample is available within the assumed zone of activation and its precise location within the bioshield is not known, although it was at least 790 mm from the inner surface of the bioshield. All analytes in the rebar samples are at or near the LOD. Consequently, the activities were derived entirely from the SGHWR rebar activation fingerprint [44]. The rebar fingerprint is scaled by the discrepancy between the observed and measured concrete activation activities for ^{60}Co.

86 A simplified geometry was adopted for paint and contamination inventory estimation which treats the bioshield as an 8.05 x 8.05 x 6.99 m cuboid with a cylindrical cut-out for the reactor passing through the centre of the square faces. The mapping of the simplified geometry onto the plan of the bioshield is shown in Figure **2.10**. This approximation neglects the construction joints at the corners of the bioshield, although this is offset by the assumption that surface is available for contamination where the bioshield integrates with the primary containment.

87 The paint volume was derived assuming paint covers all of the surfaces of the simplified bioshield geometry except the east and west faces, as these correspond to where the bioshield joins with the primary containment and as such are not painted. Activity values are derived from the measured specific activities of the inner surface paint sample in the LSD core (as no paint sample from the inner face of the bioshield was present in the ATS core). The activation and contamination components are not separated in this case as the sample data are likely to be representative of all sources of activity in the paint.

88 The contaminated concrete volume is derived by assuming contamination extends to a depth of 20 mm from all surfaces of the simplified bioshield geometry. Unlike for paint, all surfaces are included in the volume estimation as the contamination in the flexcell joint demonstrates that contamination can be expected where the bioshield

connects to the primary containment, as well as surfaces accessible to the primary containment atmosphere. The contamination is derived by scaling the activities of FP-028 to the measured ^{137}Cs concentration in the concrete samples where contamination is observed (the core sample adjacent to the flexcell and the core sample adjacent to the inner surface of the bioshield). The sample of flexcell material is also pessimistically included in the average and maximum activity calculations.

89 Maximum and average activity concentrations and an estimate of the radioactive inventory are presented in Table **2.10**, based on the data and approach described in Section 2.10.2, [51] and [14].

Table 2.10: Estimated SGHWR bioshield disposal inventory, including maximum and average activity concentrations and inventory based on average activity concentrations, presented for an inventory reference date of 01/01/2027. 100% of this activity will be disposed of in-situ.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]
^{3}H	8.02E+03	4.03E+02	3.08E+05
^{14}C	4.29E+00	1.10E+00	8.42E+02
^{134}Cs	4.57E-04	8.50E-05	6.50E-02
^{137}Cs	5.18E+02	8.15E+00	6.23E+03
^{60}Co	9.96E+00	2.03E+00	1.55E+03
^{241}Am	3.19E+00	8.17E-02	6.25E+01
^{152}Eu	1.23E+02	2.43E+01	1.86E+04
^{154}Eu	6.31E+00	1.08E+00	8.24E+02
^{155}Eu	2.16E-01	4.23E-02	3.24E+01
^{55}Fe	8.06E+01	2.32E+00	1.77E+03
^{63}Ni	6.36E+01	9.29E+00	7.11E+03
^{90}Sr	8.18E-02	1.22E-03	9.34E-01
^{241}Pu	7.11E-02	1.06E-03	8.11E-01
^{133}Ba	2.60E+01	5.11E+00	3.91E+03
^{235}U	1.20E+01	3.06E-01	2.34E+02
^{238}Pu	7.64E-03	1.14E-04	8.72E-02
^{239}Pu	1.53E-03	2.29E-05	1.75E-02
^{240}Pu	1.53E-03	2.28E-05	1.75E-02
^{39}Ar	1.10E+01	2.17E+00	1.66E+03
^{41}Ca	2.30E+01	5.37E+00	4.11E+03
^{113m}Cd	6.17E-01	1.64E-02	1.25E+01
^{151}Sm	2.04E+01	4.01E+00	3.07E+03
^{204}Tl	6.11E-01	1.62E-02	1.24E+01
Total	**8.92E+03**	**4.69E+02**	**3.58E+05**

2.10.4 Sensitivity Analysis and Further Characterisation

90 There is considerable uncertainty in the estimate of the bioshield inventory. In particular, the sampling of the bioshield, consisting of two cores, is insufficient to consider it fully characterised. The primary risk is that the cores do not capture the most activated portion of the bioshield and may therefore be underestimating the activation inventory. However, the ATS core was drilled into the area of expected highest neutron flux at the reactor mid-point and the LSD core was drilled into the area of highest flux predicted by the activation modelling (the higher measurements for the LSD core compared to those of the ATS core support this modelling prediction). Thus, measurement of substantially higher activity in any additional cores is not expected. In addition, the assumed depth of activation (1,500 mm from the core) over which the average activity concentration is applied to calculate the total inventory is conservative.

91 The uncertainty relating to the contamination contribution to the bioshield inventory is considered negligible as the contamination makes up less than 2% of the overall bioshield average inventory estimate, and the paint contribution is less than 0.5%. The geometry assumptions used in the calculation are also conservative.

92 There is uncertainty in the amount of rebar present, with 3% by volume assumed, and the rebar activities are also not underpinned by any sample data from the rebar. The rebar content may be greater than 3%, but the current activity contribution is only 0.5% of the overall bioshield average inventory estimate. In addition, the calculation is conservative in that the bioshield volume occupied by the concrete is not reduced to account for the displacement by the rebar, but is effectively double-counted when calculating the activity.

93 The uncertainty in the bioshield inventory can be addressed by making alternative assumptions and exploring the effect on the calculated inventory. Two possible approaches have been considered: i) applying the maximum activity concentration to the entire activated bioshield, and ii) scaling measured core activities into line with the (much higher) activation modelling activities. The second of these approaches has been shown through application to be the more conservative (resulting in a factor of 14.6 increase in the bioshield inventory compared to a factor of 4.2 increase), so this approach has been adopted and is explained in detail below.

94 In order to derive this alternative inventory, the adopted activation inventory activities have been scaled by a factor equal to the ratio between the measured and modelled ^{41}Ca activities extrapolated to the inner face of the bioshield. Calcium-41 is chosen as a marker because calcium is a key component of Portland cements and should therefore be among the most abundant elements in the bioshield. Of the radionuclides analysed for in the cores, calcium is also the element that comprises the highest mass fraction of the concrete composition assumed in the activation modelling. It should therefore provide a better marker to scale to than radionuclides which arise from the activation of ‘trace’ elements as these will be subject to greater variation. The extrapolated ^{41}Ca activity on the inner face of the bioshield is 33.4 Bq/g for the measurement data and 496.8 Bq/g for the modelled activation. The scaling factor for the alternative activation inventory is therefore 14.9 (for both concrete and rebar components).

95 This alternative inventory for the bioshield is presented in Table **2.11**. The average activity of the bioshield at the inventory reference date is 6,800 Bq/g, which is almost

entirely tritium (a beta emitter). The overall activity of the alternate, more conservative, bioshield inventory therefore is still below the upper threshold of low-level waste (LLW) in the UK, which is 4 GBq/tonne (4,000 Bq/g) alpha and 12 GBq/tonne (12,000 Bq/g) beta/gamma.

Table 2.11: Alternative SGHWR bioshield activation inventory, presented for an inventory reference date of 01/01/2027.

Radionuclide	Alternative Bioshield Average (Bq/g)	Alternative Bioshield Disposal Inventory (MBq)
^{3}H	5.97E+03	4.56E+06
^{14}C	1.62E+01	1.24E+04
^{134}Cs	1.25E-03	9.55E-01
^{137}Cs	1.41E+01	1.08E+04
^{60}Co	3.01E+01	2.30E+04
^{241}Am	1.20E+00	9.20E+02
^{152}Eu	3.61E+02	2.76E+05
^{154}Eu	1.60E+01	1.22E+04
^{155}Eu	6.29E-01	4.81E+02
^{55}Fe	3.45E+01	2.64E+04
^{63}Ni	1.34E+02	1.02E+05
^{90}Sr	1.22E-03	9.34E-01
^{241}Pu	1.06E-03	8.11E-01
^{133}Ba	7.59E+01	5.80E+04
^{235}U	4.52E+00	3.45E+03
^{238}Pu	1.14E-04	8.72E-02
^{239}Pu	2.29E-05	1.75E-02
^{240}Pu	2.28E-05	1.75E-02
^{39}Ar	3.22E+01	2.46E+04
^{41}Ca	7.99E+01	6.11E+04
^{113m}Cd	2.43E-01	1.86E+02
^{151}Sm	5.96E+01	4.56E+04
^{204}Tl	2.41E-01	1.84E+02
Total	**6.83E+03**	**5.22E+06**

96 As discussed later in Section 2.18, the bioshield is the single most significant contributor to the SGHWR end state inventory estimate, comprising almost 60% of the total activity. Therefore, any additional characterisation that can be undertaken following the core segmentation and post-operational clean-out of the area would be beneficial. However, it is expected that any new bioshield inventory estimate underpinned by such data would be bounded by the alternative estimates presented above. Both the reference and alternative inventories will be considered in the radiological PA.

2.11 Mortuary Tubes

2.11.1 Feature Description

97 Ten storage locations for irradiated items were provided in the construction phase of the SGHWR Primary Containment [52], which are referred to as the mortuary tubes. Vertically cast into the bioshield, each tube consists of a ‘cast-in’ liner measuring approximately 9” (0.23 m) diameter by 32’ (9.8 m) long. The tops of the mortuary tubes are sited at the 132’ 10” AOD level, at the top of the bioshield. The tubes run the full 23’ (7.0 m) height of the bioshield and extend a further 9’ (2.7 m) vertically down into the east wall of the primary containment under the reactor bioshield. The lower end of each tube is fitted with a 90° bend that exits into area 111 at 100’ 9” AOD. Based on the inner diameter (0.235 m), outer diameter (0.244 m) and length (9.754 m), the total volume of the metal of a single mortuary tube liner is estimated to be 0.035 m^3, which equates to a total steel mass of 275 kg [51].

98 The mortuary tube positions are identified as Z1 to Z10, where Z1 is located furthest south and Z10 furthest north. A plan view of the locations of the mortuary tubes on the top face of the bioshield is shown in Figure **2.13**. The mortuary tube liners are between 509 mm from the inner edge of the bioshield at the closest point and 1,265 mm at the farthest point, which means they are all within the radius of activation considered in Section 2.10.

99 Nine of the ten mortuary tubes contain stored items that have yet to be removed. Although the items will be removed during decommissioning and the tubes cleaned, it is expected that residual radioactivity will remain within the tube structures. Therefore, an estimate of the residual activity that may remain in the tubes is included here, with a fingerprint based on the previously stored items. However, this is considered to be an indicative estimate as there is uncertainty regarding the degree of cleaning that can be achieved and there is uncertainty as to the tube contents in some cases, due to the quality of the historical documentation and inability to access the contents at this stage [52]. A photograph of the tops of the mortuary tubes taken in August 2015 is presented in Figure **2.14**.

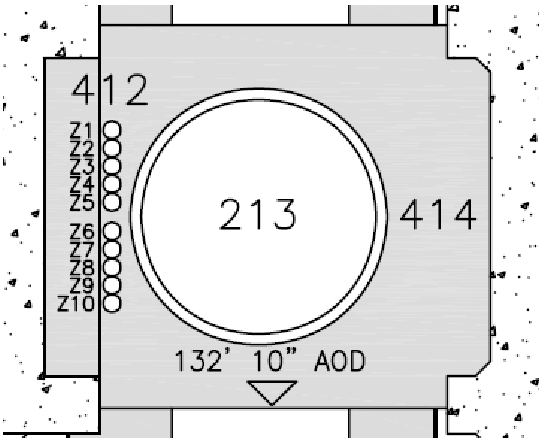


Figure 2.13: Plan view of the top of the bioshield (grey) showing the locations of the ten mortuary tubes (circles labelled Z1-10 on left of Figure) relative to the location of the reactor core (large double circle labelled 213) [53, Sheet 5].

Figure 2.14: Image of the tops of the mortuary tubes taken in August 2015. Image looks north (closest visible mortuary tube is Z1); the top of the reactor is on the left. Mortuary tube Z3 is covered by a yellow square lead-filled blanket whose purpose is to shield the current contents; this will be removed as part of decommissioning [52].

2.11.2 Origin and Constraints on Radiological Inventory

100 Radiological data for the tubes consists only of dose measurements taken either when the contained items were emplaced or during a survey with a probe at the top or lowered into a tube as access allowed. The current understanding of the mortuary tube contents is recorded in Table **2.12**. It is understood that all items stored in the mortuary tubes are components of the reactor, with the exception of the three cans in Z5 that contain solid ILW debris recovered during final cleaning operations on the ponds.

101 During reactor decommissioning the stored items in the mortuary tubes will be removed and the tubes will be cleaned. It is expected that the metal tube liners will remain in-situ, as they are cast into the bioshield. The degree and location of the activity that could remain in the tubes, in terms of contamination and activation of the tubes themselves or loose contamination released from the stored items, is unknown. Therefore, a high-level conservative estimate has been developed that assumes there are five potential sources for the residual activity in the mortuary tubes based on the items stored and the location of the tubes:

- Contamination carried over from items that came from the reactor core.
- Contamination carried over from items that may have been in contact with the moderator circuit.

- Contamination carried over from items that came from the ponds.
- Contamination arising from the degradation of activated stored items.
- Activation of the metal tubes themselves due to reactor neutron flux in the bioshield.

102 The potential activity from each source is considered separately and the total inventory is derived from the sum of all potential sources of activity. Each of the contamination activities is derived by scaling a suitable fingerprint to an activity which is expected to be limiting of the contamination present in the mortuary tubes.

Table 2.12: Understanding as of July 2016 of the contents of the SGHWR mortuary tubes. Summarised from [52].

Mortuary tube	Item	Mass [kg]	Dose [mSv/h]	Year of dose measurement
Z1	RIG C removed from the reactor core, circa 31’ (9.4 m) long	60	>500	1994
	RIG A removed from the reactor core, circa 25’ (7.6 m) long	55		
	RIG C Liner Support tube, circa 14’ 9” (4.5 m) long	30		
Z2	RIG B, circa 26’ (7.9 m) long	60	668	2004
Z3	None / believed to be empty		32.25	2004
Z4	Unknown item on end of extension bar	>125	300	2003
Z5	Believed to be a can containing channel tube stool from reactor grid position U09	Unknown	1300	1994
	Three cans containing solid ILW from the ponds	Unknown	189.2	2004
Z6	Remaining lengths of niobium channel tube	137	295.5	1969
Z7	Remaining lengths of niobium channel tube	137	270.0	1969
Z8	Remaining lengths of niobium channel tube	137	202.5	1969
Z9	Remaining lengths of niobium channel tube	137	295.5	1969
Z10	4.3 m long can containing ILW reactor components, contents uncertain	109	25000	1994

Reactor Core / Primary Circuit Contamination

103 It is unlikely that any contamination of the mortuary tubes from stored items that were retrieved from the reactor core would exceed that of the primary circuit pipework. Thus, a conservative activity estimate was derived by assuming the primary circuit pipework fingerprint [25, Tab.10] scaled to an activity of 200 Bq/g, which is equal to

the lowest primary circuit pipe activity measured for the contaminated run (between the primary manifold and chemical clean) [25, §6.1.2]. A maximum activity of 400 Bq/g was assumed based on the maximum activity of primary circuit pipework [25, §6.1.2]. A reference date of 14/09/2015 was adopted for this contribution. As it is unknown whether contamination is loose or fixed, it is all conservatively assumed to be present after cleaning (the same applies for the moderator circuit and ponds contamination components). The results of the radiological PA, using this preliminary inventory as a first pass input, will determine the extent of decontamination needed. Future optimisation work will determine whether such decontamination is feasible or whether a case could be made for leaving the activity in-situ.

Moderator Circuit Contamination

104 It is assumed that any contamination of the mortuary tubes from stored items that were in contact with the moderator circuit will not exceed that of the moderator circuit pipework itself. Therefore, a conservative activity estimate was derived by assuming the moderator circuit fingerprint (FP-030) scaled to 322 Bq/g, which is the activity for the section of the moderator circuit between the reactor and the clean-up plant. Similarly, the maximum activity estimate was derived by scaling FP-030 to 650 Bq/g, the maximum activity reported for the moderator circuit pipework. A reference date of 14/09/2015 was adopted for this contribution.

Ponds Contamination

105 Consistent with the preceding contribution estimates, it is assumed that the activity of the mortuary tubes arising from stored items retrieved from the ponds will not exceed that of the ponds' liners. It is acknowledged that, given the diluting effect of pond water, only a fraction of the activity arising from highly active items placed within the ponds would have ended up on the pond wall; however, in the absence of any other information this assumption is believed to be appropriate for a first pass inventory estimate to be used as an input to the PA, the results of which will then inform future characterisation and clean-up priorities and requirements. The possibility of a significantly higher residual inventory in the mortuary tubes is considered in the sensitivity analysis (Section 2.11.4). The ponds contamination activity estimate was derived by assuming the ponds fingerprint (FP-034) with the maximum and average activity of the fuel pond liner (respectively 11,340 Bq/g and 3,289 Bq/g). A reference date of 14/09/2015 was adopted.

Degradation of Activated Stored Items

106 With the exception of the ILW items from the ponds in tube Z5, which are contained in three cans, the known contents of the mortuary tubes are all reactor components and so are expected to be activated. However, the degree of activation and the amount of material that may have degraded and contaminated the tubes is unknown. Therefore, an indicative estimate has been derived. The SGHWR activation modelling study [44, Tab.7-11] calculates an average activation activity of 58.8 MBq/g for SGHWR Zircaloy fuel channel tubes, which has been assumed to be representative of reactor core items. If it is arbitrarily assumed that 100 g of activated metal separates from stored items and remains in each mortuary tube following cleaning, and that the contamination is uniformly dispersed throughout the tube, the activity of the liner would be 21,400 Bq/g. An equivalent estimate for the maximum activity was derived using

the maximum flux Zircaloy fuel channel tubes activation, resulting in an activity of 23,000 Bq/g. It is acknowledged that the assumption of 100 g of activated material remaining following cleaning cannot be underpinned. However, for a first pass inventory estimate to be used as an input to PA, the results of which will then inform future characterisation and clean-up priorities and requirements, it is believed to be appropriate. The possibility of a significantly higher residual inventory in the mortuary tubes is considered in sensitivity analysis (Section 2.11.4).

107 The bioshield rebar activation fingerprint at 01/01/2006 was applied to the calculated activity concentrations. The different alloys used for reactor components and higher neutron fluxes in the reactor will result in a different activation profile to that calculated for the rebar. However, given the large uncertainties in the derived activities, as well as in the current and historical mortuary tube contents, derivation of a new fingerprint for this indicative inventory estimate was not considered meaningful. This could be carried out in the future if PA results indicate that the mortuary tubes warrant more detailed characterisation.

Activation of Mortuary Tube Liners

108 The activation of the mortuary tube liners themselves by neutrons from the reactor was calculated using the average and maximum flux activities of bioshield rebar in the 18” to 38” radial interval (respectively 3,790 Bq/g and 18,600 Bq/g) from the SGHWR activation modelling study [44, Tab. 7-31 and 7-32]. The metal of the mortuary tube liners is all in the radial interval 20” to 50” from the inner edge of the bioshield so the neutron flux at this depth should be appropriate. The bioshield rebar activation fingerprint is applied for a reference date of 01/01/2006.

2.11.3 Inventory Estimate

109 The overall inventory estimate assumes that the activities from all five sources apply uniformly to all of the steel in the vertical segment of all ten mortuary tube liners. This assumption pessimistically neglects the limited pathway for certain sources of contamination (e.g. only one tube is known to currently contain waste from the ponds) and the fact that only the segment of the tubes within the bioshield would be activated. The potential for contamination to pool at the elbow bend at the bottom of the mortuary tubes is neglected; it is assumed that any accumulation of loose contamination at the bend would be removed when the tubes are cleaned.

110 The total mass of all ten mortuary tube liners is estimated to be 2,750 kg based on the known dimensions of the liners and the density of steel from Table **2.6**. Each of the inventory contributions are decayed to a reference date of 01/01/2027 and added together to obtain the total disposal inventory. Table **2.13** records the average activity concentration at the original reference date for each contribution and at the reference date for the inventory estimate. Due to the substantial amount of short-lived activation products (particularly ^{55}Fe) in the rebar activation fingerprint, the activity contributions due to direct activation and stored item activation contamination are significantly reduced by the inventory reference date.

111 The overall activity concentration of the mortuary tube liners at the reference date is 2,945 Bq/g, which equates to a disposal inventory of 8,096 MBq. The activity is dominated by the assumed contribution from the ponds, which is conservatively assumed to be present in all tubes although only one tube is known to contain ponds

waste. The maximum and average activity concentrations for each radionuclide and the estimated radioactive inventory for the mortuary tubes are presented in Table **2.14**.

Table 2.13: Decayed average activity concentration and disposal inventory contributions of activity sources for the SGHWR mortuary tubes at the inventory reference date of 01/01/2027 [51].

	Original activity concentration [Bq/g]	Decay time to reference date [y]	Decayed activity concentration [Bq/g]	Disposal Inventory [MBq]
Primary circuit contamination	200	11.3	130	357
Moderator circuit contamination	322	11.3	179	492
Ponds contamination	3,290	11.3	2,500	6,874
Stored item contamination	21,400	21.0	116	318
Direct activation	3,790	21.0	21	40
		Total:	2,945	8,096

Table 2.14: Estimated SGHWR mortuary tubes in-situ disposal inventory, including maximum and average activity concentrations, and a total inventory based on the average activity concentration and the rebar activation fingerprint. All data are presented for an inventory reference date of 01/01/2027. Although conservative, this inventory is considered to be preliminary because only limited data are currently available.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]
^{3}H	3.24E+02	1.61E+02	4.43E+02
^{14}C	3.48E+01	1.67E+01	4.61E+01
^{137}Cs	2.94E+03	8.54E+02	2.35E+03
^{57}Co	3.50E-06	1.73E-06	4.76E-06
^{60}Co	4.75E+01	2.44E+01	6.70E+01
^{241}Am	1.63E+02	4.75E+01	1.31E+02
^{94}Nb	1.46E+00	4.90E-01	1.35E+00
^{125}Sb	3.19E-02	1.59E-02	4.38E-02
^{152}Eu	1.60E+00	7.94E-01	2.18E+00
^{154}Eu	6.18E+00	1.84E+00	5.06E+00
^{155}Eu	4.63E-01	1.39E-01	3.83E-01
^{55}Fe	1.12E+02	1.23E+02	3.39E+02
^{63}Ni	2.98E+02	1.40E+02	3.85E+02
^{90}Sr	4.43E+03	1.28E+03	3.53E+03
^{241}Pu	5.25E+02	1.52E+02	4.19E+02
^{133}Ba	5.32E-03	2.66E-03	7.32E-03
^{234}U	5.36E+00	1.56E+00	4.28E+00

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]
^{235}U	2.27E-01	6.58E-02	1.81E-01
^{236}U	2.27E-01	6.58E-02	1.81E-01
^{238}U	1.70E+00	4.93E-01	1.36E+00
^{238}Pu	5.74E+00	1.68E+00	4.61E+00
^{239}Pu	2.56E+02	7.43E+01	2.04E+02
^{240}Pu	2.09E+02	6.05E+01	1.66E+02
^{243}Cm	9.44E-03	2.74E-03	7.53E-03
^{244}Cm	3.54E-01	1.03E-01	2.82E-01
^{113m}Cd	8.58E-01	9.40E-01	2.58E+00
^{204}Tl	8.50E-01	9.31E-01	2.56E+00
Total	**9.37E+03**	**2.95E+03**	**8.11E+03**

2.11.4 Sensitivity Analysis and Further Characterisation

112 As there are no sample data for the mortuary tubes, there is significant uncertainty in the derived inventory estimate (INV-SGHWR-011). The approach applied here has attempted to bound this by adopting very conservative assumptions when deriving an activity estimate. Therefore, developing an alternative inventory estimate that simply applies the maximum activity concentration is considered to be of limited value in sensitivity analysis.

113 One of the key observable trends in Table **2.13** is the large reduction in the assumed activation inventory over time due to the significant fraction of short-lived activation products such as ^{55}Fe and ^{60}Co. Therefore, the alternative inventory for the mortuary tubes presented here explores the effect of the adoption of a different fingerprint for the activated reactor component debris.

114 An alternative activation fingerprint was derived for activated debris from stored items remaining in the mortuary tubes based on the average modelled activation of the Zircaloy fuel channel tubes [44, Tab.7-11]. To derive the fingerprint, isotopes were screened to eliminate those which have half-lives of less than one year or that contribute less than 0.1% to the total activity. The screening exercise reduced the 128 isotopes predicted in the modelling to a set of 18 radionuclides that were then renormalised to derive the fingerprint. In line with the reference inventory, the alternative fingerprint was scaled to an overall activity of 21,400 Bq/g over the mass of the mortuary tube liners.

115 The Zircaloy fingerprint includes seven radionuclides that do not appear anywhere else in the derived SGHWR inventory: ^{93m}Nb, ^{178n}Hf, ^{85}Kr, ^{59}Ni, ^{193}Pt, ^{121m}Sn and ^{93}Zr (Table **2.15**). This can be attributed to the different composition of the Zircaloy as compared to the rebar assumed in the reference inventory. In the reference inventory the mortuary tube activity from stored item contamination reduced from 21,400 Bq/g at 01/01/2006 to 116 Bq/g at the inventory reference date. The lower proportion of short-lived activation products in the alternative fingerprint means that the alternative inventory activity reduces much less, from 21,400 Bq/g at 01/01/2006 to 6,010 Bq/g at the inventory reference date.

116 The activity of the mortuary tubes from all sources in the alternative inventory is 9,300 Bq/g. Assuming that the short-lived daughters of ^{90}Sr, ^{125}Sb, ^{93}Zr, ^{121m}Sn and ^{239}Pu (respectively ^{90}Y, ^{125m}Te, ^{93m}Nb, ^{121}Sn and ^{235m}U) are in secular equilibrium with their parents, they would contribute an additional 1,400 Bq/g for a total liner activity of 10,800 Bq/g. The majority of the activity is beta/gamma, with a small alpha component accounting for 400 Bq/g. The alternative disposal inventory activity is at the upper end of the LLW category but still below the ILW threshold of 4 GBq/tonne (4,000 Bq/g) alpha and 12 GBq/tonne (12,000 Bq/g) beta/gamma.

117 Characterisation of the mortuary tubes will be undertaken following core segmentation and post-operational clean-out in this area.

Table 2.15: Alternative activated component debris fingerprint and assumed activity derived from the activation modelling of the Zirconium channel tubes derived from [44, Tab.7-11]. Presented for a reference date of 01/01/2006.

Radionuclide	Proportion of fingerprint	Activity of mortuary tubes due to stored items [Bq/g]
^{133}Ba	7.17E-03	1.53E+02
^{14}C	9.73E-03	2.08E+02
^{60}Co	3.60E-01	7.69E+03
^{134}Cs	1.36E-01	2.91E+03
^{178n}Hf	1.73E-03	3.70E+01
^{93m}Nb	1.13E-01	2.42E+03
^{94}Nb	5.08E-03	1.08E+02
^{63}Ni	2.11E-01	4.50E+03
^{244}Cm	1.28E-03	2.73E+01
^{55}Fe	3.30E-02	7.06E+02
^{85}Kr	1.16E-03	2.47E+01
^{59}Ni	1.04E-03	2.21E+01
^{193}Pt	1.56E-03	3.33E+01
^{238}Pu	1.23E-03	2.63E+01
^{125}Sb	8.55E-02	1.83E+03
^{121m}Sn	7.55E-03	1.61E+02
^{204}Tl	2.18E-02	4.65E+02
^{93}Zr	1.95E-03	4.17E+01
Total	**1.00E+00**	**2.14E+04**

Table 2.16: Estimated alternative SGHWR mortuary tubes in-situ disposal inventory, including maximum and average activity concentrations, inventory based on average activity concentrations and alternative Zircaloy activation fingerprint for the stored item debris. All data are presented for an inventory reference date of 01/01/2027.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Alternative Inventory (MBq)
^{3}H	3.24E+02	1.60E+02	4.40E+02
^{14}C	2.23E+02	2.40E+02	6.59E+02
^{134}Cs	2.72E+00	2.72E+00	7.47E+00
^{137}Cs	2.94E+03	8.54E+02	2.35E+03
^{57}Co	3.50E-06	1.73E-06	4.76E-06
^{60}Co	5.23E+02	5.44E+02	1.50E+03
^{241}Am	1.63E+02	4.75E+01	1.31E+02
^{94}Nb	1.17E+02	1.17E+02	3.22E+02
^{125}Sb	1.01E+01	1.01E+01	2.77E+01
^{152}Eu	1.60E+00	7.94E-01	2.18E+00
^{154}Eu	6.18E+00	1.84E+00	5.06E+00
^{155}Eu	4.63E-01	1.39E-01	3.83E-01
^{55}Fe	9.08E+01	2.28E+01	6.26E+01
^{63}Ni	4.19E+03	4.32E+03	1.19E+04
^{90}Sr	4.43E+03	1.28E+03	3.53E+03
^{241}Pu	5.25E+02	1.52E+02	4.19E+02
^{133}Ba	4.13E+01	4.13E+01	1.14E+02
^{234}U	5.36E+00	1.56E+00	4.28E+00
^{235}U	2.27E-01	6.58E-02	1.81E-01
^{236}U	2.27E-01	6.58E-02	1.81E-01
^{238}U	1.70E+00	4.93E-01	1.36E+00
^{238}Pu	2.40E+01	2.56E+01	7.05E+01
^{239}Pu	2.56E+02	7.43E+01	2.04E+02
^{240}Pu	2.09E+02	6.05E+01	1.66E+02
^{243}Cm	9.44E-03	2.74E-03	7.53E-03
^{244}Cm	1.31E+01	1.32E+01	3.64E+01
^{113m}Cd	6.94E-01	1.41E-01	3.89E-01
^{204}Tl	1.06E+01	1.08E+01	2.97E+01
^{93m}Nb	1.06E+03	1.06E+03	2.90E+03
^{178n}Hf	2.49E+01	2.49E+01	6.84E+01
^{85}Kr	6.86E+00	6.86E+00	1.89E+01
^{59}Ni	2.38E+01	2.38E+01	6.55E+01
^{193}Pt	2.67E+01	2.67E+01	7.35E+01
^{121m}Sn	1.25E+02	1.25E+02	3.43E+02
^{93}Zr	4.48E+01	4.48E+01	1.23E+02
Total	**1.54E+04**	**9.30E+03**	**2.56E+04**

2.12 Primary Containment

2.12.1 Feature Description

118 The primary containment comprises a massive concrete structure with walls 1.2-1.5 m thick extending from Level 1 to Level 6 (Figure **2.15**); it housed the reactor core and numerous support operations and processes, including steam drums, clean-up plant, electrical control etc. The bioshield and mortuary tubes lie within the primary containment but due to their differing contamination characteristics they are described separately (see Sections 2.10 and 2.11).

119 The majority of the primary containment forms a single contiguous space with the reactor and bioshield roughly in the centre. The space comprises a number of areas and platforms that have unique room designations, but these are not meaningfully separated by walls or complete floors: the areas comprising the main space of the primary containment are Rooms 111, 211, 212, 311, 312, 411, 412, 413, 414, 511, 512 and 513. In addition, there are two rooms on Level 6 separated by thick walls from the main primary containment space that housed experimental circuitry – the two-element loop (Room 611) and cluster loop (Room 612) rooms.

120 The location and dimensions of the primary containment are shown in Figure **2.15** to Figure **2.18** below, and the dimensions are summarised in Table **2.17**.

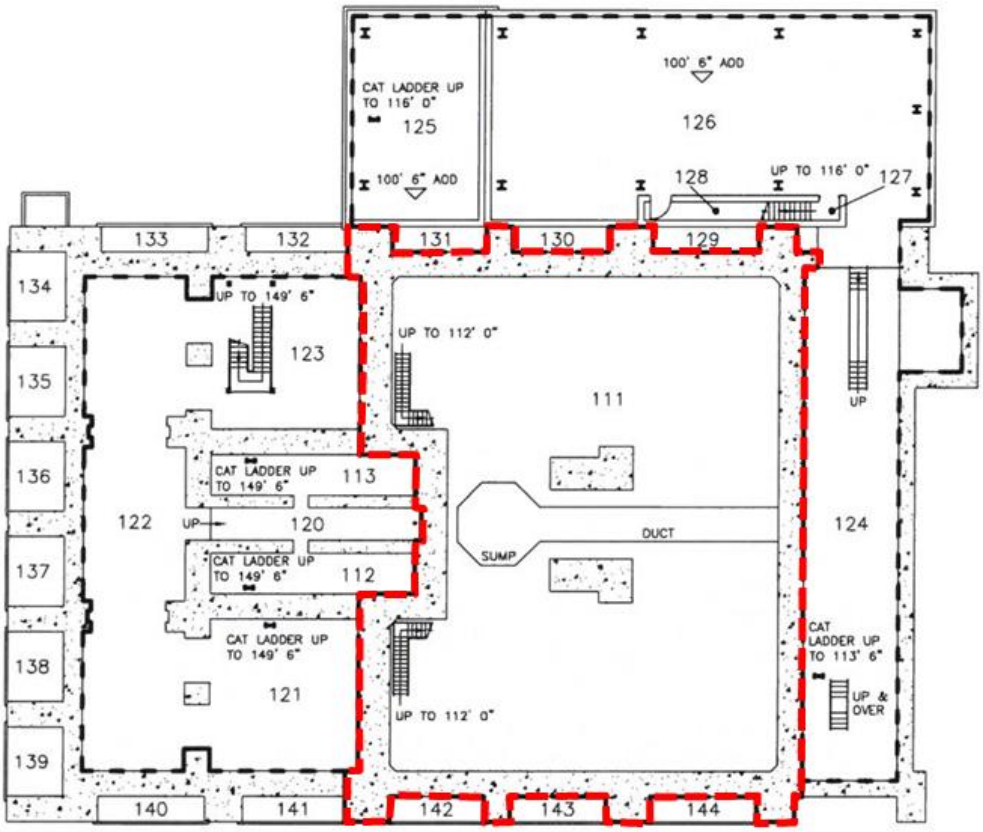


Figure 2.15: SGHWR primary containment on Level 1 (Room 111), boundary highlighted in red (from [53, Sheet 2]).

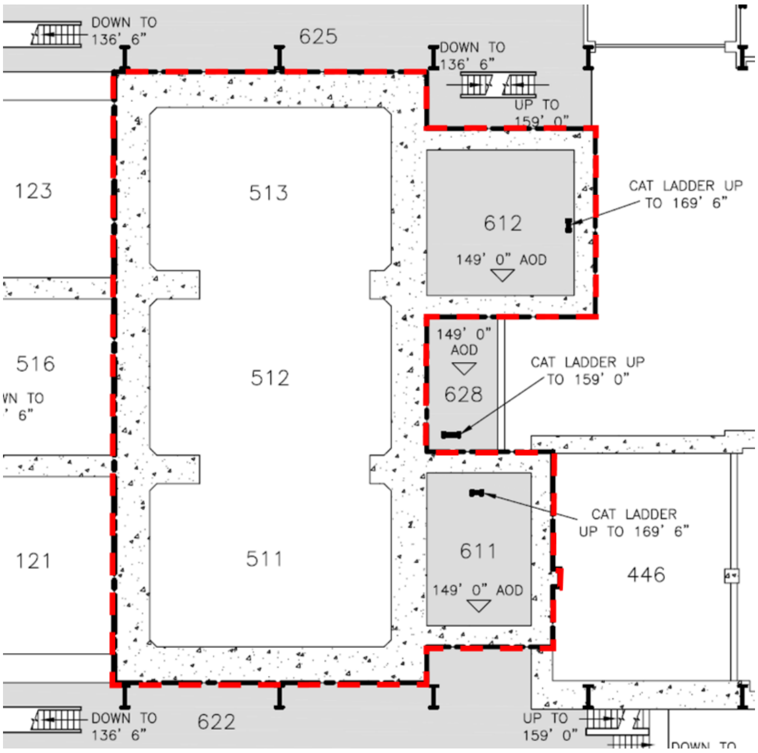


Figure 2.16: SGHWR primary containment on Level 6, boundary highlighted in red (from [53, Sheet 7]).

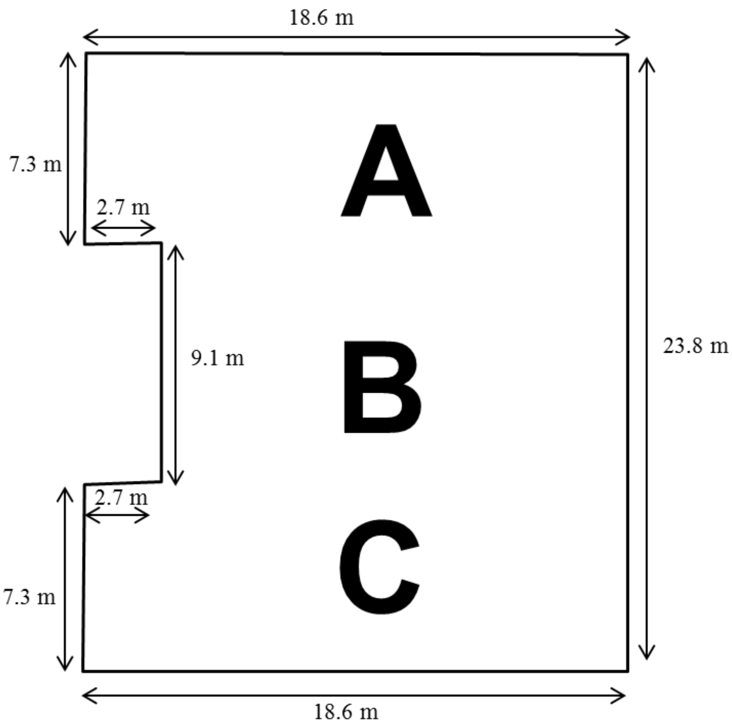


Figure 2.17: Schematic floor plan including dimensions of the SGHWR primary containment at the 94'6" mAOD (28.80 mAOD) level. The area has been split into three regions (A, B and C), which correspond to the regions referred to in Table **2.17**. From [54].

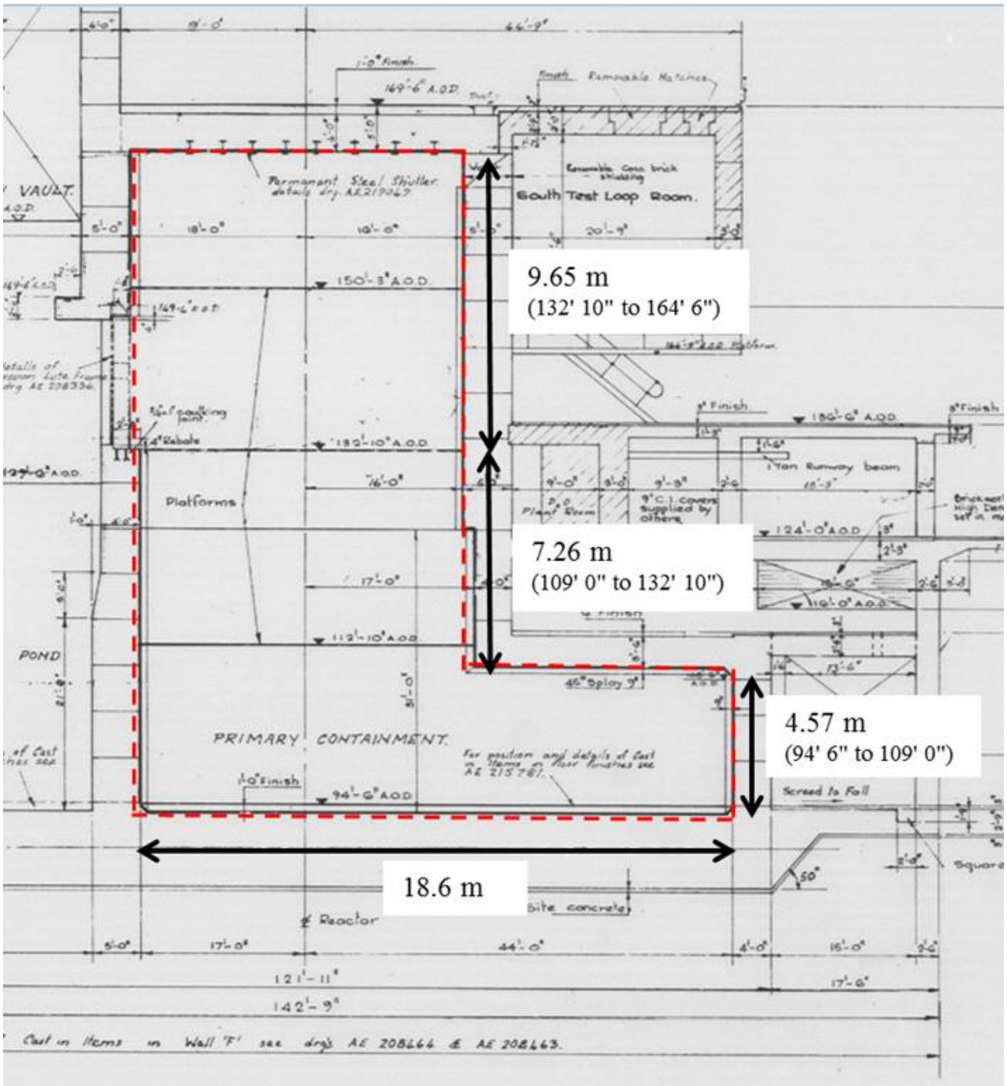


Figure 2.18: Vertical cross-section through the primary containment showing wall heights [35].

Table 2.17: Wall and floor dimensions of the SGHWR primary containment [54].

Floor		
Region	**Length [m]**	**Width [m]**
A	18.6	7.3
B	15.9	9.1
C	18.6	7.3
Walls		
Region	**Length [m]**	**Height [m]**
94’ 6” – 109’	90.1	4.57
109’ – 132’ 10”	71.8	7.26
132’ 10” – 164’ 6”	67.8	9.65

2.12.2 Origin and Constraints on Radiological Inventory

General Primary Containment Concrete Contamination

121 The main space of the primary containment was exposed to contamination due to operational leaks from liquid circuits. Wall penetrations also provide potential pathways for spread of contamination.

122 Analytical results from two primary containment characterisation campaigns in 2005 and 2019 are available. The 2005 characterisation [43, Tab.9; 54] comprises samples taken from depth intervals from nine concrete cores from the primary containment basement structure (Room 111). Two of these cores are from the wall and the remaining seven cores are from the floor. Plant experts determined the locations of the core samples using their facility knowledge to identify areas of specific interest, such as areas of known or high levels of contamination, along with general areas not expected to be contaminated, to build an overview of the radiological condition of the structure. The following radionuclides were analysed in this characterisation programme: ^{3}H, ^{241}Am, ^{134}Cs, ^{137}Cs, ^{65}Zn, ^{57}Co, ^{58}Co, ^{60}Co, ^{54}Mn, ^{40}K, ^{234}Th, ^{235}U, ^{226}Ra, ^{214}Pb, ^{212}Pb, ^{152}Eu and ^{154}Eu.

123 The maximum radionuclide activities recorded in the nine cores are for ^{137}Cs and ^{60}Co in paint layer samples from floor cores, with up to 1,200 Bq/g recorded in the surface paint layer (Table **2.18**). The next highest activities are recorded for ^{3}H (maximum 830 Bq/g) from 50-100 mm depth within the concrete floor; however, ^{3}H was also measured at levels of up to 750 Bq/g in the deepest (100-155 mm) floor core samples. The data for ^{137}Cs, ^{60}Co and ^{3}H from all nine cores are plotted in Figure **2.19**. The figure indicates a rapid fall off of activity for ^{137}Cs and ^{60}Co into the concrete core, with much higher activity in the paint layers. In contrast, for ^{3}H the activity in the concrete is typically higher than in the paint and penetrates throughout the full core depth at above LOD values.

124 Most other radionuclides reported were at the LOD or at very low levels close to the LOD and consistent with an origin from naturally-occurring radioactivity (e.g. ^{212}Pb, ^{214}Pb; see Section 2.6).

125 Previous sampling within the primary containment has been biased towards high activity circuits for waste sentencing purposes and general area samples to determine applicability of external contamination fingerprints. Magnox Ltd [55], recognising that there was limited characterisation data for the sub-structure contamination and that this would be significant for the end state risk assessment, planned a sampling campaign using a DQO methodology for the walls, floors and bioshield of the primary containment. Sixteen cores were taken in 2019 from the floor and walls of Room 111 as well as one core from the octagonal sump below the reactor (shown in Figure **2.15**). The floor sample locations were based on areas of known interest, and systematic locations were selected across each wall face to ensure that the walls were sampled with an evenly distributed coverage.

126 The 2019 samples were analysed via a number of methods [56]. Notable additions to the analytical suite from that undertaken in 2005 include ^{14}C, ^{55}Fe, ^{63}Ni, ^{90}Sr, ^{99}Tc, uranium, plutonium and curium isotopic analysis. The presence of a discrete/ widespread paint layer was not evident in the 2019 samples and as such was not analysed for. Results are summarised in Table **2.19**. Of the new additions to the

analytical coverage, maximum results of 15 Bq/g ^{14}C, 10 Bq/g ^{63}Ni, 0.3 Bq/g ^{55}Fe, 0.22 Bq/g ^{90}Sr, and < 0.07 Bq/g ^{99}Tc were reported. Very low or LOD results were reported for uranium, plutonium and curium isotopes.

127 The data show that, except for tritium, contaminants in the concrete are bounded by the 150 mm core depths used in the Room 111 sampling. Average concrete activities over a depth of 150 mm are adopted and should be representative of the total activity present for these radionuclides. A 1-mm-thick paint layer is assumed to cover all walls and the floor of the primary containment (see Section 2.9). This is a conservative assumption as the internal faces (walls, floors and ceilings) of the primary containment were shot blasted as part of the asbestos cleaning process.

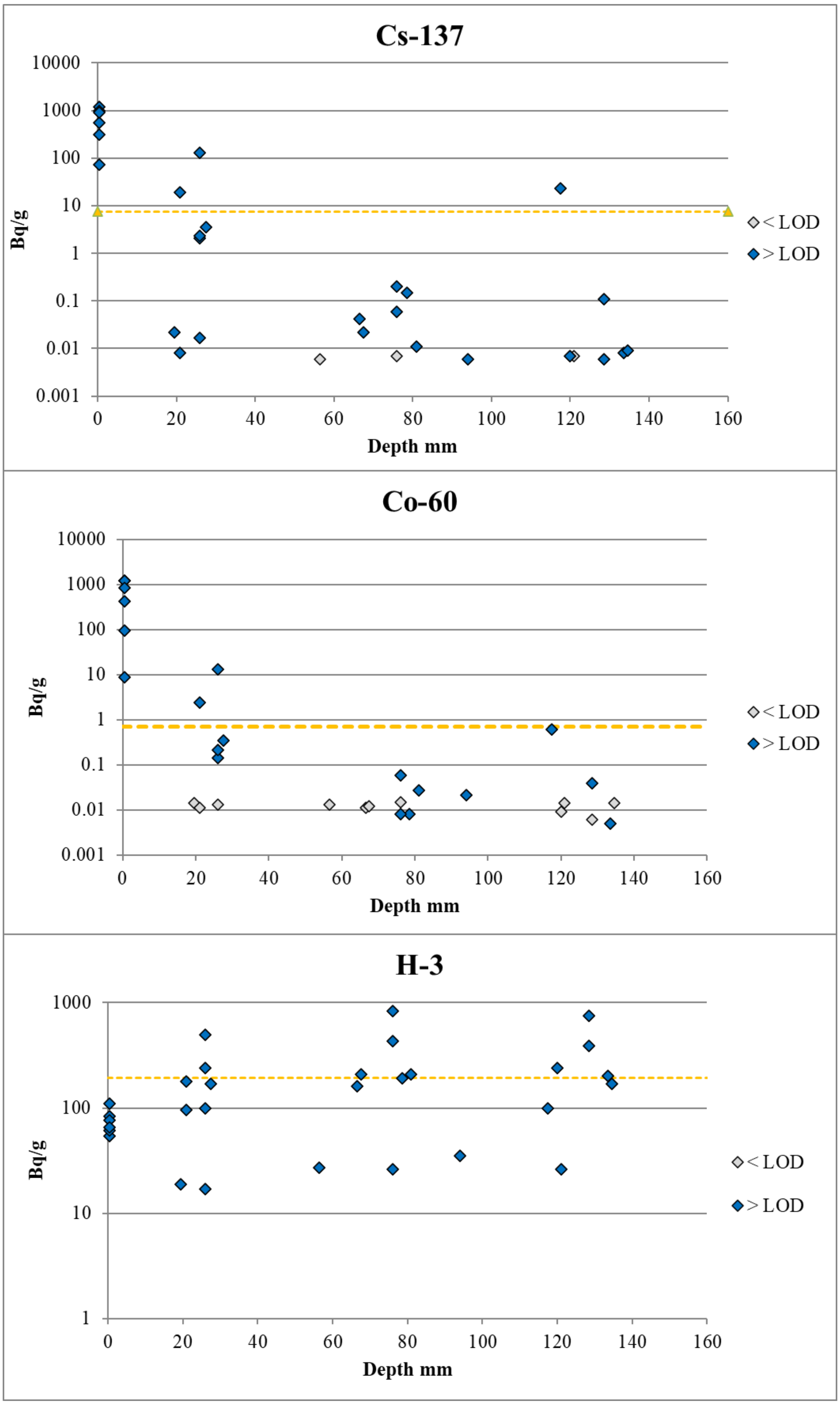


Figure 2.19: ^{137}Cs, ^{60}Co and ^{3}H activity profiles from 2005 SGHWR primary containment concrete cores (Room 111). Samples at ≈0 mm are paint. Depth indicates sample interval centre. The hatched orange line indicates average activity in the concrete samples only (i.e. excluding paint).

Table 2.18: Summary of key analytical results (Bq/g) of SGHWR primary containment (Room 111) paint and concrete core samples analysed in 2005.

Sample Type		^{3}H	^{241}Am	^{134}Cs	^{137}Cs	^{57}Co	^{60}Co	^{235}U	^{152}Eu	^{154}Eu
Paint	Average	73.8	0.54	1.40	578	<1.72	539	<1.78	<14.6	<3.62
	Maximum	110	1.10	3.00	1,200	<3.40	1,200	<3.5	<29	<6.90
Concrete	Average	199	<0.008	0.01	6.69	0.03	0.63	0.03	0.08	<0.03
	Maximum	830	<0.07	0.06	130	0.21	13	0.22	0.36	<0.17

Table 2.19: Summary of key analytical results (Bq/g) of SGHWR primary containment (Room 111) concrete core samples analysed in 2019 [55, 56].

Radionuclide	Average [Bq/g]	Maximum [Bq/g]
Gross alpha	<0.523	4
Gross beta	11.56	67
^{3}H	60.13	230
^{14}C	4.41	15
^{137}Cs	11.54	71
^{60}Co	0.884	6
^{108m}Ag	0.009	0.044
^{133}Ba	<0.035	0.310
^{152}Eu	<0.094	<0.500
^{241}Am	<0.014	<0.040
^{55}Fe	0.157	0.310
^{63}Ni	3.25	10
^{90}Sr	<0.098	0.220
^{99}Tc	<0.056	<0.070
^{233}U §	<0.008	<0.010
^{234}U α	0.005	0.008
^{235}U §	<0.001	<0.001
^{238}U α	0.005	0.007
^{238}Pu α	0.002	0.004
^{239,240}Pu α	0.003	0.012
^{239}Pu α	0.002	0.007
^{240}Pu α	0.001	0.005
^{241}Pu β	<0.080	0.170
^{242}Pu α	<0.001	<0.001
^{241}Am α	0.004	0.022
$^{243+244}$Cm α	<0.002	0.017
^{243}Cm α	<0.00004	0.00038
^{244}Cm α	<0.002	0.017

§ - denotes measurement by inductively coupled plasma mass spectrometry (ICP-MS); α – denotes measurement by alpha spectrometry; β – denotes measurement by liquid scintillation counting.

Primary Containment Bulk Concrete Tritium Contamination

128 As tritium activities remain substantial in the final segments of sampled cores from Room 111, it cannot be assumed that the 150 mm core depth will capture the full tritium activity. An additional inventory entry is therefore derived to account for the contamination of the bulk primary containment concrete.

129 Data for deeper intervals of concrete in the primary containment structure are only available from the two bioshield cores discussed in Section 2.10.2, which first pass through the primary containment structure before entering the bioshield. The first 5’ of

each of these cores is taken to correspond to the segment through the primary containment wall. The core data confirm elevated tritium activity concentrations at deeper core intervals, with an average tritium activity of 17.8 Bq/g over the first 5’ (Table **2.20**). The majority of other radionuclides are at or close to the LOD over this interval, with the exception of ^{137}Cs, which is elevated in a single sample associated with the flexcell joint noted in Section 2.10.2.

130 The walls of the primary containment vary in thickness from 4’ to 5’; for the derivation of the bulk concrete tritium inventory, it is pessimistically assumed the primary containment walls are 5’ (1,524 mm) thick in all places. The first 150 mm of concrete is excluded from this bulk concrete activity because it is covered by the estimate derived from the Room 111 data above.

Table 2.20: Summary of tritium analytical results (Bq/g) for SGHWR primary containment bulk concrete using the first 5’ segments of the 2005 LSD and ATS bioshield cores (Section 2.10.2).

Sample Type	Activity	^{3}H
Concrete - Bulk	Average	17.8
	Maximum	57.0

131 It is acknowledged that the bioshield cores were positioned in locations of expected high neutron flux rather than to capture tritium ingress, and therefore they may not be representative of tritium ingress across the whole primary containment (INV-SGHWR-006). However, the approach is believed to be reasonable given the high mobility of tritium, and the impact of uncertainty related to the adequateness of characterisation data is considered in the sensitivity analysis (Section 2.12.4).

Octagonal Sump and Duct

132 The octagonal sump, located directly under the core, collected active effluent to feed into the active drainage. A duct leads out of the sump to the west wall of the primary containment. The octagonal sump and duct are both three feet deep. Within the duct and adjacent to the octagonal sump, a further square sump is set into the duct floor (2’ square by 3’ deep).

133 One core was taken from the floor of the octagonal sump in 2019; a further core planned to be taken from the deeper square sump could not be taken due to poor access [55; 56]. The top section of the sump core had the highest bulk alpha activity of any of the cores in the primary containment (4 Bq/g); however, the activities of individual radionuclides were not exceptionally high compared to the other samples taken in Room 111. A number of radionuclides were excluded from the analytical suite for this core, notably tritium. The tritium activity from the floor core with the highest tritium concentration from Room 111 was therefore assumed. Data for ^{134}Cs, ^{57}Co, ^{154}Eu, ^{40}K and ^{226}Ra were only available for the 2005 primary containment sampling campaign, so the average and maximum activities of these radionuclides were adopted from the 2005 cores to develop an estimate for the octagonal sump and duct. The activities of key radionuclides adopted for the sump are presented in Table **2.21**. The contribution of other radionuclides was estimated using the SGHWR primary external contamination fingerprint (FP-028).

Table 2.21: Summary of key analytical results (Bq/g) for the SGHWR octagonal sump, presenting the adopted maximum and average from the sump core and from other cores in the primary containment [54; 56]. Red font indicates results at the LOD.

Sample Type		^{3}H	^{241}Am	^{134}Cs	^{137}Cs	^{57}Co	^{60}Co	^{235}U	^{152}Eu	^{154}Eu	^{226}Ra
Data from sump core (D60/PRI/CON/111/Floor Core/07)	Average	-	2.73E-02	-	2.23E+01	-	1.37E+00	1.00E-03	1.23E-01	-	-
	Maximum	-	4.00E-02	-	6.70E+01	-	4.10E+00	1.00E-03	3.00E-01	-	-
Data from primary containment cores	Average	1.90E+02	-	1.02E-04	-	5.07E-08	-	-	-	1.14E-02	1.11E-01
	Maximum	2.30E+02	-	5.15E-04	-	4.24E-07	-	-	-	5.64E-02	9.04E-01

Two Element and Cluster Loop Rooms

134 The two element loop room (Room 611) and cluster loop room (612) on Level 6 housed the two element loop and cluster loop experimental circuits. The walls and floors of both of these rooms are made of barytes concrete. No material sampling data are available for either room.

135 The cluster loop circuit was used in line with the primary circuit. For the derivation of the inventory for Room 612, it is assumed that the contamination is equal to that of the LSD plant room, which shares a contamination pathway (primary circuit) and should be pessimistic as it is one of the more contaminated process plant rooms. The derivation of the LSD plant room inventory is discussed in Section 2.13.2.

136 The two element loop was never used, and so the potential for contamination in that room is minimal [22]; consequently no inventory has been derived.

2.12.3 Inventory Estimate

137 Maximum and average radionuclide activity concentrations and a radionuclide inventory based on the average activity concentration data have been derived for the primary containment structure to be disposed in-situ based on:

- The known and assumed physical attributes of the contaminated structure, including material densities (Table **2.6**) and room surface areas ([14] and underlying references).
- The available radionuclide characterisation data from Room 111, as collated in [14].
- The tritium data for deeper intervals of primary containment concrete from the ATS and LSD bioshield cores.
- The SGHWR primary external contamination fingerprint (FP-028).
- The available radionuclide characterisation data for the LSD plant room (Section 2.13.2).
- The proportion of the primary containment below the assumed demolition datum.

138 The top of the bioshield, 132'10" mAOD (40.49 mAOD), is almost exactly 1 m below the south side ground level (41.61 mAOD)[11]. This essentially corresponds to the top of Level 3 in the general building area. It is assumed that the primary containment will be demolished to the top of the bioshield. Using the calculated volumes of the differing height sections of the primary containment main space [54], the proportion of the contaminated structure remaining in-situ (67%) can be estimated (Table **2.22**). The

[11] As observed in Figure **2.3**, there is a difference of approximately 1 m in ground level from the south to the north side of the SGHWR (41.61 mAOD to 40.53 mAOD). The higher of the two values has been used here since it is assumed unlikely that ~1 m of the bioshield would be removed when the ground could be reprofiled. It is also more conservative for the inadvertent human intrusion calculations to assume that the bioshield activity remains in a single location rather than be distributed across the backfill. Thus, there is a small uncertainty in the proportion of the inventory that may contribute to the in-situ or backfill components (INV-SGHWR-007), but the total inventory remains the same.

sump in the primary containment is all to be disposed of in-situ, and the two element and cluster loop rooms will be demolished and contribute to the backfill.

Table 2.22: Estimated volumes of primary containment components to be disposed of in-situ and as backfill based on the surface areas of each component.

Component	Assumed Depth of Contamination (mm)	Volume Fraction Disposed In-situ [%]	Building Level	Disposal	Contaminated Volume (m^3)
Octagonal sump and duct concrete	150	100	Below L1	In-situ	9.8
Primary containment paint[12]	1	67.3	L1-L3	In-situ	1.3
			L4 and above	Backfill	0.7
Primary containment surface concrete	150	67.3	L1-L3	In-situ	202.4
			L4 and above	Backfill	98.1
Primary containment bulk concrete	1374	67.3	L1-L3	In-situ	1853.9
			L4 and above	Backfill	899.0
Cluster Loop Room concrete	550	0	L6	Backfill	107.8
Cluster Loop Room paint	1	0	L6	Backfill	0.2
Total					**3173**

139 The inventory for the SGHWR primary containment structure is developed from the above data in [14] and underlying references. The overall inventory for the primary containment is derived from the sum of the following inventories derived for each component:

- The average activity concentration derived from the paint sample data in Room 111 applied to the assumed volume of paint which will remain in-situ on the walls and floor of the primary containment following demolition of the structure (1.3 m^3)[12].
- The average activity concentration derived from the concrete sample data in Room 111 applied to the top 150 mm of concrete remaining in-situ for the walls and floor of the primary containment (202.4 m^3).
- The average activity for the octagonal sump and duct applied to the top 150 mm of the concrete of the sump and duct (9.8 m^3).

[12] Note that, as per paragraph 127, the internal faces (walls, floors and ceilings) of the primary containment have been shot blasted and surface coatings removed as part of the asbestos cleaning process. However, a paint layer is conservatively assumed to be present.

- The bulk tritium contamination derived from the average activity in relevant segments of the bioshield cores applied to the estimated unaccounted-for volume of the primary containment concrete remaining in-situ (i.e. concrete deeper than 150 mm and below the demolition cutline; estimated to be 1853.9 m^3).

140 The maximum and average activity concentrations and the derived in-situ disposal inventory and alternative inventory are presented in Table **2.23**.

141 The activity associated with the primary containment above the demolition cutline, consisting of about a third of the paint, surface concrete and bulk concrete and the entirety of the cluster loop room, is included in the backfill inventory which is discussed in Section 2.17.

Table 2.23: SGHWR primary containment in-situ disposal inventory, including maximum and average activity concentrations, and a disposal inventory based on the average activity concentrations. An alternative inventory based on the maximum average activity concentrations is also presented (see discussion in Section 2.12.4). All data are presented for an inventory reference date of 01/01/2027.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative Inventory [MBq]
^{3}H	2.47E+02	1.04E+01	5.14E+04	1.99E+05
^{14}C	1.50E+01	4.32E-01	2.14E+03	7.28E+03
^{134}Cs	2.13E-03	1.23E-06	6.08E-03	2.49E-02
^{137}Cs	7.30E+02	9.21E-01	4.57E+03	4.12E+04
^{57}Co	5.87E-09	5.66E-12	2.81E-08	1.97E-07
^{60}Co	7.02E+01	3.38E-02	1.68E+02	1.25E+03
^{241}Am	1.11E+00	7.35E-04	3.65E+00	1.37E+01
^{152}Eu	9.50E+00	6.82E-03	3.38E+01	1.89E+02
^{154}Eu	1.24E+00	8.99E-04	4.46E+00	1.81E+01
^{55}Fe	1.35E+00	2.55E-03	1.26E+01	2.55E+01
^{63}Ni	4.48E+02	3.99E-01	1.98E+03	5.74E+03
^{90}Sr	1.83E-01	8.03E-03	3.98E+01	8.95E+01
^{241}Pu	3.03E+00	6.24E-03	3.10E+01	6.47E+01
^{133}Ba	1.88E-01	2.17E-03	1.08E+01	9.28E+01
^{99}Tc	7.00E-02	5.74E-03	2.85E+01	3.52E+01
^{233}U	1.00E-02	7.70E-04	3.82E+00	5.05E+00
^{234}U	8.00E-03	5.30E-04	2.63E+00	4.03E+00
^{235}U	3.50E+00	2.20E-03	1.09E+01	1.14E+02
^{238}U	7.00E-03	5.24E-04	2.60E+00	3.52E+00
^{238}Pu	3.26E-01	2.27E-04	1.13E+00	2.56E+00
^{239}Pu	6.53E-02	1.72E-04	8.53E-01	3.35E+00
^{240}Pu	6.52E-02	1.43E-04	7.08E-01	2.76E+00
^{242}Pu	1.00E-03	5.98E-05	2.96E-01	5.07E-01
^{242}Cm*	7.58E-09	4.99E-10	2.48E-06	3.81E-06
^{243}Cm	3.15E-04	3.23E-06	1.60E-02	1.53E-01
^{244}Cm	1.24E-02	1.28E-04	6.33E-01	6.05E+00
^{252}Cf*	1.37E-04	9.04E-06	4.49E-02	6.90E-02

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative Inventory [MBq]
^{226}Ra	1.49E+01	1.45E-02	7.19E+01	4.89E+02
Total	**1.55E+03**	**1.22E+01**	**6.05E+04**	**2.55E+05**

* ^{242}Cm and ^{252}Cf originally reported as a combined activity. For the purpose of this inventory estimate, it is conservatively assumed that the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.12.4 Sensitivity Analysis and Further Characterisation

142 The main source of uncertainty for the inventory derived for the primary containment is the adequateness of the characterisation data (INV-SGHWR-006). The primary containment main space surface inventory (consisting of the paint and top 150 mm of concrete) is based on a large number of cores exclusively from the lowest level of the primary containment (Room 111). As the primary containment main space extends a further 16 m above the top of Room 111, the sampling data is insufficient to consider the primary containment fully characterised. The other contributions to the inventory are also underpinned by few cores: the bulk concrete by two cores, and the sump by one core. No sampling data are available for the cluster loop transducer room, with the inventory assumed to be equal to the LSD plant room.

143 Calculating the inventory based on the maximum, rather than average, activity values for each material gives an alternative total disposal inventory as shown in Table **2.23**. The maximum activity values are conservative because they are derived from the maximum for each radionuclide from all samples, rather than the sample with the highest total activity; it is highly unlikely that a single sample would contain the maximum for every radionuclide.

144 The alternative disposal inventory is greater than the main disposal inventory estimate by a factor of four. Most radionuclides increase by a factor of between one and ten, with the largest proportional increase observed for ^{235}U (10.5-fold increase) and the lowest proportional increase observed for ^{99}Tc (1.23-fold increase). The dominant contaminant remains tritium followed by ^{137}Cs.

145 Additional characterisation that can be undertaken following the core segmentation and post-operational clean-out of the primary containment area would help to reduce uncertainty in the inventory estimate. However, given that sampling has been both biased towards known contamination areas and also considered systematically in more recent campaigns, it is expected that any new primary containment inventory estimate underpinned by such data would be bounded by the alternative estimate presented above. Both the reference and alternative inventories will be considered in the radiological PA.

2.13 Secondary Containment

2.13.1 Feature Description

146 The secondary containment comprises a concrete structure extending from Level 1 to Level 9 that housed the turbine / alternator, emergency water supplies, additional circuit supplies, plantrooms, ponds complex, effluent facilities and workshop areas. Circuits /

systems in the primary containment fed into the secondary containment, allowing the transfer of contamination to some areas.

147 Key components of the secondary containment feature include:

- Process plantrooms that supported specific systems and operations in the core and primary containment. Plantrooms were generally used to clean-up systems by removing radiological and chemical contaminants. Due to the processes involved there was significant potential for radiological and chemical contamination.
- Moderator process areas including filter room / ion exchange rooms, etc.
- Pond clean up areas / rooms where numerous operations were carried out in support of maintaining pond conditions and supporting operations.
- Secondary containment primary circuit process areas. The primary circuit was the highest activity circuit in the reactor. There was no heat exchange process prior to electricity generation in the turbines and therefore a significant portion of the total contamination would have carried into the secondary containment. The primary circuit clean-up processes to remove radiological and chemical contamination occurred in the secondary containment and there is significant potential for contamination to have been deposited.
- Ventilation system and support areas for the secondary containment.
- Secondary containment general and ancillary areas. The general and ancillary areas cover all open areas within the secondary containment, including the turbine floor, walkways, high level structures, vehicle loading areas, and access and exit points. These did not generally support process operations, but were exposed to the containment environment.
- Waste processing areas for waste operations.

148 The secondary containment also includes a set of areas which have unique basal floor elevations, base slab thicknesses or exposure to groundwater that makes them of specific interest for the end state PA. These areas are:

- The cofferdams, which were installed in the basement to ensure that there was no discharge outside of the facility and which allowed ground water monitoring during operation.
- The Effluent Vault (room 124) and delay tank rooms (rooms 125 and 126), which form basement (L1) structures to the south and west of the primary containment. The rooms were used for storage of sludge and effluent prior to discharge to the External Active Sludge Tanks (EAST) or ALES, respectively.
- The condenser cell (Room 241), the area supporting the condenser, which condensed primary circuit water after the turbine.
- The steam labyrinth and pipe corridors (Room 243), which contained the main steam and feed pipes between the reactor and the turbine.

149 It should also be noted that a number of rooms were merged and/or repurposed following the reactor shutdown. For example, the Failed Can Detection (FCD) plantroom (room 431) was combined with the waste sorting area (428), south transducer room (429), and the two element transducer room (432), and subsequently used as a thorium store.

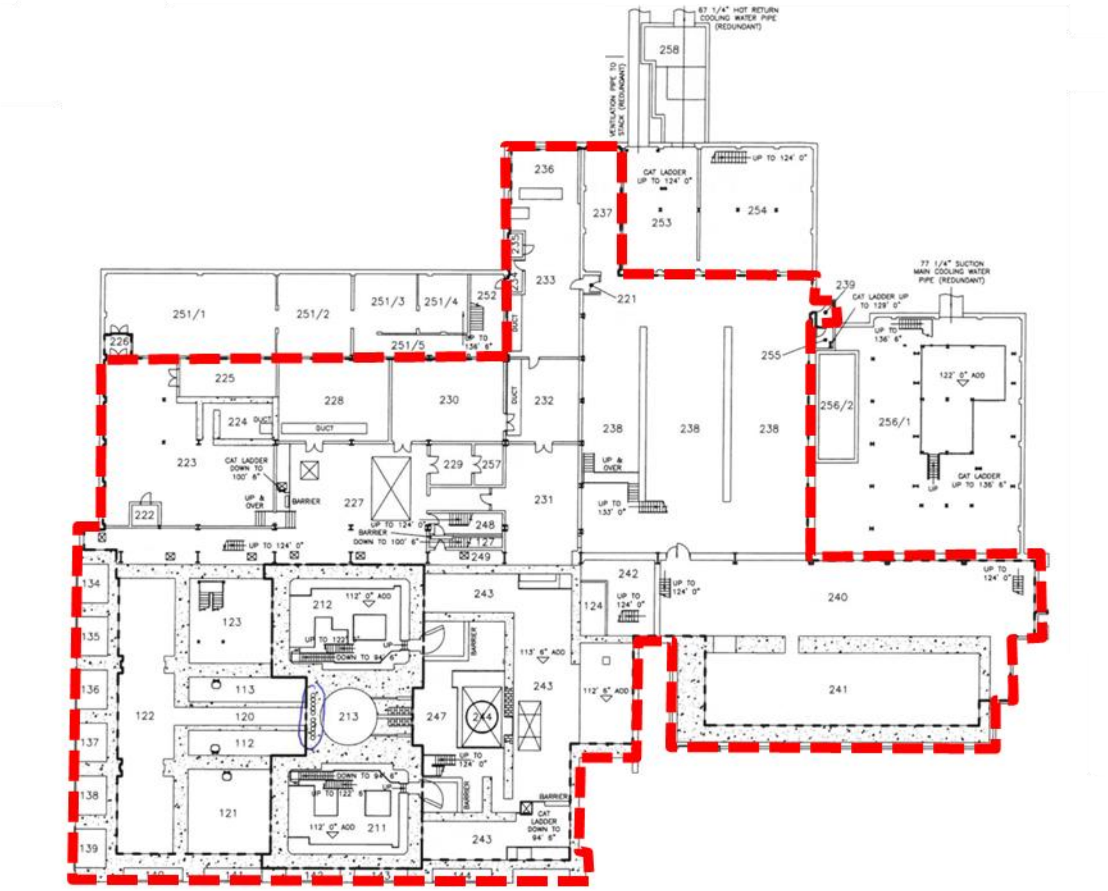


Figure 2.20: SGHWR secondary containment on Level 2 (from [57]).

2.13.2 Origin and Constraints on Radiological Inventory

150 As described above, the SGHWR secondary containment comprises dozens of rooms and components, from contaminated waste processing rooms to stairwells and inactive stores. As a consequence, the clean-up and characterisation programme covered a range of options:

- No characterisation to extensive characterisation.
- No decontamination to decontamination to OoS levels.

151 The specific origins and constraints of the radiological inventory of each room/component are captured in [14] and in underlying references.

152 It is noted that data for Levels 4 and above is patchy with many rooms uncharacterised; however, many of these are deemed to have low contamination significance based on their operating history [22] (INV-SGHWR-010).

153 The details of a number of key inventory and engineering components are described below.

Process Plant Rooms

154 The only detailed material sampling data available for the process plant rooms structure in most areas was collected prior to decontamination of the rooms [58; 59]. The highest activity measurements recorded in the secondary containment came from a single

historical core taken in 2005 from Room 247 (Level 2) [60], an access area to the primary containment. This yielded paint activities of 570 Bq/g ^{3}H, 1,500 Bq/g ^{137}Cs and 10,000 Bq/g ^{60}Co. In the underlying concrete ^{137}Cs and ^{60}Co activities fell significantly (2.1 Bq/g and 0.88 Bq/g, respectively); while ^{3}H activities rose to as high as 2,500 Bq/g in the 48-96 mm deep sample. Results from other radionuclides are much lower and typically at the LOD. The ^{137}Cs, ^{60}Co and ^{3}H profiles through this core are shown in Figure **2.21**. No attempt has been made to exclude the inventory associated with the paint because some rooms are not yet decontaminated/are still in use and decontaminated rooms may still include some paint; inclusion of the activity associated with paint pre-decontamination will conservatively over-estimate the inventory where a room has been decontaminated.

155 Different approaches to the derivation of the inventory for the process plant rooms have been adopted depending on the circumstances of the individual rooms. The different approaches are as follows:

- The ion chamber room (Room 247), LSD plant room (334) and neutron shield plant room (516) each have historical cores from 2005 [45; 59] that are adopted in the derivation of the inventory for each room. Fingerprint FP-003 (D60 Secondary Containment General Area) has been adopted for room 247 and FP-028 (SGHWR Primary External Contamination) for room 334.
- The thorium store (431), formally the FCD plant room, was combined with the deuterising plant room (427), waste sorting area (428), south transducer room (429), and two element transducer room (432) when the walls in this area were removed. An inventory for the floor and walls of the combined areas of rooms 428, 429, 431 and 432 has been derived from a pair of historical cores from the FCD plant room [59] and FP-028. The floor area associated with the Room 427 is treated separately as this room was associated with the moderator process areas and has elevated tritium contamination.
- For the north transducer loop room (332) and cluster loop transducer room (628) there is no sampling data. The activity densities for the LSD plant room are adopted as the source of contamination is the same (primary circuit) and the LSD plant room is one of the more contaminated primary circuit process plant rooms. A set of health physics monitoring surveys have been undertaken in these rooms that did not identify any hotspots.
- The two element transducer room (432), and loop make-up room (723) were designed as process plant rooms but did not house any active circuitry or operations and therefore the potential for contamination is minimal. No inventory has been derived for these rooms.

156 The two element loop room (611) and cluster loop room (612) are considered process plant rooms but are located in the primary containment; the approach for these areas is discussed in Section 2.12.

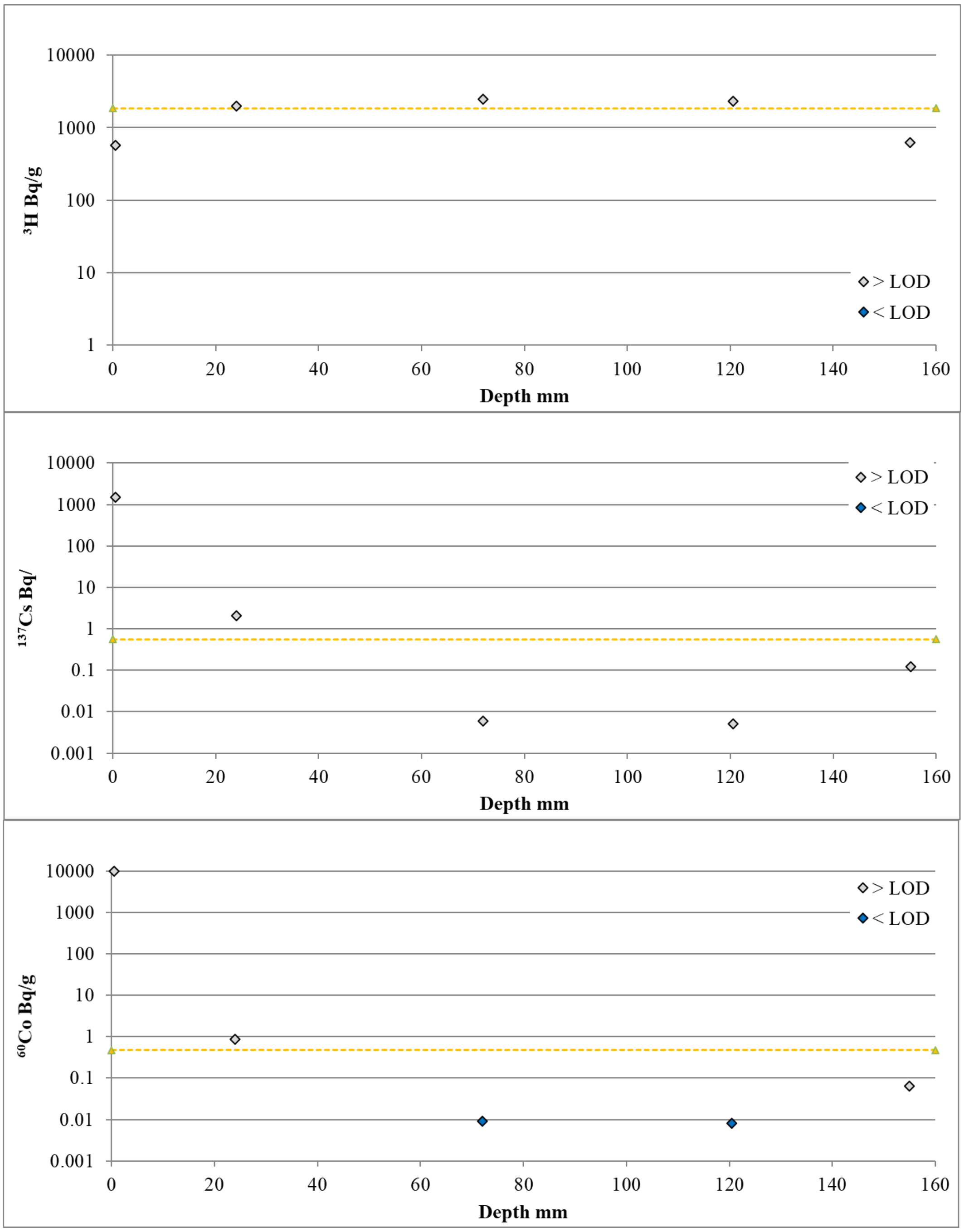


Figure 2.21: ^{3}H, ^{137}Cs and ^{60}Co activity profiles through an SGHWR secondary containment concrete core from Room 247 (sample taken in 2005). Samples at ≈0 mm are paint. Depth indicates sample interval centre. The dashed orange line indicates average activity in the concrete samples only (i.e. excluding paint).

Moderator Process Areas

157 Given the history, provenance and available sampling data, the primary contaminant of concern in these areas is tritium. During operations, the moderator circuit became activated to form high concentrations of tritium, which would then be transferred to the plantrooms in the secondary containment.

158 As tritium is intrinsically linked to water, any loss of containment or spills led to tritium contamination of the process area concrete floors and walls. Limited sampling and analysis (typically one core per room) indicates significant ingress of ^{3}H contamination into the concrete of the ion exchange room (Room 336) and the deuterising plant room (Room 427) [22; 61]. The highest activity levels were found in the deuterising plant room where the average activity to a depth of 100 mm was 1,350 Bq/g ^{3}H. Other radionuclides are typically much lower in activity concentration with values commonly approaching the LOD.

159 Different approaches to the derivation of the inventory for the moderator process rooms have been adopted depending on the circumstances of the individual rooms:

- For the D_2O ion exchange room (Room 336), the walls, ceiling and floor of the room are known to be made of barytes concrete. Sample data consist of a single core through the south wall (into Room 334) with analysis undertaken to a depth of 350 mm [45]. As the sample activity data are measured for barytes concrete, the density of barytes concrete is adopted for the derivation of the inventory for this room; fingerprint FP-030 is applied based on the SGHWR moderator circuit.
- The deuterising plant room (Room 427) no longer exists, its walls having been removed and the space merged into Room 431. However, the deuterising plant room had a substantially different contamination profile to the surrounding rooms, so a separate inventory has been derived for the floor area of the former Room 427. The floor of the room is a 3” screed over barytes concrete (which forms the roof of Room 336). Data are available from two cores of the screed layer taken in 2019 [62]. The inventory is derived from the average activity of these cores assuming the density of normal concrete; fingerprint FP-030 is applied.
- The D_2O hold-up tank room (formally Room 246) no longer exists and is now part of the ion chamber room (Room 247). There are no cores targeting the former D_2O hold-up tank room specifically, but the average measured tritium activity in Room 247 (1,855 Bq/g) is higher than the average for either of the other moderator process rooms; this suggests the Room 247 samples are capturing tritium contamination associated with the moderator circuit. The inventory for Room 246 is therefore included in the inventory estimate for Room 247.

Primary Circuit Process Areas

160 The primary circuit process areas within the secondary containment were deplanted in the Phase 1 decommissioning and a limited amount of concrete decontamination was undertaken [22]. A number of the rooms in this group have been repurposed since deplanting, and some will also be reused during the reactor core segmentation. A number of approaches were taken to derive inventory contributions for the primary circuit process areas:

- For the main pump pit (240), data consist of three concrete cores from January 2023 targeting oil spills [63] and a single historical core from 2005 [58]. The 2005 data were decayed to January 2023 and an inventory was derived from the average activities of each radionuclide from all cores. Missing radionuclides were derived from FP-003 (D60 General Area waste).

- The inventory for the combined boiler feed pump area, POWDEX filter room and feed heater valves (329/1, 329/2 and 329/3) is derived from three historical cores from 2005 [58; 59] and two cores targeting oil spills from 2023 [63]. Only the top 150 mm of core WA/SAMPLE/0949 has been included in the average as it crosses into Room 330. The waste volume calculation assumes contamination of both the floors and walls of the rooms; a pessimistic simplification has been made that includes the walls of area 445 (which is above 329/1 and 329/2 and open to 329/3) in the waste volume calculation.
- An inventory is derived for the extract pump pit (328), which is within Room 329, from a single historical core from the pit [59]. Missing radionuclides were derived from FP-003 (D60 General Area waste).
- The inventory for the Feed Heater Cell (Room 330) is derived from six floor cores, two rubble samples and 16 chipping samples from the floor, wall and plinths of the room [64]. Fingerprint FP-003 was used to derive missing radionuclides.
- The inventory for the Phillips filter room (Room 326/2) has been derived from two concrete cores taken in 2023 targeting oil spills [63]. Fingerprint FP-003 was used to derive missing radionuclides.
- There are no sample data for the Emergency Cooling Water (ECW) tank room (446); activities have been assumed to be equal to those derived for the adjacent Feed Heater Cell due to the proximity and shared source term (primary circuit).
- The characterisation of the deaerator tank room (922) consisted of six cores and nine chipping samples [65]. The inventory was derived conservatively by assuming contamination consistent with the worst-case sample. Fingerprint FP-026 (off-gas beds) was used to derive missing radionuclides.

161 The Ancillary Cooling Water (ACW) system and the cooling water switch room are considered primary circuit process areas but are located outside the secondary containment; the approach for these areas is discussed in Section 2.15.

Pond Clean Up

162 Numerous operations were carried out in the pond clean-up areas, located on Level 2 of the building, in support of maintaining pond conditions and supporting operations. Characterisation of the pond clean up rooms was completed in 2013 [66; 67]. Analysis identified a wide range of activities and sub-fingerprints. The pond clean-up areas were prone to hot spots of abnormal contamination profiles in terms of their radionuclide ratios, likely associated with spills from specific operations [22]. Analytical results have shown ^{137}Cs to be the principal radionuclide of concern in the rooms, although ^{3}H, ^{60}Co, ^{55}Fe and ^{63}Ni were also identified. The extent of contamination ingress was predominantly confined to the paint layer, with limited penetration into the concrete or masonry beneath [22]. The pond clean-up areas were decontaminated after the 2013 sampling by scabbling. In decontaminated areas, sampled paint is excluded from the derivation of average and maximum activities. The pond clean-up areas have been surveyed using health physics instrumentation to ensure no hot spots of contamination remain.

163 Fingerprint FP-018 was derived for the SGHWR pond clean up areas and used to determine the activities of radionuclides missing from the analytical suite for each area.

The activities for plutonium and curium isotopes were derived from the bulk alpha activities based on the ratio of alpha emitting isotopes in FP-018. Similar inventory derivation approaches are adopted for each of the individual rooms:

- For the Fuel Transfer Facility (Room 222) and chemist store (225) the inventories were derived from three cores for each room [66]. Following decontamination, paint is assumed to be present on all of the walls but not the floor.
- For the pond clean up room (223), the inventory was derived from 20 cores [66]. Following decontamination, paint is assumed to be present on the entirety of the east wall only.
- For the filtration and ion exchange plant room (224) the inventory was derived from three cores [66]; no paint is assumed to remain following decontamination.
- For the Rig Plant room (228) the inventory was derived from six cores [66]. Paint is assumed to be present on the entirety of the walls but not the floor.
- A separate inventory has been derived for the duct running from Room 224 to 228 from three cores taken from within the duct [66]. There is a crack in the floor of the duct (located towards the west) which was identified as having elevated contamination during the back-out survey. Further decontamination of the area (scabbling of the floor) resulted in increasing contamination suggesting that contamination has tracked and pooled within this crack. As the paint layer was typical of surface contamination found within the floor, there is reasonable supposition to extrapolate the contamination profile within the paint to that within the crack. Therefore, an estimate of the activity within the crack based on the specific activities of the nearby paint has been developed, pessimistically assuming 1 m^3 of pooled paint.

Ventilation System and Support Areas

164 The ventilation system for the secondary containment was modified several times over the years to support modified operations, in particular after Phase 1 decommissioning to maintain containment. During operations the ventilation system would have been exposed to contaminated gaseous streams from throughout the reactor operations. Identified contamination associated with these areas is very low, approximately 2.5 Bq/g ^{137}Cs, and largely confined to the paint [22, §8.8]. A number of the ventilation system areas have been decontaminated. The approach to inventory derivation for the ventilation system and support areas for the secondary containment was:

- The off-gas beds (236) and off-gas plant (237), which were used to clean out-going gases, were decontaminated to OoS. The fingerprint FP-026 (off-gas beds) was derived from material prior to decontamination. Material sampling consists of floor and wall cores from each room [68]. The alpha, Ni, Fe and Sr results obtained for Room 237 were applied to Room 236. Each of these rooms share a common source term and pathway/distribution mechanisms; therefore extrapolating this data across both rooms is appropriate.
- The inventory for the lower ventilation plant room (238) is based on a single historical core [58]; missing radionuclides have been derived from FP-003 (D60 General Area waste). The associated airlock (221) is assumed to be clean.

- An inventory was derived for the ventilation plant and associated rooms on Level 5, comprising the secondary containment cooler room (522/1, 522/2), upper ventilation plant room (523, 524), V46 plant room (525), plenum and recirculating fan (526), from cores and chipping samples [69]. Radionuclides missing from the analytical suite were derived from Fingerprint FP-026. The alpha, ^{241}Pu, Ni, Fe and Sr results obtained for Room 522/1 were applied to the entire area.
- The heat and vent plant room (842) is within the secondary containment but is adjacent to, and associated with, the Climate Control Room (CCR) plant rooms on Level 8 (which are outside the secondary containment). The inventory for room 842 has been derived from the average activity data for the CCR plant rooms. The approach and data used for the CCR plant are discussed in Section 2.15.
- There is a single historical core from the floor of the input ventilation duct (726, 727), which received fresh air from outside and circulated it through into the secondary containment [59]. The activity from this core was applied to the concrete floor in both areas as well as the floor area of the recirculation ventilation duct (728).
- The remaining secondary containment ventilation system and support areas are the Level 4 ventilation plant rooms, clean-up and filter beds (435-438). These areas have not been characterised and an inventory has not been derived.

165 Some ventilation system areas are located outside the secondary containment; the approach for these areas is discussed in Section 2.15.

Secondary Containment General Areas

166 This group comprises all open areas within the secondary containment, including the turbine floor, walkways, high level structures, vehicle loading areas, access and exit points and the steam labyrinth and pipe corridors. These did not generally support process operations, but were exposed to the containment environment. Contamination levels are generally very low across these areas. The approach for key rooms in this group was as follows:

- The inventory for the ponds access area (423) was estimated by assuming uniform activity on the floor and walls derived from a single historical core [59]. The area of the pond covers (423/1) is neglected as these are made of metal and are expected to be removed. Fingerprint FP-003 was used to derive missing radionuclides.
- The inventory for the vehicle unloading bay (433) was estimated assuming uniform activity on the floor and walls equal to the average activities derived from a single historical core [59]. A bunded area in the unloading bay contained some of the feed pipework and regulating valves for the reactor. The inventory for the floor of the bunded area was derived separately from a historical core targeting this component [59]. Fingerprint FP-003 was used to derive missing radionuclides.
- The inventory for the turbine floor (722) access area was derived by assuming contamination on the floor only, as the space is open to the secondary

containment. Activity data were derived from a single historical core [58] and Fingerprint FP-003 was used to derive missing radionuclides.

- Room 827 (roof of neutron shield plant room) covers the majority of the top of the primary containment. The inventory was derived from a single historical core [58] assuming uniform contamination of the floor. The floor area taken up by the charge machine platform has been included in the waste volume estimate. Fingerprint FP-026 (off-gas beds) was adopted, as the primary contamination pathway is expected to be exposure to the secondary containment atmosphere.
- The inventories for the roofs of the north and south suppression chambers (923 and 924) were derived from a single historical core from the roof of the north suppression chamber [59]. The inventory was derived assuming uniform contamination of the floor only as both areas are open platforms. Fingerprint FP-026 (off-gas beds) was adopted, as the primary contamination pathway is expected to be exposure to the secondary containment atmosphere.

167 A number of general containment and ancillary areas have not been characterised and/or do not have an inventory derived as they are expected to have negligible contamination.

Cofferdams

168 The SGHWR cofferdams surround the underground portion of the ponds and primary containment structure (Figure **2.22**). The cofferdams were originally built to allow access for monitoring the below-ground structures in the vicinity of the ponds and primary containment, but due to groundwater ingress an alternative monitoring regime of sampling this groundwater was implemented [70, §1.1]. The voids are filled with groundwater, the level of which depends upon the water table; the contact between the cofferdams and groundwater makes them an important component for the end state. The cofferdams have been filled with pea gravel to provide structural support.

169 Groundwater monitoring in the cofferdams commenced in the early 1990s. In 1994 the groundwater in the cofferdams was found to contain elevated levels of tritium at concentrations of 1.3, 2.1 and 4.1 Bq/g in Cofferdams 130, 132 and 139 respectively [70]. By 2004 these activity concentrations had dropped to 0.05, 0.21 and 0.76 Bq/g respectively.

170 Selected cofferdams were sampled during 2005 [71] and 2013 [70; 72]. During 2005 cores were drilled from the walls of selected cofferdams (130, 132, 135, 139 and 142), targeting cofferdams where the groundwater had been observed to have elevated tritium and gross beta activities. A layered analysis was carried out on the cores, with total core depths varying between 105 mm and 320 mm. The 2013 sampling campaign, which collected sediment and water samples, adopted a pessimistic sampling strategy; targeting the same set of dams targeted in the 2005 campaign as well as an additional dam (144) in order to ensure at least two cofferdams were sampled on each side of the building [70].

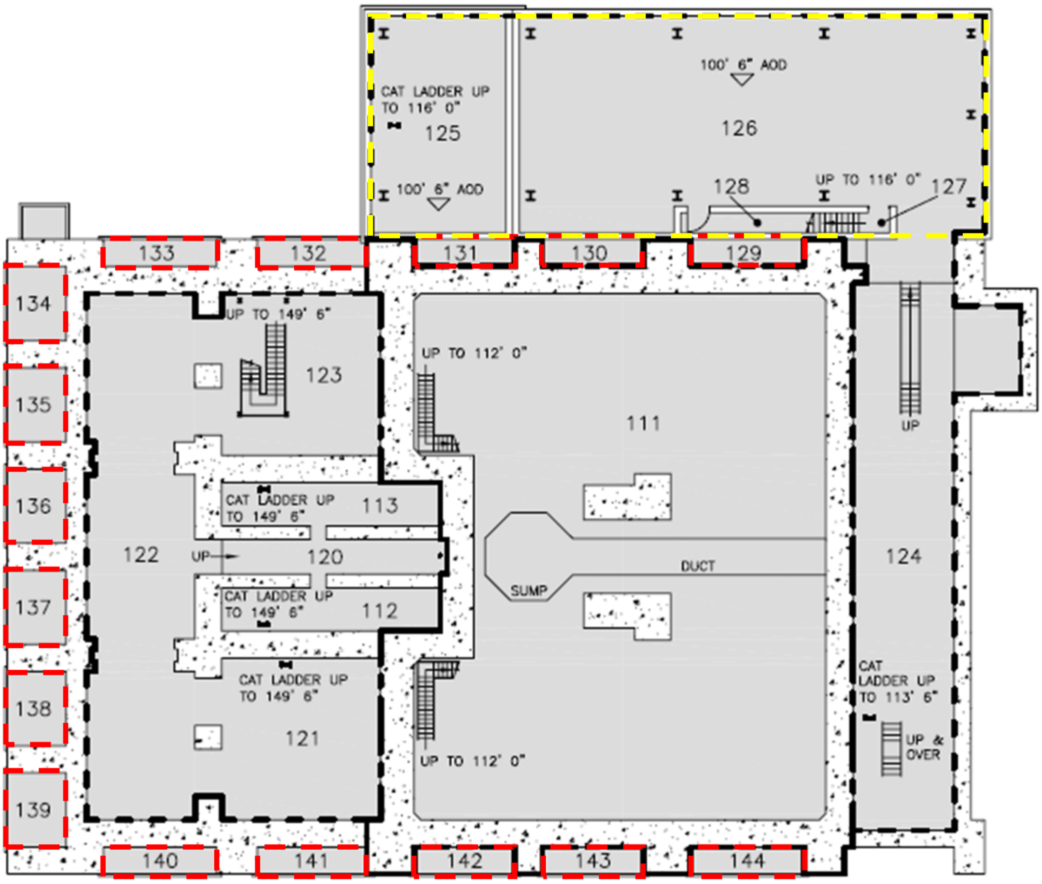


Figure 2.22: Locations of the SGHWR cofferdams (red) and delay and sludge tank rooms (yellow) on Level 1. Plans are orientated such that north side is at the bottom of the figure. Adapted from [53, Sheet 2].

171 Tritium, ^{235}U and gamma spectroscopy were commonly reported across the two sampling dates. The inventory for these radionuclides comprises three distinct data groups: data from the cofferdams grouped along the north (numbers 140 – 144), east (134 – 139) and south (129 – 133, and 145) sides. Adjacent cofferdams are interlinked by 6" pipework at the base; therefore, a degree of homogeneity of more mobile radionuclides between cofferdams is expected. Results have been extrapolated to the unsampled cofferdams.

172 During 2013 additional alpha and beta analysis was undertaken for ^{55}Fe, ^{14}C, ^{63}Ni, ^{90}Sr, ^{234}U/^{238}U, Pu and Cm, but was limited to the sediment from one cofferdam (number 132). Results were all at LOD except for ^{90}Sr (0.036 Bq/g), ^{234}U (0.00528 Bq/g) and ^{238}U (0.00476 Bq/g). The results of these radionuclides were then conservatively applied to the remaining coffer dam structures.

173 Measured contamination levels in the cofferdams are very low, with maximum activities recorded for ^{3}H of 9.2 Bq/g (2005) in a brick sample from cofferdam 6 and 3.6 Bq/g ^{137}Cs (2013) in a sediment sample from cofferdam 142.

174 With the exception of tritium, the majority of the activity is located in the sediment and in the top segment of the cores only (top 50 to 70 mm). The component activity has

been derived from the average of all samples for the grouped cofferdams, including sediment samples. No specific contamination fingerprint for the cofferdams has been derived. However, it is assumed that the SGHWR ponds fingerprint (FP-034) is appropriate as it is possible that contamination in the cofferdams originated via ingress from fractures in the pond walls[13]. The resulting activity concentrations at the inventory reference date of 01/01/27 for the north, south and east cofferdam groups are presented in Table **2.24**.

175 The top 200 mm of the concrete or brick walls of each cofferdam is assumed to be uniformly contaminated over the full height (6.28 m for all south cofferdams, 8.69 m for east and north cofferdams). Brick and concrete surface areas were derived from building plans. The pea gravel within the cofferdams, which is expected to remain in-situ, is assumed to be clean as discussions with plant staff indicate that it was added to the cofferdams in 2017, after contamination ingress.

Table 2.24: The maximum and average activities for the North, South and East Cofferdams at the inventory reference date of 01/01/2027.

Radionuclide	North Cofferdams activity (Bq/g)		South Cofferdams activity (Bq/g)		East Cofferdams activity (Bq/g)	
	Average	Maximum	Average	Maximum	Average	Maximum
^{3}H	2.58E-01	8.32E-01	5.03E-01	1.37E+00	4.81E-01	2.73E+00
^{14}C	2.26E-02	3.03E-02	5.14E-03	5.19E-03	4.15E-03	5.29E-03
^{137}Cs	5.16E-01	2.61E+00	7.88E-02	8.45E-01	5.81E-02	4.14E-01
^{57}Co	4.06E-07	5.23E-07	2.68E-07	4.18E-07	5.07E-07	8.37E-07
^{60}Co	1.79E-02	6.69E-02	2.58E-03	6.50E-03	2.92E-03	6.50E-03
^{241}Am	7.86E-03	3.54E-02	3.26E-03	8.35E-03	3.99E-03	7.30E-03
^{94}Nb	2.15E-03	2.30E-03	7.90E-04	8.00E-04	2.16E-03	3.50E-03
^{125}Sb	7.18E-04	1.14E-03	4.73E-04	6.07E-04	5.96E-04	1.02E-03
^{152}Eu	2.39E-03	2.78E-03	1.45E-03	1.87E-03	2.29E-03	3.36E-03
^{154}Eu	1.15E-03	1.20E-03	5.77E-04	9.03E-04	1.11E-03	1.76E-03
^{155}Eu	1.63E-03	2.42E-03	7.49E-04	9.80E-04	2.31E-03	3.46E-03
^{55}Fe	4.25E-03	4.25E-03	4.25E-03	4.25E-03	4.25E-03	4.25E-03
^{63}Ni	3.58E-02	3.58E-02	3.58E-02	3.58E-02	3.58E-02	3.58E-02
^{90}Sr	2.60E-02	2.60E-02	2.60E-02	2.60E-02	2.60E-02	2.60E-02
^{241}Pu	2.53E-02	1.14E-01	1.05E-02	2.69E-02	1.29E-02	2.36E-02
^{234}U	5.28E-03	5.28E-03	5.28E-03	5.28E-03	5.28E-03	5.28E-03
^{235}U	4.86E-03	1.00E-02	8.16E-03	2.00E-02	9.90E-03	2.00E-02
^{236}U	4.86E-03	1.00E-02	8.16E-03	2.00E-02	9.90E-03	2.00E-02

[13] An explicit contamination pathway has not been confirmed for the cofferdams. Alternatively to contamination from the ponds, it has also been suggested that the contamination may have migrated into the cofferdams from outside the SGHWR, possibly from the EAST/ALES effluent drawpits. However, using the ponds fingerprint is conservative.

Radionuclide	North Cofferdams activity (Bq/g)		South Cofferdams activity (Bq/g)		East Cofferdams activity (Bq/g)	
	Average	**Maximum**	**Average**	**Maximum**	**Average**	**Maximum**
^{238}U	4.76E-03	4.76E-03	4.76E-03	4.76E-03	4.76E-03	4.76E-03
^{238}Pu	1.22E-03	1.22E-03	1.22E-03	1.22E-03	1.22E-03	1.22E-03
^{239}Pu	2.97E-04	2.97E-04	2.97E-04	2.97E-04	2.97E-04	2.97E-04
^{240}Pu	2.42E-04	2.42E-04	2.42E-04	2.42E-04	2.42E-04	2.42E-04
^{242}Pu	1.10E-04	1.10E-04	1.10E-04	1.10E-04	1.10E-04	1.10E-04
^{243}Cm	1.20E-06	1.20E-06	1.20E-06	1.20E-06	1.20E-06	1.20E-06
^{244}Cm	4.65E-05	4.65E-05	4.65E-05	4.65E-05	4.65E-05	4.65E-05
^{242}Cm/ ^{252}Cf	1.92E-05	1.92E-05	1.92E-05	1.92E-05	1.92E-05	1.92E-05
Total	**9.48E-01**	**3.80E+00**	**7.06E-01**	**2.39E+00**	**6.74E-01**	**3.33E+00**

Effluent, Delay and Sludge Tanks

176 The Effluent, Delay and Sludge tanks rooms were characterised in 2016 [73; 74]. Notable activities include up to 91 Bq/g ^{137}Cs, 30 Bq/g ^{60}Co and 22 Bq/g ^{63}Ni in a shallow 0-20 mm sample from a concrete core from the Effluent Vault (Room 124). Activities measured in 20-70 mm samples were much lower. No data were obtained from the No. 3 Delay Tank Room (Room 125) so the data for the adjacent Delay and Sludge Tank room (Room 126) were used given the similar contamination pathway. The characterisation data and waste volume estimates are captured in [75].

177 An additional two cores were taken in January 2023 in Room 124 targeting oil spills [63]; these cores measured lower activities than the 2016 cores (the average decay-corrected activity for 2016 cores was 1.2 Bq/g, whereas the total activity in the 2023 cores was <0.5 Bq/g). The 2023 cores were conservatively neglected in the derivation of the inventory.

178 Process knowledge indicates that the major contributor to contamination in the delay tanks would have been the moderator system. However, in addition, any process liquid from across the system could have been disposed of to the active drains. Therefore, all potential radiological and chemical contaminants could be of concern [22]. As such the D60 General Area Waste Fingerprint (FP-003) is considered the most applicable in this area.

179 The delay tank rooms (125 and 126) have a unique basal floor elevation (see Section 2.17.2). The location of these two rooms on Level 1 is indicated in Figure **2.22**. The estimated maximum and average activity for room 126 are presented in Table **2.25**.

Table 2.25: Maximum and average activity and inventory based on average activity for the delay and sludge tank room (Room 126). The disposal inventory for the No. 3 Delay Tank Room (Room 125) is also shown, based on the average activity for Room 126. Presented for an inventory reference date of 01/01/2027.

Radionuclide	Maximum Room 126 [Bq/g]	Average Room 126 [Bq/g]
^{3}H	4.50E+00	2.53E+00
^{14}C	1.40E+00	3.85E-01
^{137}Cs	4.03E+01	1.24E+01
^{60}Co	2.14E+00	4.24E-01
^{241}Am	2.06E-02	9.84E-03
^{55}Fe	5.27E-02	2.63E-02
^{63}Ni	1.30E+01	6.53E+00
^{90}Sr	1.56E-01	9.94E-02
^{241}Pu	1.83E-01	1.40E-01
^{99}Tc	4.53E-02	1.39E-02
^{129}I	9.72E-02	2.99E-02
^{233}U	5.00E-03	4.57E-03
^{234}U	1.20E-02	9.71E-03
^{235}U	6.00E-04	4.86E-04
^{238}U	1.20E-02	8.71E-03
^{238}Pu	5.53E-03	4.74E-03
^{239}Pu	9.36E-03	4.40E-03
^{240}Pu	7.63E-03	3.59E-03
^{242}Pu	5.00E-03	5.00E-03
^{243}Cm	1.74E-04	4.33E-05
^{244}Cm	6.61E-03	1.64E-03
^{242}Cm/ ^{252}Cf	6.89E-05	5.51E-05
Total	**6.20E+01**	**2.26E+01**

Condenser Cell

180 The condenser cell (Room 241), highlighted in yellow in Figure **2.23**, housed the condenser where primary circuit water was condensed after passing through the turbine. The base slab of the condenser cell is much thicker (2.74 m) than the majority of the rooms on this level (which are generally no more than 0.53 m thick).

181 The inventory is based on a set of five cores and five wall chippings from the room [76]. The highest measured activity was ^{137}Cs with 250 Bq/g in one core sample. The remaining highest activities were measured for tritium (18 Bq/g), ^{90}Sr (5.6 Bq/g), ^{14}C (3.1 Bq/g) and ^{63}Ni (1.9 Bq/g). Remaining radionuclides were at or close to LOD in all samples. Almost all contamination was confined to the top 20 mm of the cores, with the exception of tritium which was measured at elevated activities (2-3 Bq/g) in the

deepest core segments. Fingerprint FP-046 (condenser cell) was derived specifically for this component.

182 The inventory is derived for the condenser cell assuming the average activity of samples and a contamination depth of 50 mm for the floor and walls of the room. The average and maximum activities for the condenser cell at the inventory reference date of 01/01/2027 are presented in Table **2.26**.

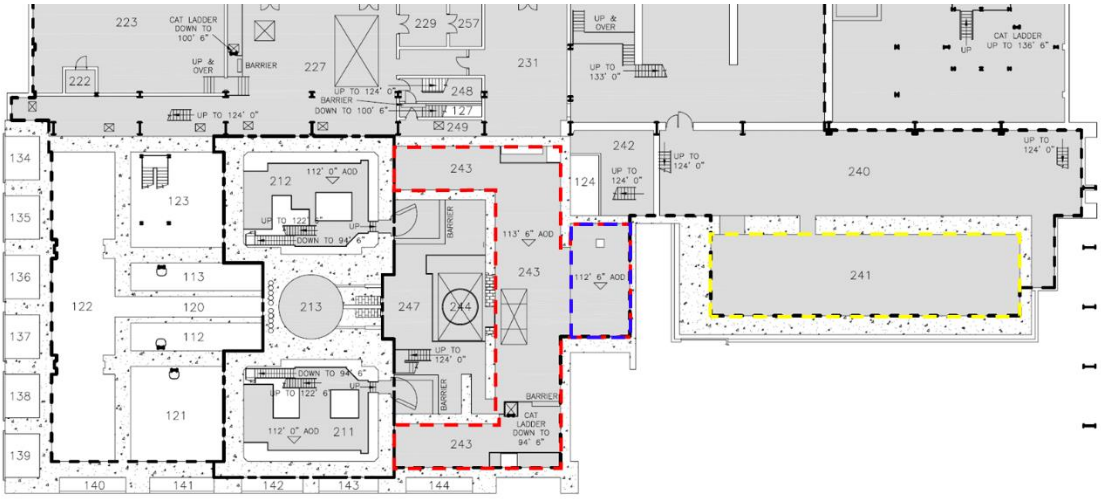


Figure 2.23: Location of the steam labyrinth and feedpipe corridors (red), raised sub-area (blue) and condenser cell (yellow) on Level 2. Adapted from [53, Sheet 3].

Table 2.26: Maximum and average activity for the condenser cell (Room 241). Presented for an inventory reference date of 01/01/2027.

Radionuclide	Average [Bq/g]	Maximum [Bq/g]
^{3}H	5.86E+00	1.06E+01
^{14}C	1.16E+00	3.10E+00
^{137}Cs	3.14E+01	2.01E+02
^{60}Co	9.02E-02	2.32E-01
^{241}Am	1.10E-02	1.10E-02
^{55}Fe	3.69E-02	5.53E-02
^{63}Ni	5.61E-01	1.78E+00
^{90}Sr	5.75E-01	4.46E+00
^{241}Pu	3.17E-01	3.17E-01
^{233}U	5.00E-03	5.00E-03
^{234}U	6.10E-03	8.00E-03
^{235}U	5.00E-03	5.00E-03
^{238}U	5.90E-03	7.00E-03
^{238}Pu	4.64E-03	4.64E-03
^{239}Pu	2.75E-03	2.75E-03
^{240}Pu	2.24E-03	2.24E-03
^{242}Pu	5.00E-03	5.00E-03
^{243}Cm	9.59E-05	1.60E-04
^{244}Cm	3.68E-03	6.14E-03
^{242}Cm/ ^{252}Cf	4.25E-04	4.25E-04
Total	**4.00E+01**	**2.22E+02**

Steam Labyrinth

183 The steam labyrinth and feedpipe corridors (Room 243) contained the main steam and feed pipes between the reactor and the turbine. Most of the floor area is at the 133’ 6” mAOD level and is located above the primary containment basement (Room 111) and the effluent vault (Room 124). A raised area on the west side of the room is partially founded on ground and partially underlain by weak concrete up to 5 m thick [21]. The location of the steam labyrinth and the raised area are presented in Figure **2.23**.

184 Characterisation consists of two concrete cores analysed in January 2023 targeting oil spills [63] and two historical cores from 2005 [45]. The data suggest that the majority of the contamination is confined to the paint; the maximum measured ^{137}Cs activity was 550 Bq/g in a paint sample in one of the 2005 cores. The maximum ^{137}Cs activity measured in the concrete was 0.37 Bq/g in one of the 2023 cores. The exception is tritium, which is observed in all segments of the 2005 concrete cores with an average activity of 11.6 Bq/g (tritium was not analysed for in the 2023 cores). To derive an inventory for Room 243, the 2005 core and paint data was decayed to January 2023 and the disposal inventory derived from the average activity of both sets of cores. Missing radionuclides were estimated using Fingerprint FP-003 (D60 General Area waste).

185 The waste volumes for Room 243 are estimated from the total area of the floor and walls, assuming a concrete contamination depth of 150 mm and a paint thickness of 1 mm. The maximum and average activities for paint and concrete are presented in Table **2.27**.

Table 2.27: Maximum and average activities for paint and concrete in the steam labyrinth and pipe corridors (Room 243). Presented for a reference date of 01/01/2027.

Radionuclide	Room 243 Paint – Average [Bq/g]	Room 243 Paint – Max [Bq/g]	Room 243 Concrete – Average [Bq/g]	Room 243 Concrete – Max [Bq/g]
^{3}H	3.16E+00	4.45E+00	3.44E+00	6.24E+00
^{14}C	7.38E+00	1.47E+01	2.69E-03	1.49E-02
^{137}Cs	1.68E+02	3.35E+02	6.13E-02	3.39E-01
^{57}Co	2.79E-09	4.81E-09	1.63E-06	7.71E-06
^{60}Co	7.05E+00	1.41E+01	2.69E-03	1.28E-02
^{241}Am	5.19E-01	8.70E-01	1.00E-01	1.08E-01
^{94}Nb	-	-	5.50E-04	6.80E-04
^{125}Sb	-	-	2.45E-03	2.75E-03
^{152}Eu	2.05E+00	3.31E+00	1.56E-02	3.31E-02
^{154}Eu	4.40E-01	7.36E-01	3.47E-03	6.31E-03
^{55}Fe	2.18E-01	4.34E-01	8.29E-05	3.96E-04
^{63}Ni	3.53E+01	7.04E+01	1.35E-02	6.43E-02
^{90}Sr	9.48E+00	1.89E+01	3.46E-03	1.91E-02
^{241}Pu	1.70E+00	2.86E+00	3.29E-01	3.54E-01
^{99}Tc	1.89E-01	3.77E-01	6.90E-05	3.82E-04
^{129}I	4.05E-01	8.07E-01	1.48E-04	8.18E-04
^{234}U	1.97E+01	3.38E+01	1.23E-01	2.99E-01
^{235}U	1.51E+00	2.60E+00	9.47E-03	2.30E-02
^{238}U	1.62E+01	2.77E+01	1.01E-01	2.45E-01
^{238}Pu	1.39E-01	2.33E-01	2.68E-02	2.88E-02
^{239}Pu	1.59E-01	2.67E-01	3.08E-02	3.31E-02
^{240}Pu	1.30E-01	2.18E-01	2.51E-02	2.70E-02
^{243}Cm	1.79E-04	3.00E-04	3.45E-05	3.72E-05
^{244}Cm	8.81E-03	1.48E-02	1.70E-03	1.83E-03
^{226}Ra	6.34E+00	1.09E+01	4.25E-02	8.52E-02
^{40}K	1.19E+01	1.70E+01	1.60E-01	3.00E-01
Total	**2.92E+02**	**5.60E+02**	**4.50E+00**	**8.24E+00**

Other

186 Other areas within the secondary containment for which an inventory was derived are:

- Waste sorting area (227) and black drum store (230), for which there is a single historical core each [58]. FP-003 (D60 General Area waste) was applied to derive missing radionuclides.
- The inventory for "area below dirty oil tank" (Room 242) is derived from two concrete cores analysed in January 2023 targeting oil spills [63]. Fingerprint FP-003 used to derive missing radionuclides. No paint was assumed as it is absent from the samples.
- Inventories were derived for the pipe corridors (321) and transverse cable tunnel (322) assuming uniform contamination on the floor and walls derived from historical sampling data for each room [58; 59]. Two cores were also taken from the pipe corridors in January 2023 targeting oils spills [63]; the maximum activity for the 2023 cores was 0.7 Bq/g, which is lower than the average activity of 2.8 Bq/g for the decay-correlated historical cores. The 2023 data were conservatively neglected for the derivation of the inventory. Missing radionuclides were derived using fingerprint FP-003.
- Sample data from eight cores, containing both fibreglass and concrete samples [69], were obtained for the Maintenance and Decontamination Pit (520). The fingerprint FP-038 (Maintenance and Decontamination Pit) was derived specifically for this room.
- The delay tank pump room (231) is directly above the delay tank room and is characterised by only a single historical core [58]. Activities in this core are substantially lower than the tank rooms themselves, although a similar contamination profile is evident. Fingerprint FP-003 was applied to this room, consistent with the approach taken for the delay tank room.
- Areas on Level 8, including the instrument active workshops (836 and 837), health physics store (838) and corridor (840), chart rooms (841 and 844), store room (845) and hoist well (846), have been decontaminated. Characterisation data consists of one core from Room 836 and chipping samples from 836, 837, 840, 842, 844 and 835 [77]. The remaining contamination is very low. The inventory for the grouped rooms was derived from the average activity of the sample data and using Fingerprint FP-026 (off-gas beds).

2.13.3 Inventory Estimate

187 Maximum and average radionuclide activity concentrations and a radionuclide inventory based on the average activity concentration have been derived for the secondary containment structure based on:

- The known and assumed physical attributes of the contaminated structure, including material densities (Table **2.6**) and room surface areas ([14] and underlying references).
- The available radionuclide characterisation data from the structure, as collated in [14].
- The relevant fingerprints presented in Section 2.5.
- Assumed depths of penetration of contamination into the building fabric.
- The proportion of the secondary containment below 40.6 mAOD (the assumed demolition datum), which is assumed to comprise Levels 1-3 of the existing structure.

188 The inventory for the SGHWR secondary containment structure is developed from the above data in [14] and underlying references. Maximum and average activity concentrations and an estimate of the radioactive inventory based on average activity concentrations are presented in Table **2.28**.

Table 2.28: SGHWR secondary containment in-situ disposal inventory, including maximum and average activity concentrations, and a disposal inventory based on the average activity concentrations. An alternative inventory based on the maximum activity concentration for each component within the secondary containment is also presented (see discussion in Section 2.12.4). Presented for a reference date of 01/01/2027.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative inventory (MBq)
^{3}H	7.42E+02	1.33E+01	5.60E+04	8.00E+04
^{14}C	4.01E+01	1.03E-01	4.35E+02	1.55E+03
^{134}Cs	1.95E-02	2.52E-05	1.06E-01	2.81E-01
^{137}Cs	9.14E+02	1.46E+00	6.14E+03	3.91E+04
^{57}Co	2.24E-05	1.47E-06	6.17E-03	1.42E-02
^{60}Co	5.86E+02	8.16E-02	3.44E+02	1.06E+03
^{241}Am	1.01E+01	2.97E-02	1.25E+02	2.38E+02
^{94}Nb	1.70E-01	5.98E-04	2.52E+00	3.67E+00
^{125}Sb	2.07E-02	4.78E-04	2.01E+00	3.21E+00
^{152}Eu	2.78E+01	1.40E-02	5.88E+01	9.43E+01
^{154}Eu	3.51E+00	2.86E-03	1.20E+01	1.84E+01
^{155}Eu	7.55E-02	7.28E-04	3.06E+00	5.02E+00
^{55}Fe	1.81E+01	7.33E-03	3.09E+01	4.74E+01
^{63}Ni	2.93E+03	1.04E+00	4.40E+03	8.38E+03
^{90}Sr	5.15E+01	3.52E-02	1.48E+02	4.61E+02
^{241}Pu	9.88E+00	8.98E-02	3.78E+02	5.00E+02
^{99}Tc	1.03E+00	7.16E-04	3.01E+00	1.18E+01
^{129}I	2.20E+00	1.53E-03	6.46E+00	2.52E+01
^{233}U	6.00E-03	4.00E-04	1.68E+00	1.84E+00
^{234}U	8.84E+01	8.50E-02	3.58E+02	6.35E+02
^{235}U	6.80E+00	8.90E-03	3.75E+01	6.62E+01
^{236}U	2.00E-02	1.48E-03	6.22E+00	1.34E+01
^{238}U	7.25E+01	7.01E-02	2.95E+02	5.23E+02
^{238}Pu	3.15E+00	8.33E-03	3.51E+01	6.54E+01
^{239}Pu	3.56E+00	9.21E-03	3.87E+01	7.33E+01
^{240}Pu	2.89E+00	7.50E-03	3.16E+01	5.97E+01
^{242}Pu	5.00E-03	4.42E-04	1.86E+00	1.86E+00
^{242}Cm*	2.19E-09	9.73E-11	4.09E-07	4.20E-07
^{243}Cm	4.19E-03	1.38E-05	5.79E-02	1.21E-01
^{244}Cm	2.14E-01	6.40E-04	2.69E+00	5.59E+00
^{252}Cf*	4.25E-04	1.43E-05	6.03E-02	6.60E-02
^{226}Ra	2.77E+01	3.85E-02	1.62E+02	3.14E+02
^{40}K	7.00E+01	1.51E-01	6.35E+02	1.24E+03
Sum	**5.62E+03**	**1.65E+01**	**6.97E+04**	**1.35E+05**

* ^{242}Cm and ^{252}Cf originally reported as a combined activity. For the purpose of this inventory estimate, it is conservatively assumed that the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.13.4 Sensitivity Analysis and Further Characterisation

189 The main source of uncertainty for the inventory derived for the secondary containment is the adequateness of the characterisation data (INV-SGHWR-006). Many rooms are only characterised by a single core and almost all rooms do not have a statistically representative set of samples (instead relying on plant and process knowledge to target areas of known or suspected contamination, an approach that can bias datasets). The derived inventory assumes that the samples can be treated as representative, scaling the average activities of the samples to derive the activity of each room. These are likely to be conservative assessments.

190 The alternative inventory for this area instead adopts the maximum activity measured for each radionuclide in each inventory contribution for each component (generally paint and concrete for each room) and assumes this will be representative of the overall activity of the contribution. The alternative inventory is presented in Table **2.28**. The inventory based on the maximum activities is 133,000 MBq, approximately twice that of the inventory based on the average activities. The activities of most radionuclides in the alternative inventory are approximately double that in the reference inventory; the largest increase is that of ^{137}Cs which increases by a factor of 6.4. The large increase in ^{137}Cs activity is driven by the sampling of ^{137}Cs hotspots in concrete in the condenser cell and pond clean up areas.

191 A substantial dataset is available for the secondary containment as a whole, although there is considerable variation in the amount of characterisation data for each component. Limited data for a component typically occurs where there is an expectation of low contamination due to process history, with targeted characterisation for expected contamination areas. As such, additional characterisation data will help to refine the inventory, but is not expected to be inconsistent with the overall estimate presented here.

2.14 Ponds

2.14.1 Feature Description

192 There are three distinct types of ponds within the SGHWR reactor:

- Fuel element ponds for the storage of spent fuel prior to off-site transport.
- Dump ponds used to condense steam that had been ‘dumped’ from the primary circuit in the event that pressure release valves on the steam drums were tripped.
- Suppression ponds, designed to condense any steam produced during a loss of coolant accident and in the event of reactor trips on the primary circuit.

193 The location of these ponds, which are adjacent to the primary containment, is illustrated in Figure **2.24**. Although areas within the ponds facility are all inter-connected, the source term varied depending on the pond in question (i.e. primary circuit for the dump and suppression ponds, and spent fuel for the fuel element pond). Therefore it can be expected that there will be variations within the type and amount of contamination between different ponds.

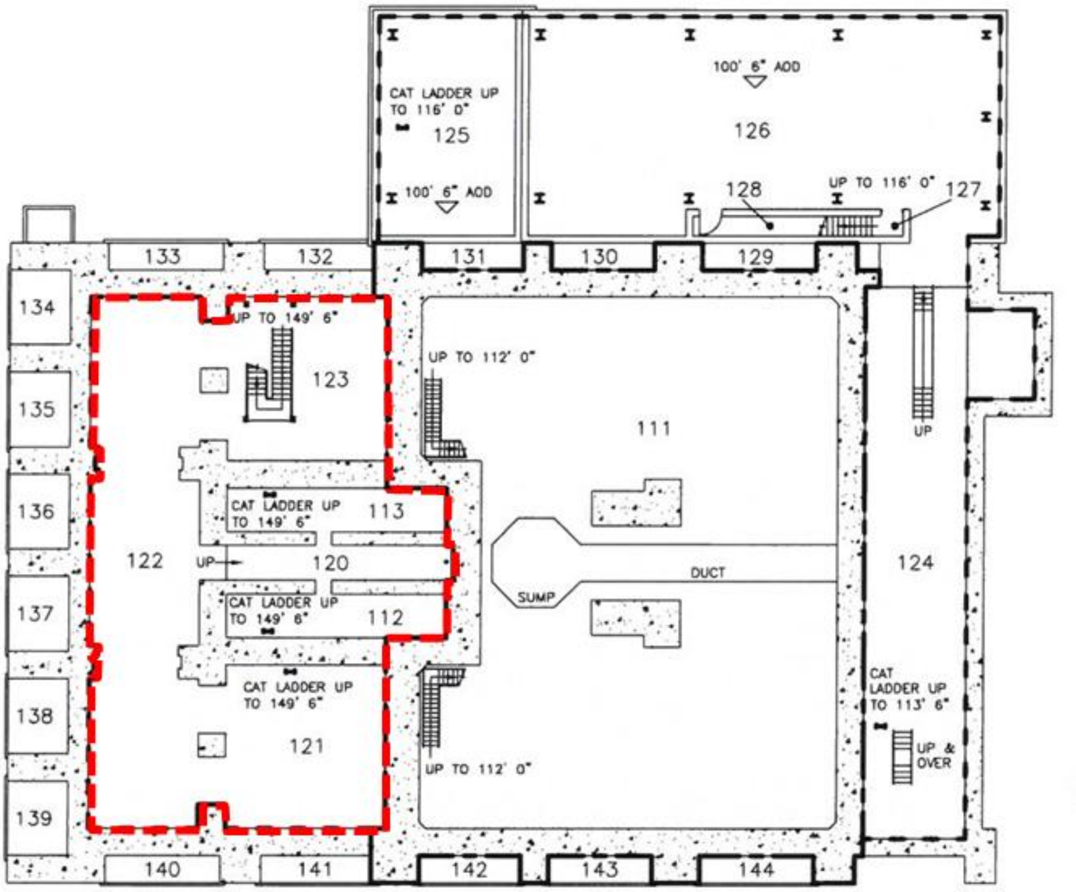


Figure 2.24: SGHWR Ponds on Level 1, comprising North Centre Dump Pond (112); South Centre Dump Pond (113); Fuel Element Transfer Tunnel (120); North Suppression Pond (121); Fuel Element Pond (122); and South Suppression Pond (123). From [53, Sheet 2].

194 The ponds were in use throughout the SGHWR operating life and are known to be relatively highly contaminated. The ponds were emptied after transfer of fuel ceased and drained between 2003 and 2005 [78]. A limited cleaning operation was completed using water jetting and decontamination agents prior to fixing remaining contamination using a waterproof paint. The ponds' concrete and fibreglass liners will remain in-situ at the IEP. An inspection in 2014 found some areas of the paint and fibreglass layer to be peeling. A number of joints and cracks have been identified in the pond walls and floor. Dimensions of the contaminated pond structure are presented in Table **2.29** and Table **2.30**.

Table 2.29: Dimensions of pond floors, where length is defined as the long axis [83].

Pond	Length [m]	Width [m]
Fuel Element Pond (122)	22.9	4.9
North Suppression Pond (121)	7.3	7.2
South Suppression Pond (123)	7.3	7.2
North Centre Dump Pond (112)	9.5	2
South Centre Dump Pond (113)	9.5	2
Fuel Transfer Tunnel (120)	9.5	1.5

Table 2.30: Dimensions of pond walls [83].

Pond	Perimeter [m]	Height [m]*
Fuel Element Pond (122)	55.6	10.7
North Suppression Pond (121)	29	10.7
South Suppression Pond (123)	29	10.7
North Centre Dump Pond (112)	23	10.7
South Centre Dump Pond (113)	23	10.7
Fuel Transfer Tunnel (120)	22	10.7

* 10.7 m represents the maintained pond liquor level.

2.14.2 Origin and Constraints on Radiological Inventory

195 A significant characterisation programme was completed for the SGHWR ponds in 2016 [79; 80; 81] comprising 17 cores from pond floor areas and 126 wall cores with associated health physics monitoring. The aims of the programme were to determine the distribution of contamination between the fibreglass liner and underlying bulk material, the depth and type of contamination ingress into the bulk material, and to assess the variability in contamination levels at varying elevations above the pond floor.

196 Many of the highest activities from the ponds characterisation programme were found in a painted fibreglass sample in a single core (core 5) from the floor of the Fuel Element Pond. Above LOD results from this sample are presented in Table **2.31** and selected results are presented in Figure **2.25** along with the underlying concrete samples from core 5. The results show that the bulk of the contamination is held within the 3-mm-thick fibreglass liner, orders of magnitude higher than that in the concrete beneath the liner. The dominant radionuclides in the fibreglass sample are ^{137}Cs and ^{90}Sr, with approximately equal activities. The tritium has a different distribution along the core length compared to other radionuclides, with relatively constant levels measured through the core depth and with a slightly lower fibreglass activity. This was observed across the ponds with pond floor tritium concrete averages ranging from 1.93 Bq/g in the 50-100 mm interval from the Dump Ponds to 2.65 Bq/g in the 0-50 mm interval from the Fuel Transfer Tunnel. This contrasts to fibreglass tritium activity levels of 0.48-0.55 Bq/g. Variation in activities across the Fuel Element Pond suggests contamination is distributed heterogeneously [79].

197 Other pond floors were found to have much lower total activities than the Fuel Element Pond. The floor beta/gamma activity fingerprint was found to vary from pond to pond with ^{90}Sr, ^{137}Cs and ^{241}Pu dominating in the Fuel Element Pond, ^{137}Cs, ^{60}Co and ^{63}Ni within the Fuel Transfer Tunnel and Dump Ponds, and ^{137}Cs, ^{63}Ni and ^{90}Sr within the Suppression Ponds [79].

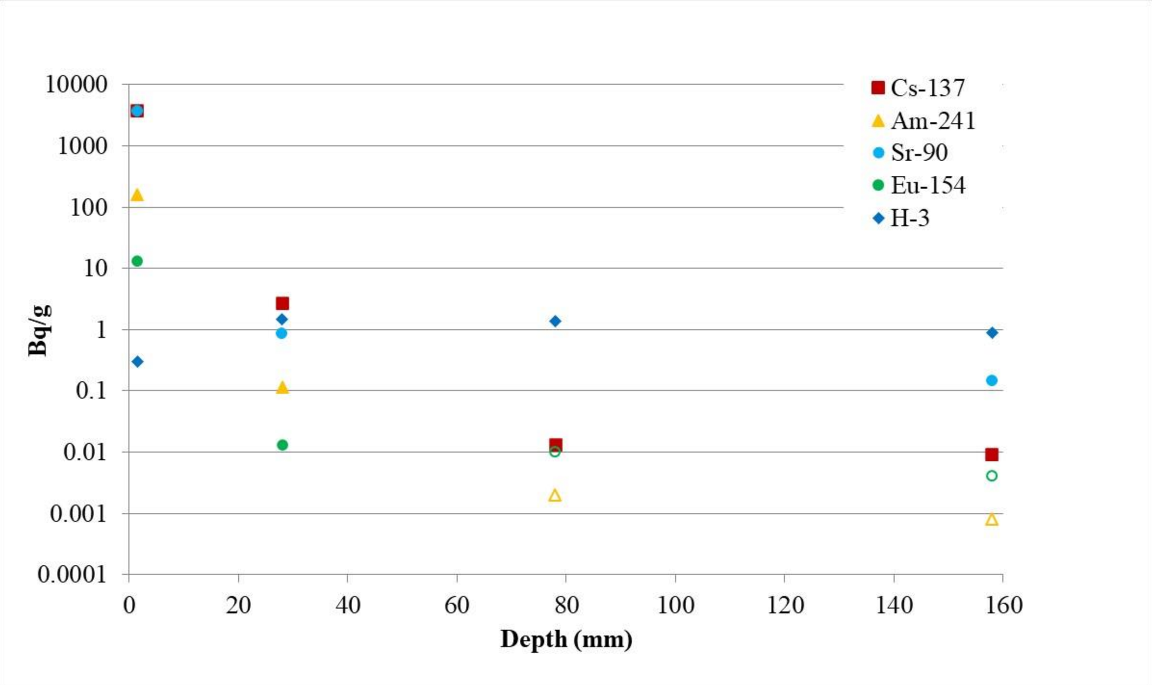


Figure 2.25: ^{137}Cs, ^{241}Am, ^{90}Sr, ^{154}Eu and ^{3}H activity profiles through SGHWR Fuel Element Pond core 5 [81]. Samples at ≈0 mm are paint/fibreglass, depths beyond that are concrete. Depth indicates sample interval centre. LOD values have hollow symbols.

Table 2.31: Above limit of detection results (Bq/g) from fibreglass sample D60/SEC/CON/122 taken from the floor of the SGHWR Fuel Element Pond [83].

	Gross Alpha	Gross Beta	^{3}H	^{137}Cs	^{60}Co	^{241}Am	^{154}Eu	^{155}Eu	^{90}Sr	^{241}Pu	^{239}Pu/^{240}Pu	^{239}Pu	^{240}Pu
D60/SEC/CON/122	340	11,000	0.3	3,700	1.9	160	13	1.6	3,650	896	162	89.19	72.81

198 Alpha contamination (principally ^{241}Am and $^{239}Pu/^{240}Pu$) was found to be largely restricted to the fibreglass layer with the highest specific activities in the Fuel Element Pond Floor (126.95 Bq/g). Beneath the fibreglass much lower activities (~0.04 – 0.18 Bq/g) were recorded.

199 Construction joints and cracks within the floor were intended to be targeted for sampling. Visible cracks were evident in a screed layer beneath the fibreglass but on removal of the screed were found not to propagate into the underlying base slab [79]. Health physics monitoring also showed background doses at the intersection of the screed cracks with the base slab. There was difficulty in locating floor construction joints even following removal of the overlying screed in the areas to be sampled. Where the screed was removed, health physics monitoring again showed background doses indicating contamination had been held up in the fibreglass layer and screed.

200 On the pond walls contamination is again located primarily in the fibreglass liner with the highest beta/gamma activities found in the Fuel Element Pond (maximum 790 Bq/g) and lowest levels in the north centre dump pond (0.6 Bq/g). Beta/gamma activity was found to be dominated by ^{137}Cs. Beneath the fibreglass beta/gamma activity is considerably reduced, with a range between LOD and 7.4 Bq/g in the upper 50 mm and generally LOD levels in deeper samples. Alpha activity follows a similar pattern with maximum specific activity of 340 Bq/g in the fibreglass (Fuel Element Pond) falling to or near LOD (generally ~0.2 Bq/g) in the underlying concrete. Tritium was found to be relatively evenly distributed throughout the walls with lowest levels in the fibreglass (0.3-0.8 Bq/g), rising slightly in the 0-50 mm section (0.9-1.9 Bq/g) and generally falling slightly in the 50-200 mm section.

201 It is generally observed across the ponds that ^{90}Sr is either dominant or at a similar activity to ^{137}Cs in the very active fibreglass samples (hotspots) but that considerably less ^{90}Sr than ^{137}Cs is present in the lower activity fibreglass samples and in the concrete. The discrepancy is likely due to the higher relative mobility of ^{137}Cs which results in greater diffusion of this radionuclide away from its source. Although it is likely that ^{137}Cs and ^{90}Sr would have been released in relatively similar quantities, it is found that ^{137}Cs is the dominant radionuclide in the inventory. The apparent overall discrepancy between the ^{137}Cs and ^{90}Sr activities in the inventory may be partially a consequence of the cleaning operations undertaken during the decommissioning of the ponds which could credibly have removed relatively more ^{90}Sr than ^{137}Cs due to ^{90}Sr being more concentrated in hotspots and generally less dispersed than ^{137}Cs.

202 Wall construction joints were cored at intersections between vertical and horizontal joints, close to the pond floor where hydrostatic pressure would have been greatest. However, analysis showed only low levels of contamination comparable to the cores targeting bulk concrete from the pond walls.

203 Several wall cracks are visible on the external walls of the Fuel Element and Suppression Ponds with white efflorescence and elevated dose rate (several 100 cps) indicating a degree of contaminant migration through the walls. Cores targeting these cracks found that the bulk of the contamination is attributable to ^{137}Cs with activities between 14.5-64 Bq/g on Level 2 and 3.2 Bq/g on the external wall of the Fuel Element Pond on Level 3. All other beta/gamma and alpha emitting radionuclides were below OoS. Tritium was found to be uniformly distributed (2.3 Bq/g to 3 Bq/g). It is noted

that the extent of cracking is difficult to assess due to the fibreglass liner on the pond internal walls and a lack of access to many of the ponds external walls.

204 Fingerprint FP-034 was derived specifically for the SGHWR Fuel Ponds [82]; this fingerprint is presented in Table **2.4** and was used to derive missing radionuclides from the sample data.

2.14.3 Inventory Estimate

205 The assumed depths of contamination for components in the ponds area are captured in Table **2.32**; their derivation is discussed below.

Table 2.32: Assumed depth of contamination penetration into SGHWR ponds materials [83].

Material	Depth [mm]
Concrete (walls and floor)	200
Concrete (construction joints, effective width)	21.6
Concrete (cracks, effective width)	43.2
Paint	Captured in fibreglass data
Fibreglass	3

206 The volumes of contaminated pond wall and floor materials were calculated using engineering plans to determine the surface area and assuming the fibreglass layer to be 3 mm thick, with the remaining activity held within the top 200 mm of underlying concrete [79; 83].

207 Evidence from the sampling suggests a lack of contamination in floor construction joints, while in contrast elevated contamination levels are observed in the wall construction joints. The lack of contamination in the floor construction joints might be attributed to the additional overlying layer of screed separating them from the pond water that was not present on the walls.

208 An estimate of the total contamination inventory associated with the wall construction joints was derived by assuming that the elevated contamination associated with the construction joints is localised within a thin plane in the joint itself running the full width of the associated wall. The length and width of the construction joints planes were derived from engineering plans. An example of the distribution of construction joints and cracks in the Fuel Element Pond (122) east wall is presented in Figure **2.26**.

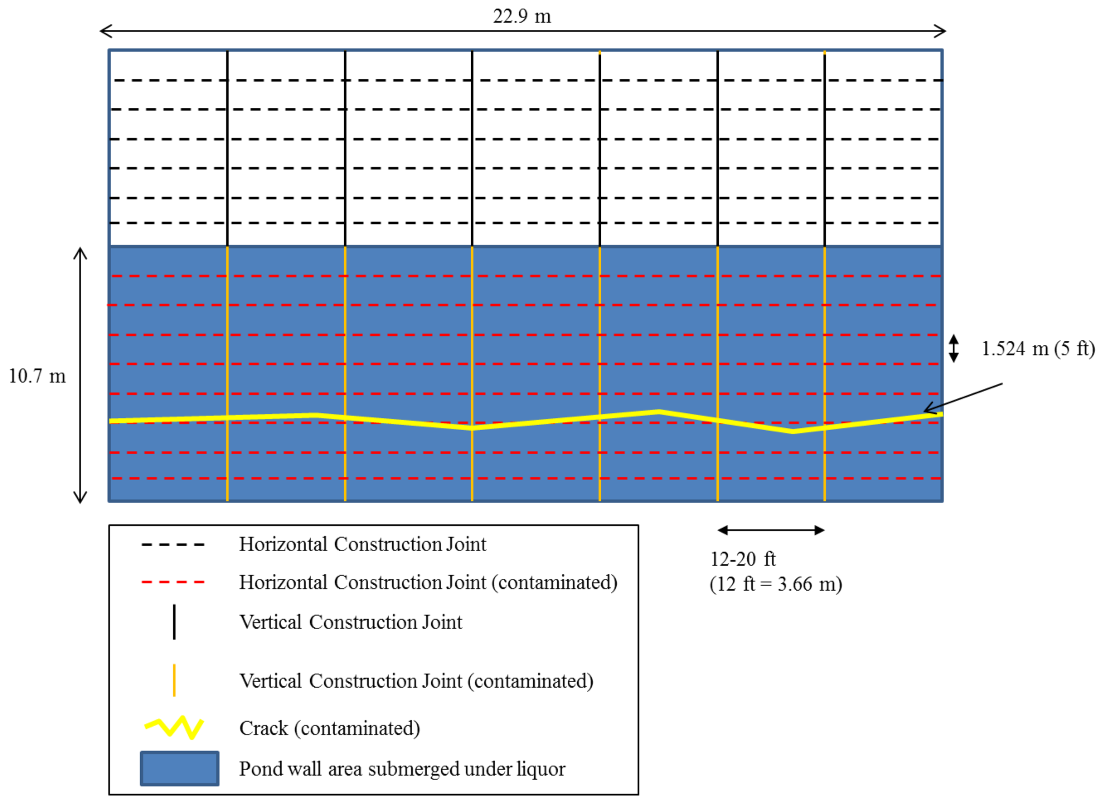


Figure 2.26: Schematic of Fuel Element Pond (122) east wall construction joints and cracks [83].

209 The measured activity concentrations are an average value over the volume of the cylindrical core taken. To derive the activity associated with a construction joint plane, which is assumed to be concentrated in a thin layer, it is therefore necessary to adopt an "effective width" that can be used to derive a waste volume from the known surface area of construction joints, which maintains the same ratio of (assumed) contaminated to uncontaminated material as in the cores. The effective width, d_{contam}, for a plane of contamination is equal to the volume of the core, V_{core}, divided by the surface area of the plane sampled by the core, A_{contam}. For a core of radius r_{core} and length l_{core}, which is positioned such that the axis is on the plane of contamination:

$$d_{contam} = \frac{V_{core}}{A_{contam}} = \frac{\pi r_{core}^2 \, l_{core}}{2 r_{core} \, l_{core}} = \frac{\pi r_{core}}{2}$$

210 A diagram of a core relative to the assumed plane of contamination and the relevant dimensions is shown in Figure **2.27**.

211 For construction joints, the cores are taken at the intersection of two planes and so the area of plane sampled is doubled and $d_{contam} = \frac{\pi r_{core}}{4}$. Given the core radius of 27.5 mm, an effective width of 21.6 mm is derived for the construction joints. This value is multiplied by the area of the construction joint planes to derive a waste volume to which the average activity of the cores can be applied. Activities from the Fuel Element Pond construction joint samples were extrapolated to other joints in this pond. Likewise, results from the Suppression and Dump Pond construction joints were extrapolated to other joints in these areas.

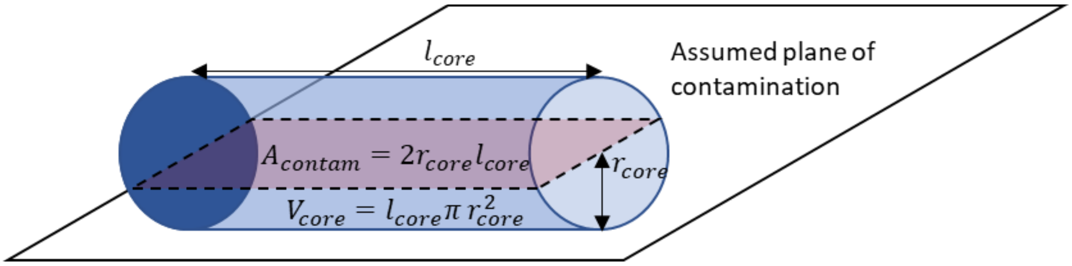


Figure 2.27: Diagram of the assumed plane of contamination associated with a construction joint or crack and the associated area sampled by a core intersecting the plane. Note construction joint cores are taken at the intersection of two planes and so the area of sampled contamination is doubled.

212 Volume calculations for the observed cracks were undertaken as for the construction joints. The planar area was estimated using known wall thicknesses and either visible measurements of the crack length, or, where partially obscured, by assuming that the cracks run the full length of the wall. The number of cracks was based upon observations of the visible exterior walls [79; 83]. An effective contamination width was calculated to maintain the sampled plane:core volume ratio as for construction joints. The cores targeting cracks were taken on the plane of a single crack rather than at the intersection of two planes as for the construction joints, so a contamination width of $d_{contam} = \frac{\pi r_{core}}{2}$ = 43.2 mm was derived for cracks.

213 The inventory for the SGHWR ponds structure is developed from the above data in [14] and underlying references. Maximum and average activity concentrations and an estimate of the radioactive inventory based on average activity concentrations are presented in Table **2.33**.

Table 2.33: SGHWR ponds in-situ disposal inventory, including maximum and average activity concentrations, a disposal inventory based on average activity concentrations, and an alternative inventory based on more pessimistic dimensional assumptions (as discussed in Section 2.14.4). All data are presented for a reference date of 01/01/2027.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative Inventory [MBq]
^{3}H	2.40E+00	8.20E-01	9.55E+02	1.02E+03
^{14}C	7.99E-01	9.61E-02	1.12E+02	1.24E+02
^{137}Cs	2.92E+03	2.93E+00	3.41E+03	7.02E+03
^{60}Co	1.75E+01	2.13E-02	2.48E+01	5.18E+01
^{241}Am	1.69E+02	1.28E-01	1.49E+02	3.07E+02
^{94}Nb	5.40E-01	2.11E-03	2.46E+00	3.70E+00
^{154}Eu	5.65E+00	1.05E-02	1.22E+01	2.08E+01
^{155}Eu	3.56E-01	1.17E-03	1.36E+00	2.06E+00
^{55}Fe	2.41E-01	2.45E-02	2.85E+01	3.14E+01
^{63}Ni	2.23E+01	1.82E-01	2.12E+02	3.39E+02
^{90}Sr	3.16E+03	3.50E+00	4.08E+03	8.48E+03

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative Inventory [MBq]
^{241}Pu	5.44E+02	6.74E-01	7.85E+02	1.31E+03
^{129}I	2.00E-01	1.03E-01	1.20E+02	1.29E+02
^{36}Cl	1.00E-01	9.29E-02	1.08E+02	1.17E+02
^{233}U	5.00E-01	4.62E-01	5.38E+02	5.80E+02
^{234}U	3.50E+00	9.46E-03	1.10E+01	1.57E+01
$^{235}U^{\#}$	3.90E-01	9.57E-03	1.11E+01	1.24E+01
$^{236}U^{\#}$	3.90E-01	9.57E-03	1.11E+01	1.24E+01
^{238}U	1.00E+00	6.67E-03	7.77E+00	9.58E+00
^{238}Pu	3.50E+00	7.84E-03	9.13E+00	1.41E+01
^{239}Pu	8.92E+01	1.24E-01	1.45E+02	2.96E+02
^{240}Pu	7.27E+01	1.01E-01	1.18E+02	2.41E+02
^{242}Pu	3.00E-01	8.29E-04	9.66E-01	1.38E+00
^{242}Cm*	2.11E-10	7.20E-12	8.39E-09	8.85E-09
^{243}Cm	5.04E-03	8.78E-05	1.02E-01	1.18E-01
^{244}Cm	1.91E-01	3.33E-03	3.88E+00	4.48E+00
^{252}Cf*	1.34E-04	4.57E-06	5.32E-03	5.61E-03
Sum	**7.01E+03**	**9.32E+00**	**1.09E+04**	**2.01E+04**

^{235}U and ^{236}U originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

* ^{242}Cm and ^{252}Cf originally reported as a combined activity. For the purpose of this inventory estimate, it is conservatively assumed that the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.14.4 Sensitivity Analysis and Further Characterisation

214 A sampling and analysis plan and a series of DQOs were developed and followed for the ponds sampling campaign [79]. The collected samples are therefore assumed to be representative of the contamination present.

215 The primary source of uncertainty in the inventory is expected to be the volume of material that is assumed to be contaminated. The alternative inventory for the ponds considers the effect of more pessimistic dimensional assumptions about the contaminated material volume. Table **2.34** records the reference and alternative dimensional parameter values and the rationale for the alternative value.

216 The overall activity for the alternative inventory is recorded in Table **2.33**. The total activity in the alternative inventory is almost double that of the reference inventory. The difference in the total activity is mostly due to the doubling of the assumed ponds liner thicknesses as the fuel pond liner comprises the majority of the ponds activity.

Table 2.34: Original reference and sensitivity values for dimensional data adopted for the ponds contamination, and the rationale for the adopted sensitivity value.

Parameter	Rationale and approach	Originally adopted value	Alternative value
Volume of contamination from cracks	It is possible that not all cracks in the ponds have been identified. The adopted volume values are doubled.	1.85 m^3	3.70 m^3
Volume of contamination from construction joints	The location and extent of construction joints is well known from building drawings. The adopted volume values are maintained.	71.6 m^3	71.6 m^3
Dimensions of ponds	The dimensions of the ponds are well known from the construction drawings. The adopted dimensional values are maintained.	-	-
Height of contamination	Contamination assumed to be to the water level in the ponds. Alternative inventory considers it possible for contamination to reach the level of the ponds access area (level 4).	10.7 m	11.5 m
Depth of contamination of the bulk concrete	Core data suggests contamination (with exception of tritium) is bounded by the core depths. The adopted depth values are maintained.	0.2 m	0.2 m
Thickness of the fibreglass pond liners	There is uncertainty in the thickness of the fibreglass liner (estimates vary from 2 mm to 3 mm). The adopted thickness is doubled.	3 mm	6 mm

2.15 Ancillary Areas

2.15.1 Feature Description

217 A considerable number of rooms in the SGHWR exist outside the secondary containment structure (Figure **2.28**). The vast majority of these rooms did not support active process operations. The rooms are grouped into larger components based on use or proximity as follows:

- Active Workshops (251 - 252), which received plant and equipment from the facility and other areas on site for maintenance. This resulted in contamination of the floor and walls. The workshops were in use throughout the operational period of the SGHWR.
- The ACW system, which continuously topped up the primary circuit to maintain water levels and pressure. Areas associated with the ACW system include the ACW pumphouse (256/1), Chiltern tank (256/2), the switch room (483/1), and an additional pumphouse (484).

- The Boiler House Basement (253), Fuel Oil Tank Room (254) and Cooling Water Washout Pit (258) are a set of adjoining rooms to the south of the primary containment that housed non-active plant and operations.
- Ventilation system and support areas.
- North Annexe miscellaneous areas, comprising primarily offices, stores and electrical facilities.
- South Annexe miscellaneous areas, which includes a number of large rooms on Level 4 as well as various switch rooms, toilets, labs, stores and offices.

2.15.2 Origin and Constraints on Radiological Inventory

Active Workshops

218 The highest levels of contamination measured in the ancillary areas are from the Active Workshops. Characterisation of this area was completed in 2012 [84] and identified a fingerprint (FP-016*) consisting primarily of ^{137}Cs, ^{3}H, ^{14}C and ^{60}Co. Contamination was removed to a level that would be consistent with the rooms being OoS of EPR 2016 and IAEA requirements (by sum of fractions) in 2021 [22]. This demonstration was in accordance with the requirements for the end state as defined at the time (2012). A hot spot of tritium contamination identified was partially removed and partially left in-situ to decay to a level that would be OoS by 2021 [22].

219 Following decontamination, the highest measured activities include 140 Bq/g tritium, 2.7 Bq/g ^{137}Cs and 0.8 Bq/g ^{14}C from Room 251/4, and 0.9 Bq/g ^{60}Co from Room 251/2. The tritium and ^{137}Cs maximum results were from concrete samples while the ^{14}C and ^{60}Co maximum results were from paint samples.

220 Inventory contributions were derived from characterisation data from each room, with radionuclides missing from the analytical suite determined from the specifically-derived fingerprint FP-016* (active workshops). Separate contributions were derived for the active workshop areas 251/1, 251/2, the active workshops stairwell (252) and the airlock (226). Areas 251/3, 251/4 and 251/5 were grouped together and an inventory was derived for the walls and floor of the space as well as the low dividing brick walls delimiting each area.

ACW System

221 Although connected to the primary circuit, the ACW system had significantly lower levels of contamination than most areas associated with the primary circuit as it acted as a clean feedwater system. The areas associated with the ACW system are the ACW pumphouse basement and Chiltern tank (256/1, 256/2), ACW switch room (483/1) and the ACW pumphouse (484).

222 Characterisation data for the ACW system consists of a single historical core from the pumphouse basement taken during the 2005 sampling campaign [58]. The radionuclide with the highest measured activity was ^{40}K with 0.35 Bq/g, the highest measured tritium activity was 0.11 Bq/g. Fingerprint FP-003 (D60 General Area waste) was used to derive the missing radionuclides. The inventory for the ACW system was derived by assuming all rooms have surface contamination equivalent to that captured by the 2005 core. The low confidence in the representativeness of the single core is offset by the low contamination significance of this area.

Boiler House Basement, Fuel Oil Tank Room and Cooling Water Washout Pit

223 The inventories for the Boiler House Basement (253), Fuel Oil Tank Room (254), Cooling Water Washout Pit (258) and the adjacent airlock and cable duct (323 and 324) were derived separately using core and chipping samples from each room [68]. The activities are uniformly low with the highest measured activity being 2.4 Bq/g of tritium measured in the airlock. Missing radionuclides were derived from FP-026 (SGHWR off-gas beds). For Ni, Fe and Sr, the results obtained for the Fuel Oil Tank Room (Room 253) were applied to the other rooms in this group.

Ventilation System and Support Areas

224 The primary part of the ventilation system located outside the secondary containment is the CCR vent plant and equipment, and plant rooms on Level 8 (852, 853, 854, 855, 859). Measurement data consist of one floor core from each of Room 859 and 852, a total of nine chipping samples from various rooms, one wood, metal and paint sample and one smear sample [85]. Uniform very low activities were found within paint, with very little differentiation between the paint and underlying brick or concrete. Therefore, activity calculations on paint as a separate layer were not required (average activities were formed from combining the results from the paint and concrete samples). The rooms were grouped together and the average activities were applied to derive the inventory for all rooms. Missing radionuclides were derived using fingerprint FP-026.

225 The heater room/main airlet (559/560) on Level 5 lies outside the secondary containment but is accessed through and associated with the other ventilation plant on Level 5 (which lie inside the secondary containment). Characterisation data consist of one core and a number of paint and chipping samples; fingerprint FP-026 (SGHWR off-gas beds) was applied to the averaged activities to derive missing radionuclides [69].

226 There is a single further vent plant room on Level 6 (663) for which no inventory is derived; no sample data are available, although the area has been remediated and is considered to have background contamination levels.

North Annexe Miscellaneous Areas

227 Rooms outside the secondary containment in the North Annexe did not support any active processes, being largely offices, stores and electrical facilities. Airlocks into the secondary containment are present on Levels 3, 4 and 6. Activities were uniformly low with the highest activity measured in Room 357, where a gross alpha activity of 1.1 Bq/g and a gross beta activity of 1.5 Bq/g was measured in a wall chipping sample [86].

228 The construction in parts of the North Annexe is known to be mixed brick and concrete, but the exact ratio of each material is not known. The inventory estimates for these rooms pessimistically adopt the density of concrete for scaling activity densities that are derived from brick samples. Where an inventory was estimated, fingerprint FP-026 (SGHWR off-gas beds) was adopted to derive the activities of radionuclides missing from the analytical suite as the primary contamination pathway is exposure to the containment atmosphere.

229 Characterised areas in the North Annexe consist of:

- The cable basements on Level 3 (Rooms 352, 353, 357 and 358): the characterisation data for each room consist of one floor core and at least two chipping samples from 2014 [86].
- The airlock (360), turnstile area (361) and monitoring area (362) on Level 3: the characterisation data consist of one floor core and at least two chipping samples for each room from 2014 [86].
- Two of the cable mezzanines on Level 5 (Rooms 551-552): there is a single core each from 2005 [87; 88].
- The 90 ton tank room (951) and lift motor room (952): sample data are available from 2014 [89].

230 The ventilation plant rooms on Levels 5 and 8 are treated separately due to having a common function and source term. The remaining rooms outside the secondary containment in the North Annexe are uncharacterised as they consist of offices, corridors, toilets, stores and electrical facilities. These uncharacterised rooms are assumed to be inactive (see INV-SGHWR-010).

South Annexe Miscellaneous Areas

231 Rooms outside the secondary containment in the South Annexe had a number of functions. The active workshops, ACW system, Boiler House Basement and associated areas are part of the South Annexe but are covered in their own sections. The approaches adopted for inventory derivation for the remaining rooms in the South Annexe and outside the secondary containment were as follows:

- The cooling water switch room (485) and the associated airlock (439) had no radiological processes occurring within them as they housed electrical distribution boards. A radiological transit route did exist via movements of personnel in and out of the secondary containment area through the airlock [90]. The inventory for these rooms was derived from a set of two paint and two concrete samples from the area [91]. Fingerprint FP-003 (D60 General Area waste) was applied.
- The Segmented Gamma Scanner (SGS) area (Room 470) incorporates the former area of both Rooms 470 and 466. The SGS has now ceased use and been removed; the room has been decontaminated and is used as a construction office. The inventory has been conservatively derived from a single historical core from 2005 from the mechanical workshop [58]. Fingerprint FP-003 was applied to derive missing radionuclides. The room has been surveyed to ensure no loose contamination.
- The new fuel room (458) amalgamates former rooms 458, 459, 460, 461, 462 and 463. This room has been decontaminated and is now the entry / exit airlock for the secondary containment, body monitors and health physics lab. There is no characterisation data for this room so the activities for the adjacent SGS area were conservatively adopted.
- The vehicle airlock (476) is used to admit vehicles to the secondary containment. The inventory has been derived from a single historical floor core from 2005 [58]; fingerprint FP-003 was applied to derive missing radionuclides.

- The Winfrith Abrasive Cleaning Machine (WACM) area (Room 480) incorporates the area of the former Rooms 480, 481 and 482. Room 480 previously held the auxiliary boilers which provided domestic heating and steam to the plant. The WACM has now been dismantled and removed, with the room in use as a construction materials store. The inventory has been derived from single historical core taken in 2005 from near one of the boilers [58]; fingerprint FP-003 was applied to derive missing radionuclides.

232 Remaining uncharacterised rooms in the South Annexe consist of the laundry and various switch rooms, toilets, labs, stores and offices. The majority of these rooms are in the office complex on Level 6. These uncharacterised rooms are assumed to be inactive (see INV-SGHWR-010).

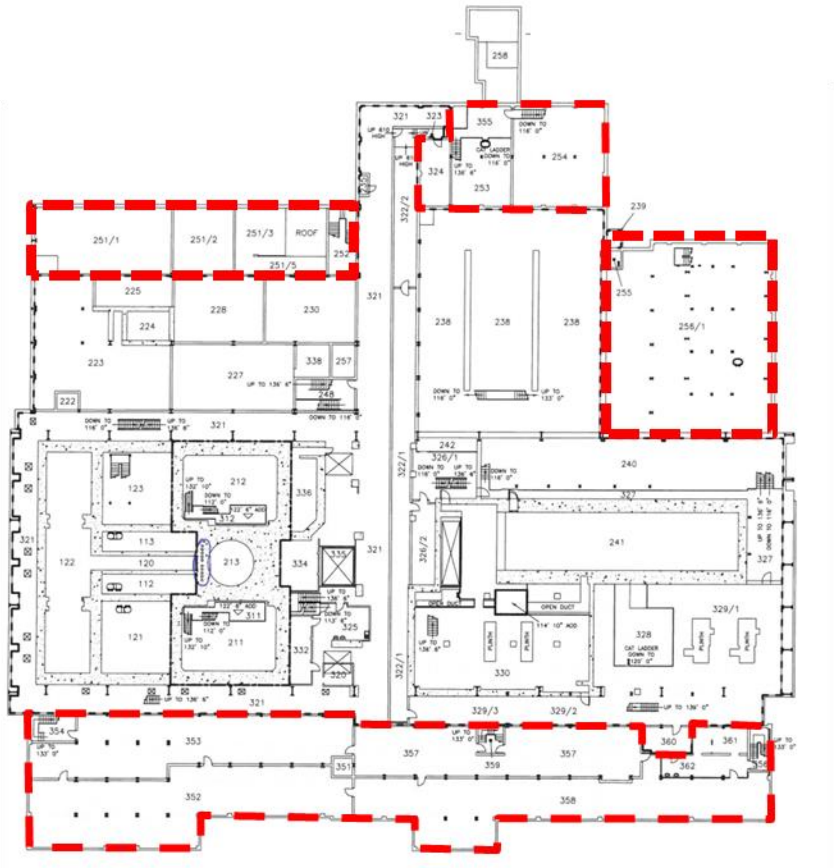


Figure 2.28: Examples of SGHWR ancillary areas on Levels 2 and 3. Indicated areas are Active Workshops (top left); the Fuel Oil Tank room, Boiler House Basement and airlock (top centre); ACW pumphouse basement (top right); and cable basements (bottom). Edited from [57].

2.15.3 Inventory Estimate

233 The inventory for the SGHWR ancillary areas was developed from the above data in [14] and underlying references. Maximum and average activity concentrations and an estimate of the radioactive inventory based on average activity concentrations are presented in Table **2.35**.

Table 2.35: SGHWR ancillary areas in-situ disposal inventory, including maximum and average activity concentrations, inventory based on average activity concentrations, and an alternative inventory based on maximum activity concentrations in each room (as discussed in Section 2.15.4). All data are presented for an inventory reference date of 01/01/2027.

Radionuclide	Maximum [Bq/g]	Average [Bq/g]	Disposal Inventory [MBq]	Alternative inventory [MBq]
^{3}H	6.34E+01	1.24E+00	2.35E+03	1.06E+04
^{14}C	7.99E-01	3.94E-02	7.45E+01	3.77E+02
^{134}Cs	5.00E-06	3.36E-07	6.35E-04	7.61E-04
^{137}Cs	1.95E+00	6.02E-02	1.14E+02	7.76E+02
^{57}Co	2.60E-11	2.44E-12	4.61E-09	4.94E-09
^{60}Co	1.41E-01	2.45E-03	4.64E+00	3.38E+01
^{241}Am	5.48E-02	3.71E-03	7.01E+00	3.34E+01
^{152}Eu	2.88E-02	1.85E-03	3.51E+00	4.67E+00
^{154}Eu	5.61E-03	4.12E-04	7.79E-01	9.34E-01
^{55}Fe	1.47E-03	4.56E-04	8.62E-01	1.19E+00
^{63}Ni	5.89E-01	2.68E-02	5.08E+01	1.68E+02
^{90}Sr	1.70E+00	5.47E-02	1.03E+02	6.55E+02
^{241}Pu	1.87E-01	2.00E-02	3.79E+01	1.06E+02
^{99}Tc	5.34E-05	4.46E-07	8.44E-04	1.15E-03
^{129}I	1.14E-04	9.56E-07	1.81E-03	2.45E-03
^{234}U	2.56E+00	8.55E-02	1.62E+02	9.82E+02
^{235}U	2.00E-01	6.13E-03	1.16E+01	7.52E+01
^{238}U	2.11E+00	8.39E-02	1.59E+02	8.45E+02
^{238}Pu	8.56E-01	1.74E-02	3.29E+01	3.17E+02
^{239}Pu	9.81E-01	2.07E-02	3.91E+01	3.65E+02
^{240}Pu	7.97E-01	1.70E-02	3.22E+01	2.97E+02
^{243}Cm	2.07E-06	1.75E-07	3.31E-04	3.31E-04
^{244}Cm	5.29E-02	1.04E-03	1.98E+00	1.95E+01
^{226}Ra	6.96E-01	2.05E-02	3.88E+01	3.39E+02
^{40}K	1.00E+00	5.52E-02	1.04E+02	5.64E+02
Sum	**7.81E+01**	**1.76E+00**	**3.33E+03**	**1.66E+04**

2.15.4 Sensitivity Analysis and Further Characterisation

234 The main source of uncertainty for the ancillary areas is the adequateness of the characterisation data (INV-SGHWR-006). The characterisation of rooms in the ancillary areas largely does not follow a DQO process for the individual rooms and there is not a statistically representative set of samples. Instead, characterisation has been undertaken based on plant and process knowledge to target areas of known or suspected contamination. The derived reference inventory conservatively assumes that

the samples can be treated as representative, scaling the average activities of the samples to derive the activity of each room.

235 The alternative inventory for the ancillary areas has been derived using the maximum activity measured for each radionuclide in each room and assumes this will be representative of the overall activity of the room. The alternative inventory is presented in Table **2.35**. The alternative inventory is 15,600 MBq, almost five times that of the inventory based on the average room activities. The individual radionuclide total activities increase by a factor of between 1 and 10, but tritium dominates both the disposal and alternative inventory estimates.

2.16 SGHWR Bulk Structure

2.16.1 Feature Description

236 It is observed that the core depths for many features in the SGHWR do not bound the tritium content as it is highly mobile in concrete. It is also observed that measurable tritium contamination is present in areas with no history of processes or activity that would lead to contamination. These observations suggest tritium has diffused throughout the SGHWR structure.

237 An inventory is therefore derived for the bulk volume of concrete in the SGHWR structure to account for tritium contamination that is not captured by the discrete features discussed in previous sections. This inventory entry comprises all of the SGHWR structure not explicitly captured by an existing inventory derivation:

- All uncharacterised rooms in the SGHWR. The majority of these rooms (159) are in the ancillary areas, with the remaining rooms largely in the secondary containment (57) and a single room with no inventory (the unused two element loop room) in the primary containment.
- Deeper intervals of structural materials not captured by core data in characterised rooms across the SGHWR.

2.16.2 Origin and Constraints on Radiological Inventory

Radiological Inventory

238 The available characterisation data comprises all of the tritium sample data for the SGHWR taken to-date. To derive an inventory for the bulk structure, a pragmatic approach is taken in which the assumed tritium contamination of the bulk structure is taken to be the median adopted tritium activity for characterised rooms that contain concrete in the SGHWR. This approach avoids the strong bias that would be introduced to a mean by the small number of very active rooms, but also includes all the source data in the derived value. The grouping of sample data into rooms rather than taking all the samples separately avoids biasing the result towards well characterised rooms underpinned by a disproportionate number of samples.

239 The sample data supporting the inventory are taken from a number of dates spanning the period 2005-2023; the median value was taken from the raw data without decay-correction to a common date. For subsequent decay to the inventory reference date (01/01/2027), the most recent sample date for any component in the SGHWR (01/01/2023) was taken as the overall sample date for the bulk structure. The 18-year

spread in sample dates will introduce some uncertainty (more than one half-life for tritium) into the derived value, but this will conservatively over-estimate the tritium content.

240 The tritium contamination of the bulk concrete of the primary containment structure is considered as a separate inventory entry, the derivation of which is discussed in Section 2.12.2.

241 The median tritium activity for components in the SGHWR is 0.814 Bq/g, which corresponds to one of the south cofferdams (Room 132). Following decay to the inventory reference date of 01/01/2027, the remaining tritium activity is 0.650 Bq/g.

Material Mass Balance

242 The total volume of SGHWR structural materials and the fraction above/below ground was derived from the calculations in the CSM [21, Tab.606/3]. To derive a bulk concrete value, the total volume of all other inventory entries (apart from backfill from the D60 stockpile; see Section 2.17) was subtracted from the estimated total volume of SGHWR structural materials. The estimated volumes of the SGHWR structural materials and radiological inventory are presented in Table **2.36** along with the unaccounted-for volume that is assumed to be tritium-contaminated bulk concrete. The approach to volume derivation neglects the different densities of other structural materials such as rebar and brick. This is expected to be conservative because:

- Tritium is not expected to diffuse significantly into metal, so neglecting the volume of rebar and other metals will be pessimistic.
- The adopted concrete density is higher than the adopted brick density, so the concrete density will be pessimistic for brick.
- For barytes concrete, the contamination per unit volume is expected to be equivalent to regular concrete (see Section 2.8), the adopted methodology will therefore be representative for barytes concrete.
- Other structural materials are expected to comprise only a small fraction of the disposal volume.

Table 2.36: The estimated total volumes of SGHWR structural materials and inventory entries above and below the assumed demolition datum (40.6 mAOD) and the derived bulk concrete volume. All volumes are for the intact structural materials.

	Total volume SGHWR structural materials [m^3]	Total volume SGHWR radiological inventory [m^3]	Assumed volume of contaminated SGHWR bulk concrete [m^3]
Above 40.6 mAOD	11,087	3,092	7,995
Below 40.6 mAOD	17,321	5,405	11,916
Total	**28,408**	**8,498**	**19,910**

2.16.3 Inventory Estimate

243 The inventory for the bulk SGHWR structure is derived from the total unaccounted-for volume of SGHWR structural materials, the density of structural concrete (Table **2.6**)

and the adopted median tritium activity at the inventory reference date (0.650 Bq/g). The total in-situ bulk tritium contamination disposal inventory is 18,600 MBq at 01/01/2027 and the total above ground disposal inventory is 12,500 MBq. The total estimated SGHWR bulk structural tritium contamination activity is 31,100 MBq.

2.16.4 Sensitivity Analysis and Further Characterisation

244 Uncertainty in the bulk SGHWR contamination estimate primarily stems from the nature of unaccounted-for contamination and the total activity. The reference estimate assumes that the only pathway for contamination of the unaccounted-for structure is via diffusion through the bulk concrete and therefore that tritium is the key contaminant of concern. This will be an accurate representation for deep intervals of concrete but the uncharacterised rooms throughout the SGHWR structure may be exposed to contamination through pathways such as the movement of personnel and exposure to the containment atmosphere (the probability of operational leaks, spills or other direct contamination is minimal as the majority of unaccounted-for areas did not house operations or active plant).

245 The alternative inventory considered here adopts the overall average activity concentrations for all radionuclides in the ancillary areas. The combined in-situ and backfill disposal inventory for the ancillary areas (i.e. the entire ancillary areas structure as it currently stands) was adopted for both the in-situ and backfill contributions for the bulk structure as the elevation of uncharacterised rooms and concrete is not expected to significantly affect the inventory. The ancillary areas are the least active feature of the SGHWR inventory and also contain the majority of the uncharacterised rooms. Given that inventories are derived for the subset of rooms in the ancillary areas that are expected to be more contaminated, this is expected to be pessimistic.

246 The adopted average activity and overall derived alternative inventory for the in-situ bulk SGHWR structure is presented in Table **2.37**. The overall average activity of the ancillary areas is lower than that of the ancillary areas in-situ disposal inventory (Table **2.35**) as the above-ground rooms are much less active on average. Tritium remains the primary component of the alternative inventory, with a slightly higher activity than the reference disposal inventory (0.76 Bq/g rather than 0.65 Bq/g). The overall activity is 1.24 Bq/g, with the primary contributing radionuclides being ^{3}H, ^{137}Cs, ^{234}U, ^{238}U and ^{40}K. The relatively high activities of ^{234}U and ^{238}U are mainly due to the large ratio of these radionuclides to ^{235}U in FP-003 (the majority of the ^{234}U and ^{238}U activities in the ancillary areas were derived from this fingerprint). The alternative total activities for the in-situ and above-ground components of the bulk structure are 35,400 MBq and 23,800 MBq respectively, giving a total of 59,200 MBq.

Table 2.37: Average activity and derived alternative in-situ disposal inventory for the SGHWR bulk structure. Presented at the inventory reference date of 01/01/2027.

Radionuclides	Average [Bq/g]	Alternative Inventory [MBq]
^{3}H	7.64E-01	2.18E+04
^{14}C	2.74E-02	7.85E+02
^{134}Cs	9.83E-07	2.81E-02
^{137}Cs	3.69E-02	1.05E+03
^{57}Co	6.82E-12	1.95E-07
^{60}Co	1.56E-03	4.47E+01
^{241}Am	3.35E-03	9.59E+01
^{152}Eu	5.31E-03	1.52E+02
^{154}Eu	1.18E-03	3.36E+01
^{55}Fe	3.50E-04	1.00E+01
^{63}Ni	1.83E-02	5.24E+02
^{90}Sr	2.87E-03	8.20E+02
^{241}Pu	1.61E-02	4.62E+02
^{99}Tc	5.05E-06	1.45E-01
^{129}I	1.08E-05	3.10E-01
^{234}U	8.44E-02	2.41E+03
$^{235}U^{\#}$	6.48E-03	1.85E+02
$^{236}U^{\#}$	1.65E-05	4.72E-01
^{238}U	7.93E-02	2.27E+03
^{238}Pu	9.24E-03	2.64E+02
^{239}Pu	1.11E-02	3.18E+02
^{240}Pu	9.17E-03	2.62E+02
^{242}Pu	7.34E-06	2.10E-01
^{242}Cm*	1.97E-14	5.62E-10
^{243}Cm	5.86E-07	1.68E-02
^{244}Cm	5.54E-04	1.58E+01
^{252}Cf*	1.49E-07	4.25E-03
^{226}Ra	2.35E-02	6.72E+02
^{40}K	1.12E-01	3.21E+03
Total	**1.24E+00**	**3.54E+04**

^{235}U and ^{236}U originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

* ^{242}Cm and ^{252}Cf originally reported as a combined activity. For the purpose of this inventory estimate, it is conservatively assumed that the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.17 SGHWR Backfill

2.17.1 Feature Description

247 Following demolition of the SGHWR above-ground structure (the demolition datum is assumed to be 40.6 mAOD, or 1 m below ground level), the rooms on Levels 1-3 will become below-ground voids [21]. It is intended that these voids will, subject to risk assessment and regulatory acceptance, be filled with clean and/or radiologically contaminated materials originating from site decommissioning work prior to capping. The below-ground voids are grouped into a number of regions as indicated in Figure **2.29**:

- Region 1, which includes the primary containment, ponds, effluent vault and cofferdams
- Region 2, which consists of the delay tank rooms, condenser cell (referred to as turbine hall in Figure **2.29**) and part of the steam labyrinth.
- The North Annexe.
- The South Annexe, which also includes the pump pit.

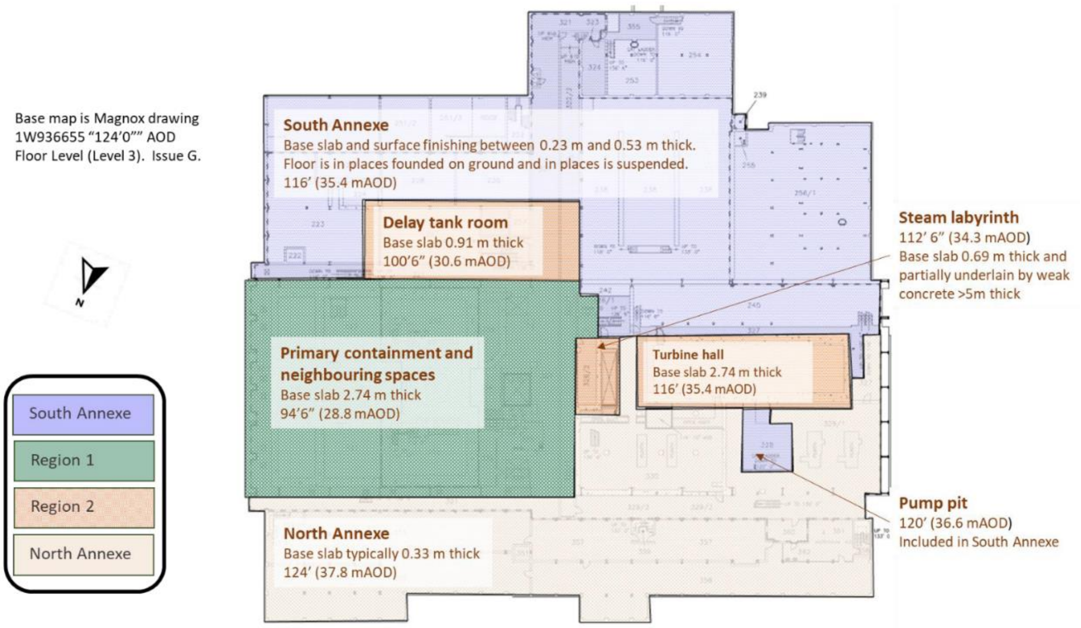


Figure 2.29: SGHWR below-ground voids conceptualised following demolition of above-ground building structure [21, Fig.606/4].

248 There are three potential on-site sources of contaminated materials in consideration for disposal to the SGHWR voids. In order of priority these include:

1. concrete blocks and brick/concrete rubble from demolition of the above-ground (Levels 4-10) SGHWR structure;
2. rubble from stockpiles already outside the SGHWR; and
3. other Winfrith site wastes.

249 Currently the need for use of 'other Winfrith site wastes' has not been identified as current volume estimates suggest stockpile rubble will still be left over after backfilling.

As such, the contaminated backfill inventory considered in this report is assumed to be limited to the first two options only (INV-SGHWR-007).

2.17.2 Origin and Constraints on Radiological Inventory

Above-ground SGHWR Structure

250 The above-ground SGHWR structure comprises portions of the main building features and components discussed previously in this report. Features with portions above 40.6 mAOD include the primary containment, the secondary containment, ancillary areas and the bulk structure. The bioshield, mortuary tubes and ponds are entirely below the demolition datum and will not contribute to the backfill inventory.

251 The Level 4 floor level is 41.6 mAOD, although some rooms have different floor levels in the range 39.2 to 42.4 mAOD. It is assumed that the entirety of the inventory for all rooms on Level 4 and above will contribute to the backfill.

252 Rooms on Levels 1-3 are generally assumed not to be demolished and so do not contribute to the backfill. Some features below Level 4 extend up multiple levels to above the 41.6 mAOD level. These areas are identified from the Level 4 floor plans. An estimate has been made for each of these areas on a case-by case basis as to the proportion of the inventory which can be considered above or below the demolition datum (this generally included all of the floor area below the demolition datum and partitioned the walls based on their total height). Rooms for which this approach is taken include the feed heater cell, boiler feed pump area and the primary containment main space. The inventory contributions of the above-ground portions of the SGHWR features to the backfill are shown in Table **2.38**.

Table 2.38: Contributions of the SGHWR features to the backfill inventory at a reference date of 01/01/2027. Not shown are the mortuary tubes, bioshield and ponds, which are entirely disposed of in-situ. The rubble mounds make a further contribution to the backfill inventory, as discussed below.

Radionuclide	Primary (MBq)	Secondary (MBq)	Ancillary areas (MBq)	Bulk structure (MBq)
^{3}H	2.71E+04	8.33E+03	4.94E+02	1.25E+04
^{14}C	1.04E+03	6.91E+02	2.78E+01	-
^{134}Cs	4.54E-03	4.10E-02	3.03E-03	-
^{137}Cs	2.03E+03	1.71E+03	2.37E+01	-
^{57}Co	2.04E-08	3.15E-03	2.08E-08	-
^{60}Co	7.60E+01	1.19E+01	1.19E+00	-
^{241}Am	3.00E+00	5.60E+01	5.48E+00	-
^{94}Nb	-	2.73E+01	-	-
^{125}Sb	-	6.57E-01	-	-
^{152}Eu	2.49E+01	1.61E+02	1.63E+01	-
^{154}Eu	3.79E+00	2.51E+01	3.60E+00	-
^{155}Eu	-	2.26E+00	-	-
^{55}Fe	6.01E+00	1.03E+01	4.44E-01	-

Radionuclide	Primary (MBq)	Secondary (MBq)	Ancillary areas (MBq)	Bulk structure (MBq)
^{63}Ni	9.24E+02	1.07E+02	1.76E+01	-
^{90}Sr	1.93E+01	5.93E+01	3.40E+00	-
^{241}Pu	1.45E+01	2.25E+02	2.23E+01	-
^{133}Ba	4.97E+00	-	-	-
^{99}Tc	1.32E+01	9.32E-02	1.80E-02	-
^{129}I	-	2.00E-01	3.86E-02	-
^{233}U	1.77E+00	1.19E+00	-	-
^{234}U	1.22E+00	2.90E+02	1.53E+02	-
^{235}U$^{\#}$	9.46E+00	2.96E+01	1.26E+01	-
^{236}U$^{\#}$	-	7.74E-01	6.16E-02	-
^{238}U	1.21E+00	2.59E+02	1.37E+02	-
^{238}Pu	5.26E-01	1.56E+01	1.53E+00	-
^{239}Pu	4.07E-01	1.85E+01	2.32E+00	-
^{240}Pu	3.38E-01	1.54E+01	2.04E+00	-
^{242}Pu	1.35E-01	1.40E+00	2.74E-02	-
^{242}Cm*	1.15E-06	9.65E-08	7.33E-11	
^{243}Cm	7.67E-03	2.62E-02	1.85E-03	-
^{244}Cm	3.03E-01	1.19E+00	8.84E-02	-
^{252}Cf*	2.08E-02	3.09E-02	5.54E-04	-
^{226}Ra	4.86E+01	1.01E+02	4.88E+01	-
^{40}K	6.28E+01	4.55E+02	3.13E+02	-
Total	**3.14E+04**	**1.26E+04**	**1.29E+03**	**1.25E+04**

^{235}U and ^{236}U originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

* ^{242}Cm and ^{252}Cf originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

Rubble Stockpiles

253 Mechanically-broken site-derived brick and concrete demolition rubble is currently stockpiled approximately 100 m to the east of the SGHWR in four stockpiles (Figure **2.30** and Figure **2.31**).

254 The origin of material within Stockpiles #1 and #2, which is estimated at 16,800 m^3 [92], is from mixed locations across the Winfrith site and includes inactive and potentially active material [93; 94]. Active material entering the stockpiles was radiologically 'cleared' through the Exploranium gate monitor prior to storage and was deemed to satisfy the RSA 93 Substances of Low Activity (SoLA) Exemption Order. Since this time Schedule 23 of EPR 2016 has been implemented (and subsequently amended) with revised assessment criteria for determining whether material is OoS of RSR. INV-SGHWR-006 and INV-SGHWR-008 capture the uncertainty relating to

whether the rubble in the mounds is OoS. Weights of materials deposited and removed were estimated at the time of storage/recovery, mostly based on lorry movements.

255 Material stockpiled in Stockpile #3 comprises approximately 3,500 m^3 broken concrete and brick derived from the demolition of the A51 and A52 facilities. Comprehensive pre-demolition characterisation of the material was undertaken and the rubble contained within this stockpile is considered to be OoS under the EPR sentencing criteria [95].

256 Material stockpiled in Stockpile #4 was sourced from the D63/D64 Cooling Tower Basins [96] and estimated to be around 1,400 m^3 [92]. Two notes for the record [97; 98] describing the pre-demolition characterisation of the slabs indicate the material is OoS.

Figure 2.30: Aerial view of Winfrith Rubble Stockpiles 1 - 4 [99].

Figure 2.31: Main Rubble Stockpile (Stockpile 1) [99].

257 A full programme of characterisation for the stockpiles is planned prior to sentencing for final disposal [99]. However, between May and June 2018 a preliminary programme of characterisation was undertaken in support of on-going technical and optioneering studies [93]. This programme involved a 'Groundhog Fusion' gamma radiation survey to identity surface/near-surface areas of potentially elevated activity, followed by sampling at depths between 0.2 and 2.9 m from 21 machine-excavated trial pits. Nineteen of the locations were selected to provide a reasonable level of geographical coverage across the four stockpiles and were essentially non-judgemental sampling locations. Two targeted locations were selected on the basis of the results of the Groundhog survey work. Location RS/12 targeted a Groundhog anomaly. The location of the anomaly is shown in Figure **2.32** and appears to be associated with some black geo-textile material. The material was also present in one of the other trial pits but with no associated contamination. The view on site is that this material was from remediation works associated with the A59 building where it was used to create a 'platform' onto which rubble was characterised. The second targeted sample location was chosen to cover an area where no Groundhog data had been collected [93].

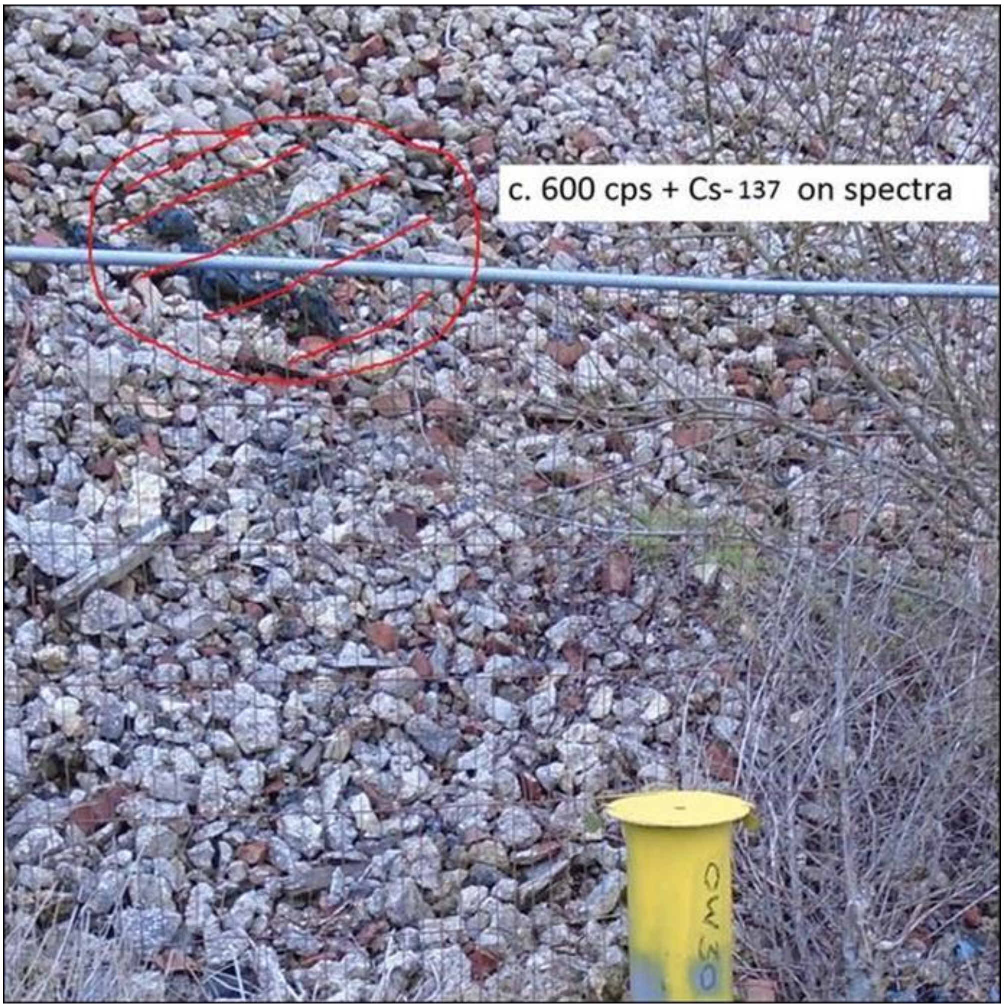


Figure 2.32: Rubble mound sample location RS/12, indicating elevated radiation, potentially associated with the black material [100].

258 The excavated material was first placed in a waste drum prior to assessment using the on-site SGS assay system. With the exception of drum RS/12, all of the filled drums had a contact dose rate of 0.1 μSv/hr. Drum RS/12 had a contact dose rate of 0.6 μSv/hr. When the gamma data was integrated with a range of site fingerprints only drum RS/12 was determined to have levels of artificial activity that may require designation as 'In Scope'; however, no formal OoS assessment was produced. Drum RS/12 was not opened and the drum and material were disposed of as waste; as such, the source of the activity was not confirmed.

259 A total of 19 samples in 2018 were subject to gross alpha / beta and high-resolution gamma spectrometry analysis. Five samples were also subject to total tritium and ^{14}C analysis. The data, in Bq/g, are summarised in Table **2.39**.

Table 2.39: Key radiological analysis for rubble mound samples. All results are in Bq/g for sample count dates on 23-25 October 2018 [93; 94]. A number of additional gamma emitters are omitted where a natural origin is suspected and/or all results reported were LOD.

	Gross Alpha (as ^{241}Am)	**Gross Beta (as ^{40}K)**	**Total Tritium**	**^{14}C**	**^{241}Am**	**^{57}Co**	**^{60}Co**	**^{134}Cs**	**^{137}Cs**	**^{152}Eu**	**^{154}Eu**	**^{155}Eu**	**^{125}Sb**	**^{235}U**
n	19	19	5	5	19	19	19	19	19	19	19	19	19	19
n > LOD	19	19	1	4	2	0	0	0	17	0	0	0	0	10
Maximum	0.249	0.236	0.126	0.006	0.006	<0.0007	<0.0020	<0.0015	0.03540	<0.0019	<0.0015	<0.0057	<0.0067	<0.0022
Average	0.148	0.121	0.041	0.005	0.002	<0.0005	<0.0007	<0.0007	0.00598	<0.0013	<0.0009	<0.0030	<0.0031	0.0012
Minimum	0.107	0.048	<0.020	0.004	<0.0008	<0.0003	<0.0002	<0.0002	<0.00055	<0.0006	<0.0006	<0.0015	<0.0009	<0.0006

260 The radiological analysis from the analysed drums shows very low levels of artificial activity in all samples and an additional component from naturally-occurring radionuclides (e.g. ^{228}Ac, ^{226}Ra, ^{235}U).

261 The most appropriate fingerprint for the rubble mounds was identified as FP-004 (A59 combined fingerprint), as A59 is a significant source of the potentially active material in the main rubble mounds. Table **2.40** presents FP-004 along with a scaled version of the fingerprint that just satisfies the EPR 2016 OoS sum of quotients levels [101; 102], which is the activity level at which more recent additions to the rubble mounds were 'cleared'[14]. The table also presents maximum activities for determinands for samples from the 2018 sampling campaign. Comparison of available analytical results against the scaled FP-004 activity indicates measured values are lower for ^{241}Am, ^{137}Cs and ^{60}Co. Measured values for ^{125}Sb are above the scaled FP-004 activity; however, all ^{125}Sb results were below the LOD. Both values are much lower than the OoS level for ^{125}Sb of 0.1 Bq/g. Measured ^{235}U exceeds the scaled FP-004 activity; however, this is likely to reflect a significant naturally-occurring component from the concrete. All values are well below the OoS level for ^{235}U of 1 Bq/g. Both ^{3}H and ^{14}C are not in the FP-004 fingerprint but were measured in some of the samples at detectable levels. The analysis suggests that the FP-004 fingerprint is generally appropriate for application to the rubble mounds, but that the inventory of some radionuclides needs estimating by an alternative approach.

262 None of the results indicate that the bulk material is classified as in-scope of radioactive substances regulation [93], although the sampling undertaken to date is not sufficient to be statistically representative of all the rubble mound material (INV-SGHWR-006).

Table 2.40: FP-004 (A59 combined fingerprint) as activity percentages on 24/10/2018 and scaled to the EPR 2016 OoS criteria. The maximum measured activity from the rubble mounds is also shown for comparison.

Radionuclides	Half-Life	FP-004 (A59 Combined Fingerprint)	FP-004 scaled to EPR16, Sch.23, Tab.2 OoS	Maximum Activity from 2018 Rubble Sampling
	[yr]	[%]	[Bq/g]	[Bq/g]
Alpha Isotopes				
^{234}U	2.46E+05	2.33E-04	1.85E-06	
^{235}U	7.04E+08	1.16E-04	9.27E-07	2.20E-03
^{238}U	4.47E+09	1.40E-03	1.04E-05	
^{238}Pu	8.77E+01	1.45E-01	1.16E-03	
^{239}Pu	2.41E+04	6.80E-01	5.41E-03	
^{240}Pu	6.56E+03	9.46E-01	7.54E-03	
^{244}Cm	1.81E+01	6.45E-01	5.14E-03	
^{241}Am	4.32E+02	1.45E+00	1.15E-02	6.40E-03

[14] The RSA93 SoLA exemption level of 0.4 Bq/g against which earlier additions to the rubble mounds were 'cleared' was also considered for scaling FP-004, but this led to a less conservative total activity than scaling to the EPR 2016 OoS levels.

Radionuclides	Half-Life	FP-004 (A59 Combined Fingerprint)	FP-004 scaled to EPR16, Sch.23, Tab.2 OoS	Maximum Activity from 2018 Rubble Sampling
Subtotal		*3.9*	*0.03*	
Non-alpha Isotopes				
^{3}H	1.23E+01	-	-	1.26E-01
^{14}C	5.70E+03	-	-	6.00E-03
^{63}Ni	1.00E+02	4.09E+00	3.26E-02	
^{60}Co	5.27E+00	4.73E-01	3.77E-03	2.00E-03
^{90}Sr	2.88E+01	2.33E+01	1.85E-01	
^{125}Sb	2.76E+00	9.34E-04	7.44E-06	6.70E-03
^{137}Cs	3.02E+01	6.43E+01	5.12E-01	3.54E-02
^{241}Pu	1.44E+01	3.99E+00	3.18E-02	
Subtotal		*96.1*	*0.77*	
Total		**100**	**0.80**	

Material Mass Balance

263 Four voids are assumed to be generated as part of the demolition activities (Figure **2.29** and Table **2.41**) with a total volume of 29,739 m^3. It should be noted that the volumes of these voids, and of the backfill constituents, are only treated here at a conceptual level and that the values presented may be subject to change as optioneering and demolition work progress (INV-SGHWR-007).

264 The intact volume of the above-ground structure (concrete and brick) of the SGHWR is intended to be demolished and used as backfill. In places the structure will be demolished using wireline cutting and the resulting concrete blocks will be placed in Region 1. The remaining structure will be demolished using conventional demolition techniques and the resulting rubble will be compacted to fill part of the remaining void space.

265 The demolition is expected to generate 6,300 m^3 of blocks and 5,840 m^3 of compacted rubble. The remaining 17,599 m^3 of void space is to be filled using material from the rubble mounds. The total volume of the rubble mounds is estimated to be 21,723 m^3, which is expected to be reduced by approximately 18% on compaction during emplacement. However, given the uncertainties in volumetric estimates of void and backfill, and in the packing efficiency and degree of compaction, it is unlikely that a precise understanding of the material balance will be achieved until implementation of demolition and disposal. Therefore, it is conservatively assumed for the purposes of the inventory calculations that all the above-ground structure and stockpiled material will be disposed of in the SGHWR voids.

Table 2.41: Key SGHWR void sizes and volumes of material assumed to fill the voids (adapted from [21, Tab.606/7]).

Constituent	Void Volume (m3)	Demolition Block Volume (m3)	Volume available for demolition arisings (m3)
SGHWR Region 1	11,649	6,300	5,349
SGHWR Region 2	3,425	0	3,425
SGHWR North Annexe	4,164	0	4,164
SGHWR South Annexe	10,501	0	10,501
Total Volume	**29,739**	**6,300**	**23,439**

266 The mass of each inventory disposal feature and the in-situ and backfill mass contributions are presented in Table **2.42**. Note that the table includes only masses of assumed contaminated/activated material associated with each feature. The bulk SGHWR structure entry accounts for the estimated remaining unaccounted-for mass of the SGHWR building, which includes parts of the secondary containment, ponds, and ancillary areas.

Table 2.42: SGHWR contaminated/activated material mass in each feature of the disposal inventory and its contribution to the backfill [14].

Feature	Total mass of inventory feature (t)	Mass disposed of in-situ (t)	Mass to backfill (t)
Bioshield	765	765	-
Mortuary tubes	3	3	-
Primary containment	7,614	4,961	2,653
Secondary containment	7,082	4,209	2,873
Ponds	1,165	1,165	-
Ancillary areas	3,728	1,891	1,837
Bulk SGHWR structure	47,785	28,598	19,187
Rubble mounds	34,621	-	34,621
Total	**102,762**	**41,591**	**61,171**

2.17.3 Inventory Estimate

267 The available characterisation data for the rubble mounds suggests that the majority of the material is at OoS levels with respect to EPR 2016 (INV-SGHWR-008), although this will not be confirmed until the time of disposal. For the purposes of the SGHWR disposal inventory it is assumed that the entirety of the material is at OoS levels as determined from the A59 fingerprint FP-004 (Table **2.40**), but also including a contribution of ^{3}H, ^{14}C and ^{235}U based on average activities from the 2018 rubble sampling (Table **2.39**).

268 The remainder of the backfill radiological inventory of the SGHWR comprises the demolished above-ground (Level 4-10) SGHWR structure.

269 The exact details of the backfilling process are not yet confirmed and, given the uncertainties in volumetric estimates of voidage and backfill and other variables, it is unlikely that a precise understanding of the material balance will be achieved until implementation of demolition and disposal.

270 Average and maximum activity concentrations and an estimate of the radioactive inventory for the backfill comprising both rubble and the SGHWR above-ground structure are presented in Table **2.43**. Maximum activity concentrations were derived directly by taking a maximum of rubble mound and SGHWR Level 4-10 data [14]. Average activities were calculated by dividing the total inventory by the estimated material mass.

Table 2.43: SGHWR backfill disposal inventory, including maximum and average activity concentrations and disposal inventory based the on average activity concentrations, presented for an inventory reference date of 01/01/2027.

Radionuclide	Maximum (Bq/g)	Average (Bq/g)	Disposal Inventory (MBq)
^{3}H	1.20E+03	8.06E-01	4.93E+04
^{14}C	4.77E+01	3.16E-02	1.93E+03
^{134}Cs	2.13E-03	7.94E-07	4.86E-02
^{137}Cs	8.23E+02	3.02E-01	1.84E+04
^{57}Co	2.24E-05	5.15E-08	3.15E-03
^{60}Co	7.02E+01	2.18E-03	1.34E+02
^{241}Am	1.11E+00	7.69E-03	4.70E+02
^{94}Nb	1.36E+01	4.47E-04	2.73E+01
^{125}Sb	2.12E-03	1.13E-05	6.89E-01
^{152}Eu	6.16E+01	3.30E-03	2.02E+02
^{154}Eu	8.59E+00	5.31E-04	3.25E+01
^{155}Eu	8.21E-01	3.70E-05	2.26E+00
^{55}Fe	1.35E+00	2.75E-04	1.68E+01
^{63}Ni	4.48E+02	3.46E-02	2.11E+03
^{90}Sr	7.67E+00	8.75E-02	5.35E+03
^{241}Pu	3.03E+00	1.64E-02	1.00E+03
^{133}Ba	1.88E-01	8.12E-05	4.97E+00
^{99}Tc	7.53E-02	2.18E-04	1.34E+01
^{129}I	1.61E-01	3.89E-06	2.38E-01
^{233}U	1.00E-02	4.84E-05	2.96E+00
^{234}U	7.54E+00	8.67E-03	5.31E+02
^{235}U$^{\#}$	3.50E+00	1.55E-03	9.47E+01
^{236}U$^{\#}$	5.00E-03	1.37E-05	8.36E-01
^{238}U	8.61E+00	1.49E-02	9.13E+02
^{238}Pu	3.26E-01	9.01E-04	5.51E+01
^{239}Pu	4.49E-01	3.41E-03	2.09E+02
^{240}Pu	4.49E-01	4.55E-03	2.78E+02
^{242}Pu	5.00E-03	2.55E-05	1.56E+00

Radionuclide	Maximum (Bq/g)	Average (Bq/g)	Disposal Inventory (MBq)
^{242}Cm*	7.58E-09	2.04E-11	1.25E-06
^{243}Cm	3.15E-04	5.84E-07	3.57E-02
^{244}Cm	1.24E-02	2.15E-03	1.32E+02
^{252}Cf*	4.25E-04	8.55E-07	5.23E-02
^{226}Ra	1.49E+01	3.24E-03	1.98E+02
^{40}K	1.90E+01	1.36E-02	8.31E+02
Sum	**2.74E+03**	**1.35E+00**	**8.23E+04**

/ * ^{235}U and ^{236}U (^{242}Cm and ^{252}Cf) originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.17.4 Sensitivity Analysis and Further Characterisation

271 The backfill consists of the above-ground portions of the primary containment, secondary containment, ancillary areas and bulk SGHWR structure (i.e. uncharacterised areas) as well as the rubble mounds. The uncertainty in the disposal inventory associated with the backfill is therefore composed of the uncertainty in the activities and volumes of these contributing features and components. To address these uncertainties the alternative inventory for the backfill makes the following assumptions:

- The inventory contributions from the primary containment, secondary containment, ancillary areas and SGHWR bulk structure assume the alternative inventories for these features.
- The inventory for the rubble mounds is derived assuming the mounds have uniform activity equal to the maximum measured activity for each radionuclide, or activity at the EPR 2016 OoS scaled fingerprint (whichever is higher).
- Each feature is assumed to contribute an additional 10% contaminated material volume than estimated in the reference inventory (equating to 63,600 tonnes of backfill rather than 57,900 tonnes). This is not assumed to affect the in-situ disposal inventories and will result in some double counting of volume.

272 Table **2.44** presents the contributions to the alternative backfill inventory for each feature and the rubble mounds.

273 Slightly over half of the backfill activity is from the primary containment, with the remainder of activity contributed primarily by the rubble mounds, bulk structure and secondary containment. The ancillary areas contribute less than 1% of the backfill activity. The average activity of the backfill is 3.5 Bq/g, the majority of which is tritium. ^{137}Cs and ^{14}C are the second and third most active radionuclides, contributing 23% and 4% of the inventory respectively.

Table 2.44: Activity contributions for each feature and the rubble mounds to the alternative backfill inventory. Presented for an inventory reference date of 01/01/2027.

Radionuclide	Primary [MBq]	Secondary [MBq]	Ancillary areas [MBq]	Rubble mounds [MBq]	Bulk structure [MBq]	Backfill Disposal Inventory [MBq]
^{3}H	9.72E+04	1.08E+04	6.29E+02	2.75E+03	1.47E+04	1.26E+05
^{14}C	3.53E+03	2.77E+03	4.02E+01	2.08E+02	5.26E+02	7.08E+03
^{134}Cs	1.28E-02	1.01E-01	4.08E-03	0.00E+00	1.89E-02	1.37E-01
^{137}Cs	1.93E+04	1.04E+04	7.25E+01	1.47E+04	7.08E+02	4.52E+04
^{57}Co	9.70E-08	7.28E-03	2.66E-08	0.00E+00	1.31E-07	7.28E-03
^{60}Co	5.91E+02	5.07E+01	2.67E+00	4.45E+01	3.00E+01	7.18E+02
^{241}Am	7.67E+00	7.14E+01	8.13E+00	4.06E+02	6.43E+01	5.57E+02
^{94}Nb	0.00E+00	6.83E+01	0.00E+00	0.00E+00		6.83E+01
^{125}Sb	0.00E+00	8.60E-01	0.00E+00	3.29E-02		8.93E-01
^{152}Eu	9.54E+01	3.64E+02	2.49E+01	0.00E+00	1.02E+02	5.86E+02
^{154}Eu	9.69E+00	5.34E+01	4.93E+00	0.00E+00	2.26E+01	9.06E+01
^{155}Eu	0.00E+00	4.85E+00	0.00E+00	0.00E+00		4.85E+00
^{55}Fe	1.20E+01	1.51E+01	1.49E+00	0.00E+00	6.72E+00	3.54E+01
^{63}Ni	2.67E+03	3.08E+02	6.75E+01	1.07E+03	3.52E+02	4.47E+03
^{90}Sr	4.33E+01	2.20E+02	6.47E+00	5.27E+03	5.50E+02	6.09E+03
^{241}Pu	3.07E+01	2.84E+02	4.10E+01	7.41E+02	3.10E+02	1.41E+03
^{133}Ba	4.43E+01	0.00E+00	0.00E+00	0.00E+00		4.43E+01
^{99}Tc	1.65E+01	2.27E-01	2.15E-02	0.00E+00	9.70E-02	1.68E+01
^{129}I	0.00E+00	4.87E-01	4.61E-02	0.00E+00	2.08E-01	7.41E-01
^{233}U	2.36E+00	1.19E+00	0.00E+00	0.00E+00		3.54E+00
^{234}U	1.88E+00	4.18E+02	2.85E+02	1.52E+02	1.62E+03	2.48E+03
$^{235}U^{\#}$	5.86E+01	4.06E+01	2.20E+01	7.62E+01	1.24E+02	3.22E+02
$^{236}U^{\#}$	0.00E+00	7.74E-01	6.16E-02	0.00E+00	3.17E-01	1.15E+00
^{238}U	1.65E+00	3.71E+02	2.78E+02	9.14E+02	1.52E+03	3.09E+03
^{238}Pu	1.21E+00	1.95E+01	2.52E+00	3.75E+01	1.77E+02	2.38E+02
^{239}Pu	1.62E+00	2.37E+01	5.18E+00	1.87E+02	2.13E+02	4.31E+02
^{240}Pu	1.33E+00	1.98E+01	4.84E+00	2.61E+02	1.76E+02	4.63E+02
^{242}Pu	2.36E-01	1.42E+00	2.74E-02	0.00E+00	1.41E-01	1.83E+00
^{242}Cm*	1.79E-06	9.65E-08	7.33E-11	0.00E+00	3.77E-10	1.88E-06
^{243}Cm	7.43E-02	3.34E-02	2.21E-03	0.00E+00	1.12E-02	1.21E-01
^{244}Cm	2.93E+00	1.52E+00	1.06E-01	1.30E+02	1.06E+01	1.45E+02
^{252}Cf*	3.23E-02	3.09E-02	5.54E-04	0.00E+00	2.85E-03	6.67E-02
^{226}Ra	2.40E+02	1.47E+02	6.40E+01	0.00E+00	4.51E+02	9.02E+02
^{40}K	5.14E+02	6.83E+02	5.01E+02	0.00E+00	2.15E+03	3.39E+03

Radionuclide	Primary [MBq]	Secondary [MBq]	Ancillary areas [MBq]	Rubble mounds [MBq]	Bulk structure [MBq]	Backfill Disposal Inventory [MBq]
Total	**1.24E+05**	**2.71E+04**	**2.06E+03**	**2.69E+04**	**2.38E+04**	**2.04E+05**

/ * ^{235}U and ^{236}U (^{242}Cm and ^{252}Cf) originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

2.18 SGHWR Inventory Summary

2.18.1 1Summary Tables

274 Estimates for the maximum and average activity concentrations and radiological inventory for the different SGHWR features have been compiled from the previous sections and are presented in Table **2.45**, Table **2.46** and Table **2.47**, respectively. The inventory is presented at the inventory reference date of 01/01/2027.

275 In the following tables, # and * denote that ^{235}U and ^{236}U, or ^{242}Cm and ^{252}Cf, were originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

Table 2.45: SGHWR maximum activity concentrations (Bq/g) summary for in-situ features and backfill presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted red. An inventory for in-situ bulk SGHWR tritium contamination is not included as a maximum activity concentration has not been derived.

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Second-ary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill
^{3}H	8.02E+03	3.24E+02	2.47E+02	7.42E+02	2.40E+00	6.34E+01	1.20E+03
^{14}C	4.29E+00	3.48E+01	1.50E+01	4.01E+01	7.99E-01	7.99E-01	4.77E+01
^{134}Cs	4.57E-04	-	2.13E-03	1.95E-02	-	5.00E-06	2.13E-03
^{137}Cs	5.18E+02	2.94E+03	7.30E+02	9.14E+02	2.92E+03	1.95E+00	8.23E+02
^{57}Co	-	3.50E-06	5.87E-09	2.24E-05	-	2.60E-11	2.24E-05
^{60}Co	9.96E+00	4.75E+01	7.02E+01	5.86E+02	1.75E+01	1.41E-01	7.02E+01
^{241}Am	3.19E+00	1.63E+02	1.11E+00	1.01E+01	1.69E+02	5.48E-02	1.11E+00
^{94}Nb	-	1.46E+00	-	1.70E-01	5.40E-01	-	1.36E+01
^{125}Sb	-	3.19E-02	-	2.07E-02	-	-	2.12E-03
^{152}Eu	1.23E+02	1.60E+00	9.50E+00	2.78E+01	-	2.88E-02	6.16E+01
^{154}Eu	6.31E+00	6.18E+00	1.24E+00	3.51E+00	5.65E+00	5.61E-03	8.59E+00
^{155}Eu	2.16E-01	4.63E-01	-	7.55E-02	3.56E-01	-	8.21E-01
^{55}Fe	8.06E+01	1.12E+02	1.35E+00	1.81E+01	2.41E-01	1.47E-03	1.35E+00
^{63}Ni	6.36E+01	2.98E+02	4.48E+02	2.93E+03	2.23E+01	5.89E-01	4.48E+02
^{90}Sr	8.18E-02	4.43E+03	1.83E-01	5.15E+01	3.16E+03	1.70E+00	7.67E+00
^{241}Pu	7.11E-02	5.25E+02	3.03E+00	9.88E+00	5.44E+02	1.87E-01	3.03E+00
^{133}Ba	2.60E+01	5.32E-03	1.88E-01	-	-	-	1.88E-01
^{99}Tc	-	-	7.00E-02	1.03E+00	-	5.34E-05	7.53E-02

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Second-ary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill
^{129}I	-	-	-	2.20E+00	2.00E-01	1.14E-04	1.61E-01
^{36}Cl	-	-	-	-	1.00E-01	-	-
^{233}U	-	-	1.00E-02	6.00E-03	5.00E-01	-	1.00E-02
^{234}U	-	5.36E+00	8.00E-03	8.84E+01	3.50E+00	2.56E+00	7.54E+00
$^{235}U^{\#}$	1.20E+01	2.27E-01	3.50E+00	6.80E+00	3.90E-01	2.00E-01	3.50E+00
$^{236}U^{\#}$	-	2.27E-01	-	2.00E-02	3.90E-01	-	5.00E-03
^{238}U	-	1.70E+00	7.00E-03	7.25E+01	1.00E+00	2.11E+00	8.61E+00
^{238}Pu	7.64E-03	5.74E+00	3.26E-01	3.15E+00	3.50E+00	8.56E-01	3.26E-01
^{239}Pu	1.53E-03	2.56E+02	6.53E-02	3.56E+00	8.92E+01	9.81E-01	4.49E-01
^{240}Pu	1.53E-03	2.09E+02	6.52E-02	2.89E+00	7.27E+01	7.97E-01	4.49E-01
^{242}Pu	-	-	1.00E-03	5.00E-03	3.00E-01	-	5.00E-03
^{242}Cm*	-	-	7.58E-09	2.19E-09	2.11E-10	-	7.58E-09
^{243}Cm	-	9.44E-03	3.15E-04	4.19E-03	5.04E-03	2.07E-06	3.15E-04
^{244}Cm	-	3.54E-01	1.24E-02	2.14E-01	1.91E-01	5.29E-02	1.24E-02
^{252}Cf*	-	-	1.37E-04	4.25E-04	1.34E-04	-	4.25E-04
^{226}Ra	-	-	1.49E+01	2.77E+01	-	6.96E-01	1.49E+01
^{40}K	-	-	-	7.00E+01	-	1.00E+00	1.90E+01
^{39}Ar	1.10E+01	-	-	-	-	-	-
^{41}Ca	2.30E+01	-	-	-	-	-	-
^{113m}Cd	6.17E-01	8.58E-01	-	-	-	-	-
^{151}Sm	2.04E+01	-	-	-	-	-	-
^{204}Tl	6.11E-01	8.50E-01	-	-	-	-	-
Total	**8.92E+03**	**9.37E+03**	**1.55E+03**	**5.62E+03**	**7.01E+03**	**7.81E+01**	**2.74E+03**

Table 2.46: SGHWR average activity concentrations (Bq/g) summary for in-situ features and backfill presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted red. Not shown is the in-situ bulk structure, which has an average activity of 0.650 Bq/g of tritium only.

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Second-ary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill
^{3}H	4.03E+02	1.61E+02	1.04E+01	1.33E+01	8.20E-01	1.24E+00	8.06E-01
^{14}C	1.10E+00	1.67E+01	4.32E-01	1.03E-01	9.61E-02	3.94E-02	3.16E-02
^{134}Cs	8.50E-05	-	1.23E-06	2.52E-05	-	3.36E-07	7.94E-07
^{137}Cs	8.15E+00	8.54E+02	9.21E-01	1.46E+00	2.93E+00	6.02E-02	3.02E-01
^{57}Co	-	1.73E-06	5.66E-12	1.47E-06	-	2.44E-12	5.15E-08
^{60}Co	2.03E+00	2.44E+01	3.38E-02	8.16E-02	2.13E-02	2.45E-03	2.18E-03
^{241}Am	8.17E-02	4.75E+01	7.35E-04	2.97E-02	1.28E-01	3.71E-03	7.69E-03
^{94}Nb	-	4.90E-01	-	5.98E-04	2.11E-03	-	4.47E-04
^{125}Sb	-	1.59E-02	-	4.78E-04	-	-	1.13E-05
^{152}Eu	2.43E+01	7.94E-01	6.82E-03	1.40E-02	-	1.85E-03	3.30E-03
^{154}Eu	1.08E+00	1.84E+00	8.99E-04	2.86E-03	1.05E-02	4.12E-04	5.31E-04
^{155}Eu	4.23E-02	1.39E-01	-	7.28E-04	1.17E-03	-	3.70E-05
^{55}Fe	2.32E+00	1.23E+02	2.55E-03	7.33E-03	2.45E-02	4.56E-04	2.75E-04
^{63}Ni	9.29E+00	1.40E+02	3.99E-01	1.04E+00	1.82E-01	2.68E-02	3.46E-02
^{90}Sr	1.22E-03	1.28E+03	8.03E-03	3.52E-02	3.50E+00	5.47E-02	8.75E-02
^{241}Pu	1.06E-03	1.52E+02	6.24E-03	8.98E-02	6.74E-01	2.00E-02	1.64E-02

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Second-ary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill
^{133}Ba	5.11E+00	2.66E-03	2.17E-03	-	-	-	8.12E-05
^{99}Tc	-	-	5.74E-03	7.16E-04	-	4.46E-07	2.18E-04
^{129}I	-	-	-	1.53E-03	1.03E-01	9.56E-07	3.89E-06
^{36}Cl	-	-	-	-	9.29E-02	-	-
^{233}U	-	-	7.70E-04	4.00E-04	4.62E-01	-	4.84E-05
^{234}U	-	1.56E+00	5.30E-04	8.50E-02	9.46E-03	8.55E-02	8.67E-03
$^{235}U^{\#}$	3.06E-01	6.58E-02	2.20E-03	8.90E-03	9.57E-03	6.13E-03	1.55E-03
$^{236}U^{\#}$	-	6.58E-02	-	1.48E-03	9.57E-03	-	1.37E-05
^{238}U	-	4.93E-01	5.24E-04	7.01E-02	6.67E-03	8.39E-02	1.49E-02
^{238}Pu	1.14E-04	1.68E+00	2.27E-04	8.33E-03	7.84E-03	1.74E-02	9.01E-04
^{239}Pu	2.29E-05	7.43E+01	1.72E-04	9.21E-03	1.24E-01	2.07E-02	3.41E-03
^{240}Pu	2.28E-05	6.05E+01	1.43E-04	7.50E-03	1.01E-01	1.70E-02	4.55E-03
^{242}Pu	-	-	5.98E-05	4.42E-04	8.29E-04	-	2.55E-05
^{242}Cm	-	-	4.99E-10	9.73E-11	7.20E-12	-	2.04E-11
^{243}Cm	-	2.74E-03	3.23E-06	1.38E-05	8.78E-05	1.75E-07	5.84E-07
^{244}Cm*	-	1.03E-01	1.28E-04	6.40E-04	3.33E-03	1.04E-03	2.15E-03
^{252}Cf*	-	-	9.04E-06	1.43E-05	4.57E-06	-	8.55E-07
^{226}Ra	-	-	1.45E-02	3.85E-02	-	2.05E-02	3.24E-03
^{40}K	-	-	-	1.51E-01	-	5.52E-02	1.36E-02
^{39}Ar	2.17E+00	-	-	-	-	-	-
^{41}Ca	5.37E+00	-	-	-	-	-	-
^{113m}Cd	1.64E-02	9.40E-01	-	-	-	-	-
^{151}Sm	4.01E+00	-	-	-	-	-	-
^{204}Tl	1.62E-02	9.31E-01	-	-	-	-	-
Total	**4.69E+02**	**2.95E+03**	**1.22E+01**	**1.65E+01**	**9.32E+00**	**1.76E+00**	**1.35E+00**

Table 2.47: SGHWR disposal inventory (MBq) summary for in-situ features and backfill presented at reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted red.

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Secondary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill	All
^{3}H	3.08E+05	4.43E+02	5.14E+04	5.60E+04	9.55E+02	2.35E+03	4.93E+04	4.69E+05
^{14}C	8.42E+02	4.61E+01	2.14E+03	4.35E+02	1.12E+02	7.45E+01	1.93E+03	5.59E+03
^{134}Cs	6.50E-02	-	6.08E-03	1.06E-01	-	6.35E-04	4.86E-02	2.26E-01
^{137}Cs	6.23E+03	2.35E+03	4.57E+03	6.14E+03	3.41E+03	1.14E+02	1.84E+04	4.13E+04
^{57}Co	-	4.76E-06	2.81E-08	6.17E-03	-	4.61E-09	3.15E-03	9.33E-03
^{60}Co	1.55E+03	6.70E+01	1.68E+02	3.44E+02	2.48E+01	4.64E+00	1.34E+02	2.29E+03
^{241}Am	6.25E+01	1.31E+02	3.65E+00	1.25E+02	1.49E+02	7.01E+00	4.70E+02	9.49E+02
^{94}Nb	-	1.35E+00	-	2.52E+00	2.46E+00	-	2.73E+01	3.36E+01
^{125}Sb	-	4.38E-02	-	2.01E+00	-	-	6.89E-01	2.74E+00
^{152}Eu	1.86E+04	2.18E+00	3.38E+01	5.88E+01	-	3.51E+00	2.02E+02	1.89E+04
^{154}Eu	8.24E+02	5.06E+00	4.46E+00	1.20E+01	1.22E+01	7.79E-01	3.25E+01	8.91E+02
^{155}Eu	3.24E+01	3.83E-01	-	3.06E+00	1.36E+00	-	2.26E+00	3.94E+01
^{55}Fe	1.77E+03	3.39E+02	1.26E+01	3.09E+01	2.85E+01	8.62E-01	1.68E+01	2.20E+03
^{63}Ni	7.11E+03	3.85E+02	1.98E+03	4.40E+03	2.12E+02	5.08E+01	2.11E+03	1.62E+04
^{90}Sr	9.34E-01	3.53E+03	3.98E+01	1.48E+02	4.08E+03	1.03E+02	5.35E+03	1.33E+04
^{241}Pu	8.11E-01	4.19E+02	3.10E+01	3.78E+02	7.85E+02	3.79E+01	1.00E+03	2.65E+03
^{133}Ba	3.91E+03	7.32E-03	1.08E+01	-	-	-	4.97E+00	3.92E+03
^{99}Tc	-	-	2.85E+01	3.01E+00	-	8.44E-04	1.34E+01	4.49E+01

Radio-nuclide	Bioshield (in-situ)	Mortuary Tubes (in-situ)	Primary (in-situ)	Secondary (in-situ)	Ponds (in-situ)	Ancillary Areas (in-situ)	Backfill	All
^{129}I	-	-	-	6.46E+00	1.20E+02	1.81E-03	2.38E-01	1.27E+02
^{36}Cl	-	-	-	-	1.08E+02	-	-	1.08E+02
^{233}U	-	-	3.82E+00	1.68E+00	5.38E+02	-	2.96E+00	5.46E+02
^{234}U	-	4.28E+00	2.63E+00	3.58E+02	1.10E+01	1.62E+02	5.31E+02	1.07E+03
$^{235}U^{\#}$	2.34E+02	1.81E-01	1.09E+01	3.75E+01	1.11E+01	1.16E+01	9.47E+01	4.00E+02
$^{236}U^{\#}$	-	1.81E-01	-	6.22E+00	1.11E+01	-	8.36E-01	1.84E+01
^{238}U	-	1.36E+00	2.60E+00	2.95E+02	7.77E+00	1.59E+02	9.13E+02	1.38E+03
^{238}Pu	8.72E-02	4.61E+00	1.13E+00	3.51E+01	9.13E+00	3.29E+01	5.51E+01	1.38E+02
^{239}Pu	1.75E-02	2.04E+02	8.53E-01	3.87E+01	1.45E+02	3.91E+01	2.09E+02	6.36E+02
^{240}Pu	1.75E-02	1.66E+02	7.08E-01	3.16E+01	1.18E+02	3.22E+01	2.78E+02	6.27E+02
^{242}Pu	-	-	2.96E-01	1.86E+00	9.66E-01	-	1.56E+00	4.68E+00
^{242}Cm	-	-	2.48E-06	4.09E-07	8.39E-09	-	1.25E-06	4.14E-06
^{243}Cm	-	7.53E-03	1.60E-02	5.79E-02	1.02E-01	3.31E-04	3.57E-02	2.20E-01
^{244}Cm*	-	2.82E-01	6.33E-01	2.69E+00	3.88E+00	1.98E+00	1.32E+02	1.41E+02
^{252}Cf*	-	-	4.49E-02	6.03E-02	5.32E-03	-	5.23E-02	1.63E-01
^{226}Ra	-	-	7.19E+01	1.62E+02	-	3.88E+01	1.98E+02	4.71E+02
^{40}K	-	-	-	6.35E+02	-	1.04E+02	8.31E+02	1.57E+03
^{39}Ar	1.66E+03	-	-	-	-	-	-	1.66E+03
^{41}Ca	4.11E+03	-	-	-	-	-	-	4.11E+03
^{113m}Cd	1.25E+01	2.58E+00	-	-	-	-	-	1.51E+01
^{151}Sm	3.07E+03	-	-	-	-	-	-	3.07E+03
^{204}Tl	1.24+01	2.56E+00	-	-	-	-	-	1.50E+01
Sub-total	**3.58E+05**	**8.11E+03**	**6.05E+04**	**6.97E+04**	**1.09E+04**	**3.33E+03**	**8.23E+04**	**5.93E+05**
^{3}H (bulk structure in-situ)								1.86E+04
Total								**6.12E+05**

276 The highest maximum activity is 8.02E+03 Bq/g for tritium in the bioshield, followed by 4.43E+03 Bq/g for ^{90}Sr in the mortuary tubes (Table **2.45**). The majority of the maximum activity concentrations are in the mortuary tubes (11 of 40) and secondary containment (11 of 40). The remaining maximum activities are in the backfill (7 of 40), bioshield (7 of 40), ponds (6 of 40) and primary containment (1 of 40). The backfill shares three joint maximum activity concentrations (two with the secondary containment and one with the primary containment) where an individual room in which a maximum overall activity is located is part disposed of in-situ and part disposed of as backfill.

277 The mortuary tubes have the majority (23 of 40) of the highest average activity concentrations; this is attributed to the fact that the mortuary tubes inventory comprises one activated component and also the conservative assumptions made in its derivation due to a lack of data. The remaining highest average concentrations are in the bioshield (8 of 40), ponds (4 of 40), secondary containment (3 of 40), and primary containment (2 of 40). The ancillary areas and backfill do not have any highest activity concentrations.

278 The total estimated radionuclide inventory for the SGHWR is 6.12E+05 MBq. The majority of the inventory, 5.30E+05 MBq, is disposed of in-situ; the estimated total activity of the backfill is 8.23E+04 MBq.

279 Figure **2.33** presents the overall activity contribution of each feature and percentage of the total disposal inventory. The bioshield contributes the highest proportion (58.6%) of the total radionuclide inventory. The bioshield dominates the inventory due to its high average activity concentrations, despite a fairly small overall mass. The next two largest contributions are from the primary containment (9.9%) and the secondary containment (11.4%), which both have moderate average activities (~10 Bq/g) and masses. The large masses of the backfill and bulk SGHWR structure result in fairly significant respective contributions (13.5% and 3.0%) to the disposal inventory despite their low average activity concentrations. Conversely, the high average activity of the mortuary tubes is offset by their very low overall mass, resulting in a small overall contribution to the inventory (1.3%). The remaining contributions are from the ponds (1.8%) and the ancillary areas (0.5%), which couple low masses with moderate to low average activities.

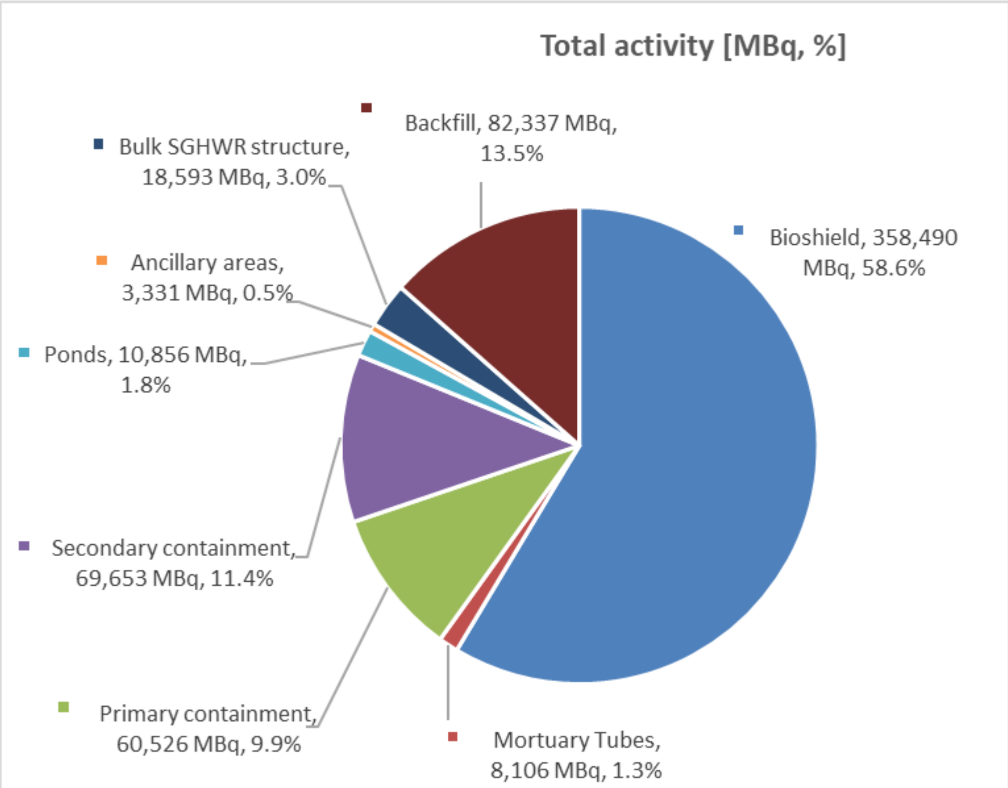


Figure 2.33: The radiological inventory contribution from each in-situ feature in the SGHWR and the backfill.

2.18.2 Inventory Location

280 In Figure **2.34** a plan view and cross-section of the SGHWR in-situ disposal structure is illustrated along with the inventory of the in-situ features (excluding elements to be included in the backfill and bulk tritium contamination which permeate the entire structure). The figure shows that the majority of the inventory is located in a relatively small volume of the structure.

2.18.3 Inventory Fingerprint

281 Figure **2.35** presents pie charts illustrating the main radionuclides in each SGHWR in-situ disposal feature and the backfill. The figure shows that ^{3}H dominates in the majority of features with the exceptions of the ponds and mortuary tubes. The similarity of the ponds and mortuary tubes is expected as the majority of the mortuary tubes assumed activity is from the ponds fingerprint. The ponds inventory also stands out due to the abundance of ^{90}Sr and actinides including U and Pu isotopes. In the bioshield a number of 'exotic' radionuclides predicted by neutron activation modelling are visible, including ^{133}Ba, ^{39}Ar and ^{151}Sm, as well as measured activation products such as ^{152}Eu and ^{60}Co. Outside of the bioshield, in the primary and secondary containment and ancillary areas, ^{137}Cs and ^{63}Ni are commonly-occurring radionuclides.

Figure 2.34: Plan (top) and cross-sectional (bottom) views of the SGHWR in-situ disposal inventory by component. Based on the Level 3 plan [53, Sheet 4] and [42]. The backfill and bulk SGHWR tritium contamination are not shown.

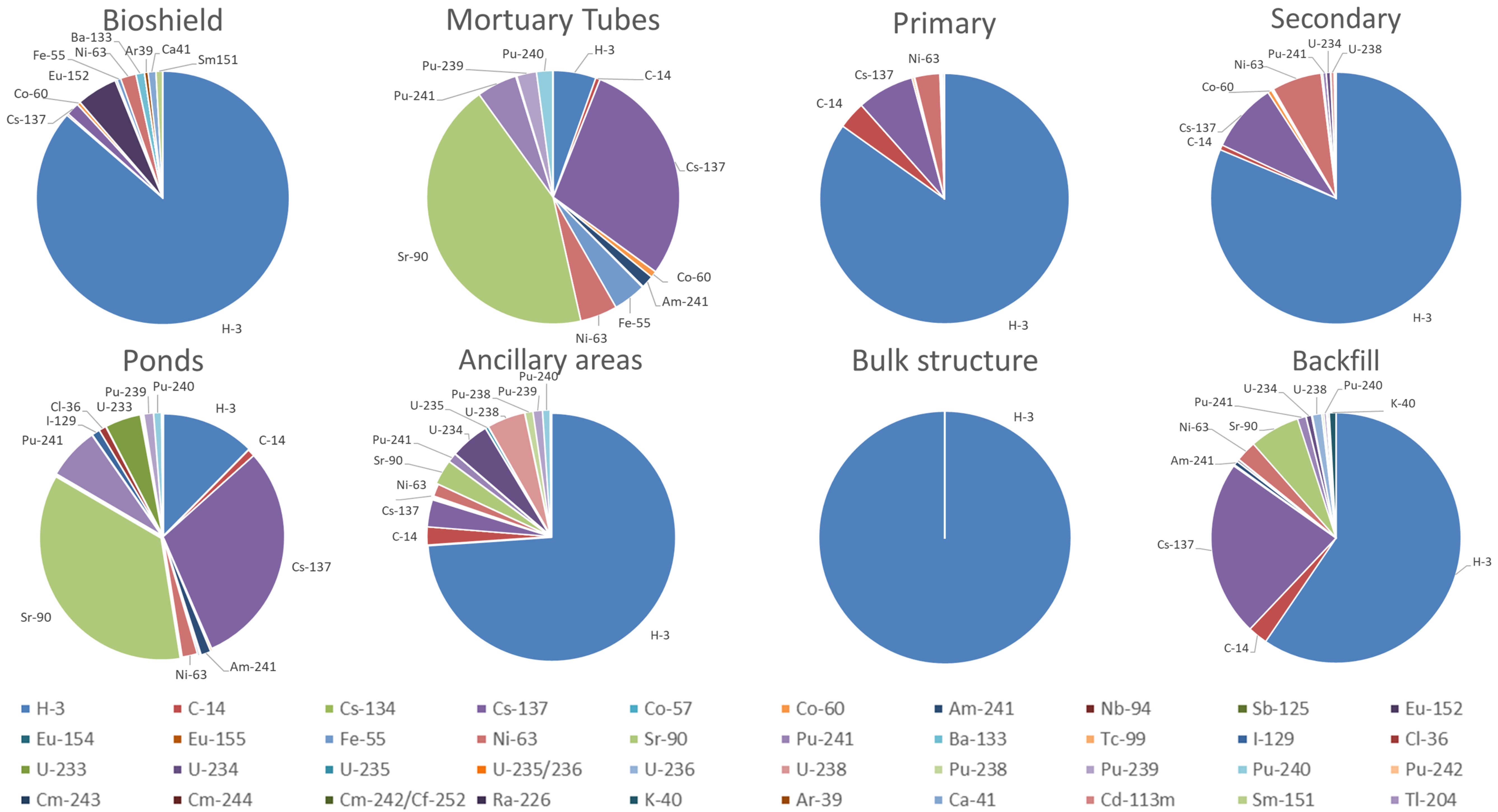


Figure 2.35: Radionuclide inventory fingerprint for SGHWR in-situ disposal features and backfill at 01/01/2027. Radionuclides contributing less than 0.3% are unlabelled. *(This page is set to print on A3.)*

2.18.4 Sensitivity Analysis Summary

282 Throughout this section, alternative inventories have been derived to explore the inventory sensitivity to uncertainties in the source data. Although the alternative inventories explore the impact of uncertainties, they are not considered to be realistic estimates. Table **2.48** records the activity difference between the reference inventory estimate and the alternative inventories calculated in these analyses for the different SGHWR components and features. The percentage of the total inventory and factor increase for the alternative inventories are also recorded. Table **2.49** presents a summary of the alternative inventory for each feature and the total SGHWR inventory under the alternative assumptions.

283 The largest increase in activity is seen in the bioshield, which was already the most active feature in the SGHWR and accounts for 88.3% of the activity in the alternative inventory rather than 58.6% in the reference inventory. The next largest contribution in the alternative inventory is the primary containment, which contributes 4.3%, overtaking the secondary containment due to the larger proportional increase in inventory. The smallest increase is seen in the ponds inventory, which is appropriate as these are the best-characterised of the SGHWR components.

284 The total activity of the alternative inventory is 5.91E+06 MBq which is a factor of 9.7 higher than the reference inventory (6.12E+05 MBq). The majority of the increase is driven by the alternative bioshield inventory. The component total activities increase by a factor of between 1 and 15 for the alternative inventory assumptions. The largest proportional increase is for ^{134}Cs, which increases by a factor of nearly 40 driven mostly by the alternative stored item activation fingerprint adopted for the alternative mortuary tubes inventory. Another consequence of the alternative stored item activation fingerprint is the addition of radionuclides ^{93m}Nb, ^{178n}Hf, ^{85}Kr, ^{59}Ni, ^{193}Pt, ^{121m}Sn and ^{93}Zr to the inventory.

Table 2.48: Comparison between the reference inventory estimate and the alternative inventory as explored in sensitivity analyses. Activity data are presented for a date of 01/01/2027.

Feature	**Reference Inventory Estimate**		**Alternative Inventory Estimate**			
	MBq	**%**	**Changes made**	**MBq**	**Increased by factor**	**%**
Bioshield	3.58E+05	58.6	Assume activity increases by uniform factor 14.6 to bring in line with activation modelling	5.22E+06	14.6	88.3
Mortuary Tubes	8.11E+03	1.3	More pessimistic inventory for stored item debris	2.56E+04	3.2	0.4
Primary Containment	6.05E+04	9.9	Adopt maximum rather than average measured activities for each component.	2.55E+05	4.2	4.3
Secondary Containment	6.97E+04	11.4		1.35E+05	1.9	2.3

Feature	Reference Inventory Estimate		Alternative Inventory Estimate			
	MBq	**%**	**Changes made**	**MBq**	**Increased by factor**	**%**
Ponds	1.09E+04	1.8	More pessimistic dimensional assumptions in derivation of contaminated volume.	2.01E+04	1.9	0.3
Ancillary Areas	3.33E+03	0.5	Adopt maximum rather than average measured activities for each component.	1.66E+04	5.0	0.3
Bulk Structure	1.86E+04	3.0	Assume average measured ancillary areas activity applies to all uncharacterised structure.	3.54E+04	1.9	0.6
Backfill	8.23E+04	13.5	Alternate inventories for contributing features, maximum activities for rubble mounds and extra 10% overall volume.	2.04E+05	2.5	3.4
Total SGHWR inventory	**6.12E+05**	**100**		**5.91E+06**	**9.7**	**100**

Table 2.49: SGHWR alternative disposal inventory (MBq) summary for each feature, presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted red.

Radio-nuclide	Bioshield	Mortuary Tubes	Primary	Second-ary	Ponds	Ancillary Areas	Bulk Structure	Backfill	Total
^{3}H	4.56E+06	4.40E+02	1.99E+05	8.00E+04	1.02E+03	1.06E+04	2.18E+04	1.26E+05	5.00E+06
^{14}C	1.24E+04	6.59E+02	7.28E+03	1.55E+03	1.24E+02	3.77E+02	7.85E+02	7.08E+03	3.02E+04
^{134}Cs	9.55E-01	7.47E+00	2.49E-02	2.81E-01	-	7.61E-04	2.81E-02	1.37E-01	8.90E+00
^{137}Cs	1.08E+04	2.35E+03	4.12E+04	3.91E+04	7.02E+03	7.76E+02	1.05E+03	4.52E+04	1.48E+05
^{57}Co	-	4.76E-06	1.97E-07	1.42E-02	-	4.94E-09	1.95E-07	7.28E-03	2.15E-02
^{60}Co	2.30E+04	1.50E+03	1.25E+03	1.06E+03	5.18E+01	3.38E+01	4.47E+01	7.18E+02	2.76E+04
^{241}Am	9.20E+02	1.31E+02	1.37E+01	2.38E+02	3.07E+02	3.34E+01	9.59E+01	5.57E+02	2.30E+03
^{94}Nb	-	3.22E+02	-	3.67E+00	3.70E+00	-	-	6.83E+01	3.98E+02
^{125}Sb	-	2.77E+01	-	3.21E+00	-	-	-	8.93E-01	3.18E+01
^{152}Eu	2.76E+05	2.18E+00	1.89E+02	9.43E+01	-	4.67E+00	1.52E+02	5.86E+02	2.77E+05
^{154}Eu	1.22E+04	5.06E+00	1.81E+01	1.84E+01	2.08E+01	9.34E-01	3.36E+01	9.06E+01	1.24E+04
^{155}Eu	4.81E+02	3.83E-01	-	5.02E+00	2.06E+00	-	-	4.85E+00	4.93E+02
^{55}Fe	2.64E+04	6.26E+01	2.55E+01	4.74E+01	3.14E+01	1.19E+00	1.00E+01	3.54E+01	2.66E+04
^{63}Ni	1.02E+05	1.19E+04	5.74E+03	8.38E+03	3.39E+02	1.68E+02	5.24E+02	4.47E+03	1.34E+05
^{90}Sr	9.34E-01	3.53E+03	8.95E+01	4.61E+02	8.48E+03	6.55E+02	8.20E+02	6.09E+03	2.01E+04
^{241}Pu	8.11E-01	4.19E+02	6.47E+01	5.00E+02	1.31E+03	1.06E+02	4.62E+02	1.41E+03	4.27E+03
^{133}Ba	5.80E+04	1.14E+02	9.28E+01	-	-	-	-	4.43E+01	5.83E+04

Radio-nuclide	Bioshield	Mortuary Tubes	Primary	Second-ary	Ponds	Ancillary Areas	Bulk Structure	Backfill	Total
^{99}Tc	-	-	3.52E+01	1.18E+01	-	1.15E-03	1.45E-01	1.68E+01	6.39E+01
^{129}I	-	-	-	2.52E+01	1.29E+02	2.45E-03	3.10E-01	7.41E-01	1.56E+02
^{36}Cl	-	-	-	-	1.17E+02	-	-	0.00E+00	1.17E+02
^{233}U	-	-	5.05E+00	1.84E+00	5.80E+02	-	-	3.54E+00	5.90E+02
^{234}U	-	4.28E+00	4.03E+00	6.35E+02	1.57E+01	9.82E+02	2.41E+03	2.48E+03	6.53E+03
$^{235}U^{\#}$	3.45E+03	1.81E-01	1.14E+02	6.62E+01	1.24E+01	7.52E+01	1.85E+02	3.22E+02	4.23E+03
$^{236}U^{\#}$	-	1.81E-01	-	1.34E+01	1.24E+01	-	4.72E-01	1.15E+00	2.76E+01
^{238}U	-	1.36E+00	3.52E+00	5.23E+02	9.58E+00	8.45E+02	2.27E+03	3.09E+03	6.74E+03
^{238}Pu	8.72E-02	7.05E+01	2.56E+00	6.54E+01	1.41E+01	3.17E+02	2.64E+02	2.38E+02	9.72E+02
^{239}Pu	1.75E-02	2.04E+02	3.35E+00	7.33E+01	2.96E+02	3.65E+02	3.18E+02	4.31E+02	1.69E+03
^{240}Pu	1.75E-02	1.66E+02	2.76E+00	5.97E+01	2.41E+02	2.97E+02	2.62E+02	4.63E+02	1.49E+03
^{242}Pu	-	-	5.07E-01	1.86E+00	1.38E+00	-	2.10E-01	1.83E+00	5.78E+00
^{242}Cm*	-	-	3.81E-06	4.20E-07	8.85E-09	-	5.62E-10	1.88E-06	6.12E-06
^{243}Cm	-	7.53E-03	1.53E-01	1.21E-01	1.18E-01	3.31E-04	1.68E-02	1.21E-01	5.38E-01
^{244}Cm	-	3.64E+01	6.05E+00	5.59E+00	4.48E+00	1.95E+01	1.58E+01	1.45E+02	2.33E+02
^{252}Cf*	-	-	6.90E-02	6.60E-02	5.61E-03	-	4.25E-03	6.67E-02	2.11E-01
^{226}Ra	-	-	4.89E+02	3.14E+02	-	3.39E+02	6.72E+02	9.02E+02	2.72E+03
^{40}K	-	-	-	1.24E+03	-	5.64E+02	3.21E+03	3.39E+03	8.39E+03
^{39}Ar	2.46E+04	-	-	-	-	-	-	-	2.46E+04
^{41}Ca	6.11E+04	-	-	-	-	-	-	-	6.11E+04
^{113m}Cd	1.86E+02	3.89E-01	-	-	-	-	-	-	1.86E+02
^{151}Sm	4.56E+04	-	-	-	-	-	-	-	4.56E+04
^{204}Tl	1.84E+02	2.97E+01	-	-	-	-	-	-	2.14E+02
^{93m}Nb	-	2.90E+03	-	-	-	-	-	-	2.90E+03
^{178n}Hf	-	6.84E+01	-	-	-	-	-	-	6.84E+01
^{85}Kr	-	1.89E+01	-	-	-	-	-	-	1.89E+01
^{59}Ni	-	6.55E+01	-	-	-	-	-	-	6.55E+01
^{193}Pt	-	7.35E+01	-	-	-	-	-	-	7.35E+01
^{121m}Sn	-	3.43E+02	-	-	-	-	-	-	3.43E+02
^{93}Zr	-	1.23E+02	-	-	-	-	-	-	1.23E+02
Total	**5.22E+06**	**2.56E+04**	**2.55E+05**	**1.35E+05**	**2.01E+04**	**1.66E+04**	**3.54E+04**	**2.04E+05**	**5.91E+06**

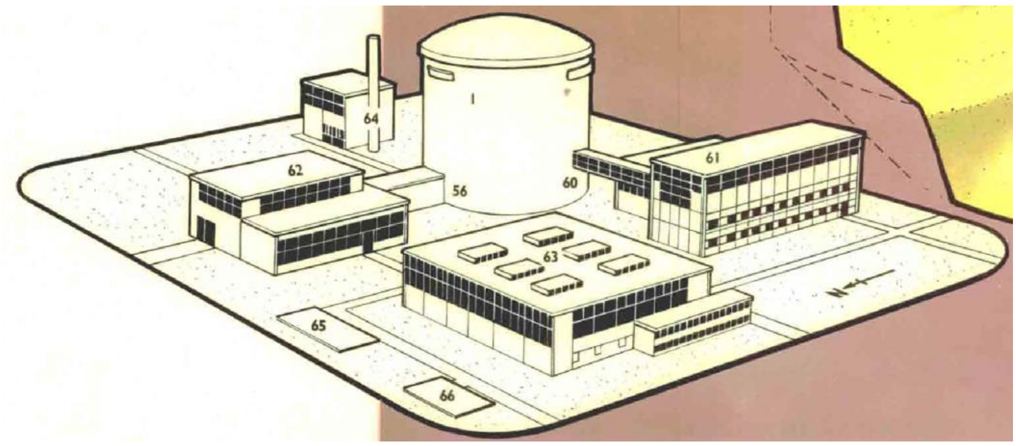
/ * ^{235}U and ^{236}U (^{242}Cm and ^{252}Cf) originally reported as a combined activity in some analyses. For the purpose of this inventory estimate, it is conservatively assumed that, in such analyses, the individual activity of both radionuclides is equal to the reported combined activity at the time of analysis.

3 Inventory Associated with the Dragon Reactor Complex

3.1 Background

285 The Dragon Reactor complex (B7) was constructed in the late 1950s and early 1960s [103, §3.2.2] and consists of a number of structures and ancillary plant on the western edge of the site, as observed in Figure **3.1**, Figure **3.2** and Figure **3.3**. The principal structure in the complex is the Dragon Reactor (building B70) itself, which is attached by a wide corridor to the fuel store building (B78) [104, §6]. The basement of the now demolished services building (B72) contains operational power cables between B70 and the control building (B71), which will be deplanted once use of the cables has ceased [104, §6].

Figure 3.1: Aerial views of the Dragon Complex in 1965 and 2012 (edited from [103, Photo.3 and cover page]).



Key:

1 = Outer concrete containment / B70 reactor building
56 = Vehicle airlock entrance
60 = Personnel walkway
61 = Control building (B71) / Western Offices
62 = Active fuel storage building (B78)
63 = Services building (removed to basement) B72
64 = Cooler building (B75) and stack
65 = Delay tanks (B76) (now removed)
66 = Fuel oil storage tanks (now removed)

Figure 3.2: Dragon Reactor Complex (edited from [105]).

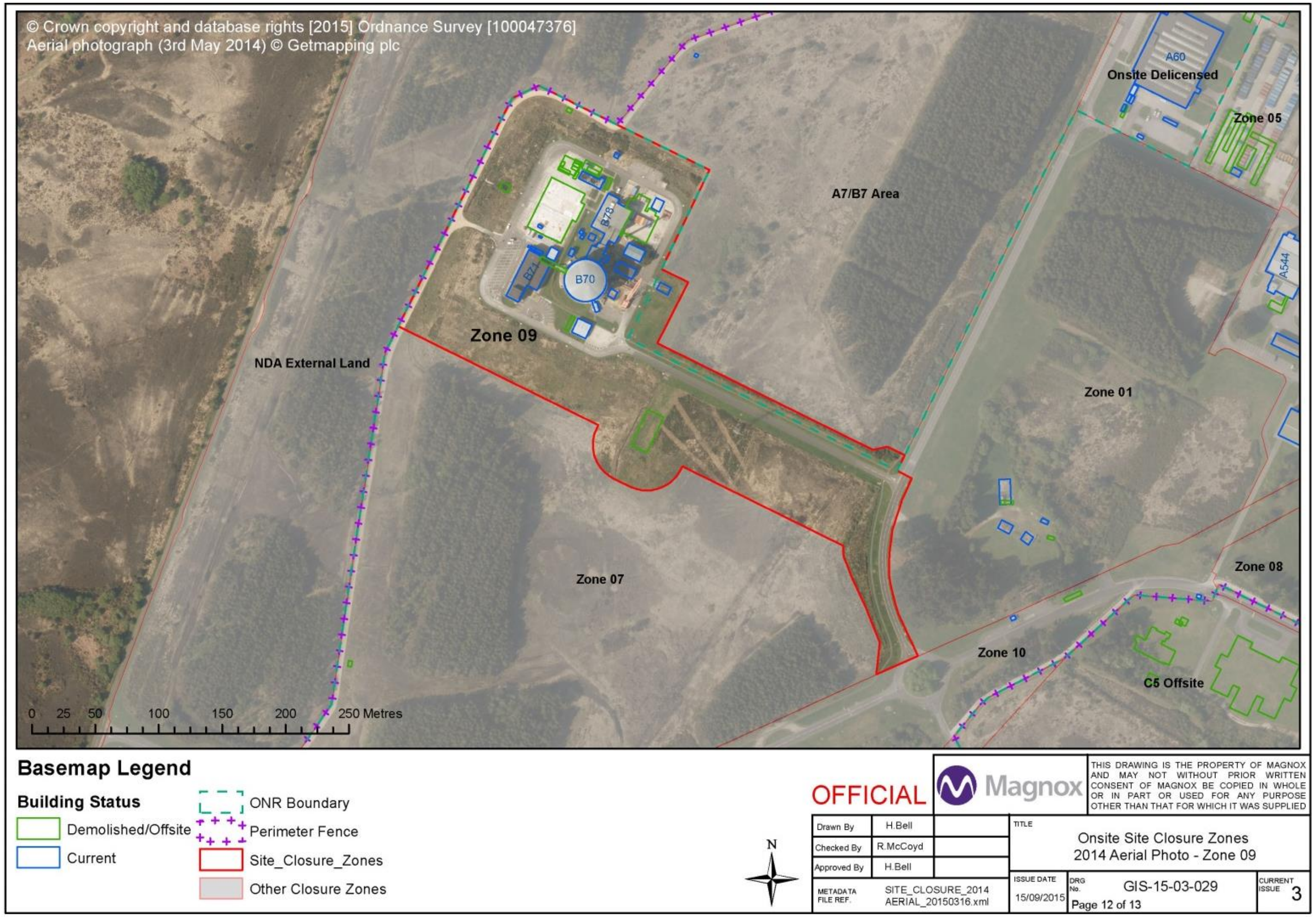


Figure 3.3: End State Zone 9 location Plan [103; Fig.1].

3.1.1 Dragon Reactor

286 The Dragon Reactor was a 20 MW high-temperature experimental reactor [106, §1]. It was built and managed as part of an Organisation of Economic Co-operation and Development (OECD) project to develop high temperature reactors (HTR) with helium coolant, and to develop graphite-coated uranium-thorium fuel cycle technology [106, §1]. The reactor contained 37 fuel element positions and it was operated with various fuel types and numerous core configurations throughout its operating life [106, §2.2]. Constructed between 1959 and 1962 [104, §6], the reactor first went critical on 23 August 1964 and operated for 10 years, until it was shut down in September 1975 [106, §2.2].

287 The B70 Dragon Reactor building is cylindrical in shape, 26 m high (above ground level) and 35.5 m in diameter, and the external structure comprises the outer containment for the facility [104, §6.1]. The Reactor basement extends 7.6 m below ground level with the top of the base slab at 27.2 mAOD, and 3.7 m of steel-reinforced concrete beneath that [104, §6.1; 107, §2.1]. Plant areas in the Dragon Reactor building are routinely referenced using heights above or below ground floor level, ranging from the reactor pit set into the base slab at -29’ (8.84 m) to +61.5’ (18.75 m).

288 The Dragon Reactor hall arrangement is illustrated in Figure **3.4**. The 2-ft-thick (0.61 m) outer concrete containment is labelled ‘Wall A’ in Figure **3.4**, with the outer annulus existing between Walls A and B. The inner steel containment shell sits inside Wall B; everything inside the steel shell is known as the inner containment. Wall C is located inside the steel shell, extending from the floor to the +18’ level where it supports a floor at the 18ft level in the inner containment and was the outer wall for several plant rooms. There also exist various below-ground structures outside of Wall A, including the service duct and ventilation, active and pipe ducts[15].

289 As presented in Figure **3.4**, the thermal shields (consisting of seven layers of inch-thick steel plates separated by running water in 1-inch gaps [108, §3.3]) surround the Reactor Pressure Vessel (RPV), which is then enclosed by the cylindrical 1.75-m-thick reinforced concrete bioshield (Wall D) [107, §2.1]. There was one penetration through the thermal shield and bioshield where the Purge Gas Pre-Cooler (PGPC) was located [108, §2], and penetrations for irradiation and viewing facilities just below the Main Shield Plug (not shown in Figure **3.4**).

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[15] Based on discussions with facility staff, it is assumed that these structures are not contaminated and that the voids associated with them will be filled with clean material. They are therefore not considered further in this report.

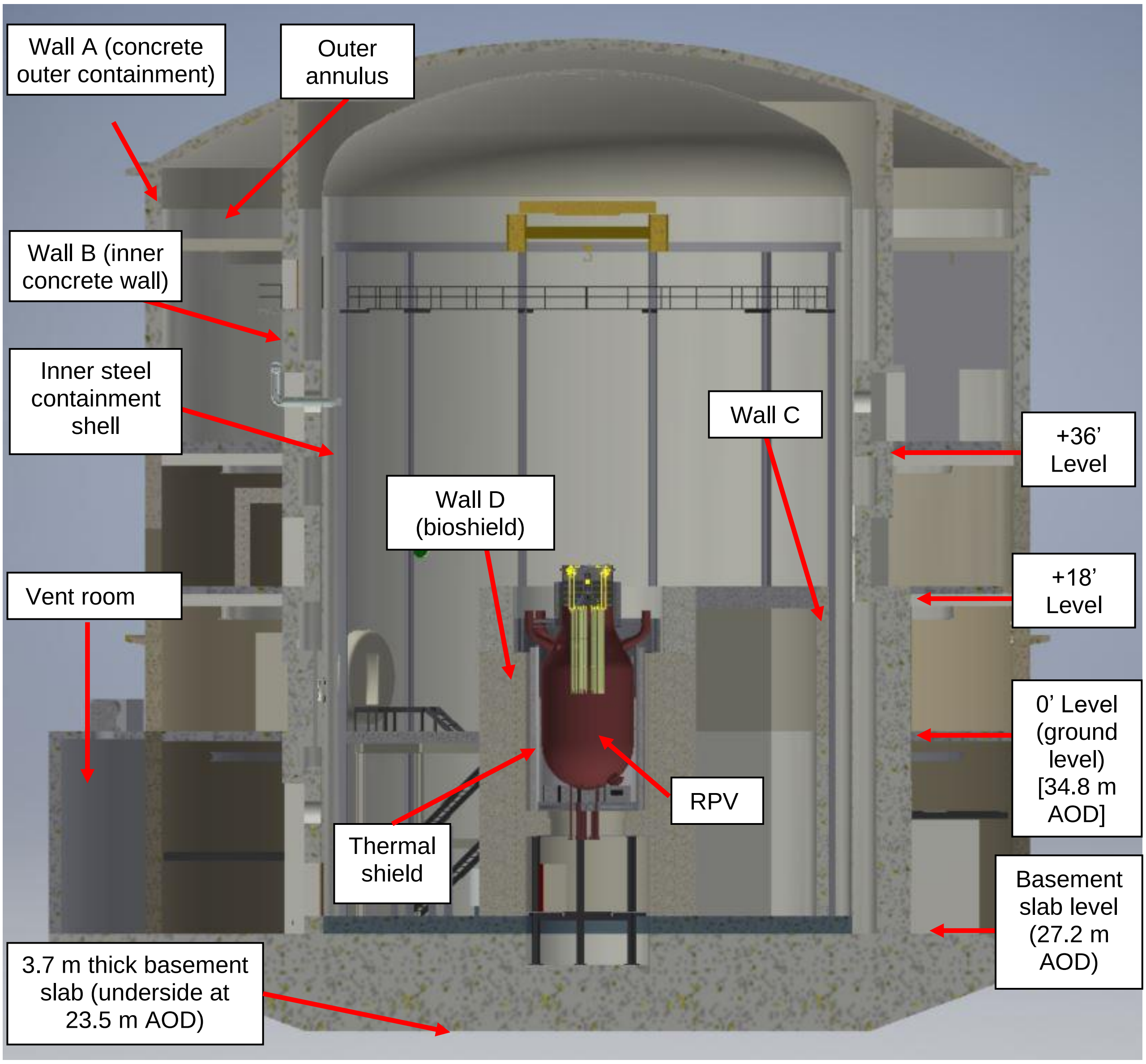


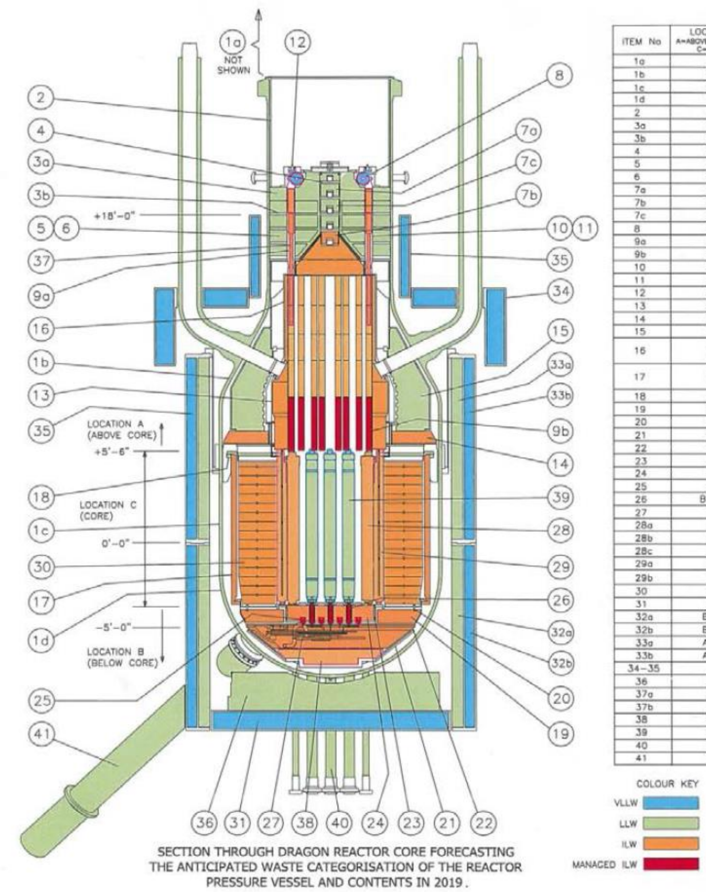
Figure 3.4: Split-view graphical model of the status in 2018 of the Dragon Reactor Building (edited from [108, Fig.1]).

290 In September 1975 the Dragon Reactor was shut down and the fuel was removed, packaged and transported to the Harwell site [106, §1.1]. All of the water systems were drained and dried. To maintain the integrity of the Dragon Reactor structure, dummy fuel assemblies were inserted and some components, such as the steel encapsulated boron carbide control rods, were left in place [106, §2.2.1].

291 Since reactor operations ceased, a significant amount of plant has been removed. The outer annulus between Walls A and B housed much of the ancillary equipment for the reactor and has now been almost fully de-planted [108, §2]. All auxiliary and primary circuit radioactive plant has been removed from the reactor facility [107, §2.3], including the Main Shield Plug and bricks above the core to the 18' level [109, §2.1]. Therefore, only the core of the reactor remains in the containment building. The next decommissioning phase is to build the plant that will be used to remotely dismantle the

reactor core and core internals; it is expected that the reactor will be fully removed over the next few years. This work includes removal of the steel thermal shield tanks outside the core using remote laser cutting, and removal of the bulk asbestos thermal lining between the shield tanks and the bioshield [109, §2.1].

292 Based on knowledge of operational history, available records and characterisation information, it is expected that the decommissioned reactor core components will range from intermediate to very low-level radioactive waste (ILW to VLLW) or OoS, as illustrated graphically in Figure **3.5** (although some areas require further investigation to enhance the available characterisation information) [107, §11]. The RPV and its content, steel thermal shields, accessible asbestos and accessible metal will be removed from the reactor hall and disposed of off-site. All that is proposed to remain at the end state are the below-ground portions of the reinforced concrete walls A-D and the below-ground section of the inner steel containment shell.



ITEM No	LOCATION A=ABOVE B=BELOW C=CORE	ITEM DESCRIPTION	WASTE CATEGORY	ASSOCIATED CAMPAIGN
1a	A	RPV ABOVE FLANGE	VLLW	N/A
1b	A	RPV BELOW FLANGE BUT ABOVE CORE	LLW	9
1c	C	RPV AT CORE HEIGHT	LLW	9
1d	B	RPV BELOW CORE	LLW	9
2	A	SUPPORT SKIRT (MSP)	LLW	1
3a	A	TOP PLATE (MSP)	LLW	1
3b	A	CENTRAL SPINE (MSP)	LLW	1
4	A	SHIELD DISCS (MSP) (4 OFF)	LLW	1
5	A	INSULATING LAYER (MSP) DIMPLED DISCS	LLW	1
6	A	GRAPHITE BLOCKS (MSP)	LLW	1
7a	A	REMOVABLE PLUGS IN MSP (5 OFF)	LLW	1
7b	A	REMOVABLE PLUGS IN MSP (5 OFF)	ILW	1
7c	A	REMOVABLE PLUGS IN MSP (5 OFF)	LLW	1
8	A	CONTROL ROD WINDING HEAD (24 OFF)	VLLW	1
9a	A	SHIELD (GUIDE) TUBE UPPER (24 OFF)	ILW	1
9b	C	SHIELD (GUIDE) TUBE LOWER (24 OFF)	MANAGED ILW	1
10	A	OUTER SLEEVE TUBE (24 OFF)	LLW	1
11	A	STEM LOCK (12 OFF)	ILW	1
12	A	PARK POSITIONS (6 OFF)	VLLW	1
13	A	STEELWORK OF PLENUM CHAMBER	LLW	3
14	A	PLENUM CHAMBER SUPPORT RING	ILW	3 & 5
15	A	UPPER NEUTRON SHIELD	LLW	5
16	A	NIMONIC LINING (UPPER CONE, MIDDLE CYLINDER & BOTTOM CYLINDER)	ILW	3
17	C	BEDPLATE SUPPORT SKIRT & SIDE NEUTRON SHIELD	ILW	6 & 7
18	A	SIDE NEUTRON SHIELD UPPER SEAL	LLW	6
19	B	STATIC REFLECTOR BEDPLATE	ILW	7
20	B	STATIC REFLECTOR LEVELLING PLATE	ILW	6
21	B	LOWER NEUTRON SHIELD	ILW	7
22	B	LOWER NEUTRON SHIELD GAS SEAL	ILW	7
23	B	CORE BEDPLATE RING	ILW	6
24	B	CORE BEDPLATE	ILW	6
25	C	SPIKE POCKETS (37 OFF)	MANAGED ILW	6
26	B, C	SPIKES (37 OFF)	MANAGED ILW	2
27	B	SPIKE PURGE LINES & CAPILLARIES	ILW	6
28a	C	INNER LIVE REFLECTORS, (30 OFF) HEAD & BASE	ILW	2
28b	C	INNER LIVE REFLECTORS, BODY	ILW	2
28c	C	INNER LIVE REFLECTORS, 'T' PIECE	ILW	2
29a	C	OUTER LIVE REFLECTORS, (30 OFF) TOP SHEATH	ILW	3
29b	C	OUTER LIVE REFLECTORS, BODY	ILW	3
30	C	STATIC REFLECTOR BLOCKS (960 OFF)	ILW	4
31	B	BOTTOM THERMAL SHIELD	VLLW	11
32a	B, C	LOWER SIDE THERMAL SHIELD (INNER)	ILW	11
32b	B, C	LOWER SIDE THERMAL SHIELD (OUTER)	LLW	11
33a	A, C	UPPER SIDE THERMAL SHIELD (INNER)	ILW	11
33b	A, C	UPPER SIDE THERMAL SHIELD (OUTER)	LLW	11
34–35	A	TOP THERMAL SHIELD	VLLW	11
36	B	THERMAL COLUMN	LLW	10
37a	C	CONTROL RODS (24 OFF) ST ST PART	ILW	1
37b	C	CONTROL RODS (24 OFF) BORON	ILW	1
38	B	LOWER STATIC REFLECTOR	ILW	7
39	C	DUMMY FUEL ELEMENTS (37 OFF)	LLW	2
40	B	RE-ENTRANT TUBE CONTENTS	LLW	6
41	B	PURGE GAS PRE-COOLER (PGPC)	LLW	8

Figure 3.5: Expected waste categorisation for Dragon Reactor core components to be decommissioned [107, §11], based on knowledge of operational history and characterisation information. Although the latter is limited in some areas, characterisation has typically focussed on areas of concern/expected contamination and is therefore expected to be bounding.

3.1.2 Dragon Fuel Store and Mortuary Hole Structure

293 The B78 Fuel Storage Building is connected to the Reactor Building B70 by a vehicle access-way and to the outer containment by a personnel access-way running alongside. The floor slab of the B78 building is contiguous with that of the B70 building vehicle airlock and there are steel rail tracks embedded in the floor slab running all the way from B78 to the reactor core. The B78 building includes a Fuel Storage Area (empty of fuel), a 30 t electric overhead travelling crane and a redundant active drain. The B78 facility also includes a number of offices on the west side together with personnel and workshop facilities. Vehicular access is provided through the north face roller shutter door.

294 The function of the B78 building was to provide a storage area for Dragon fuel elements during its operational life, although it was also used for the storage of other materials following defueling of the Dragon Reactor. The mortuary hole structure includes a 50-hole used fuel (“primary”) store and a 40-hole fresh fuel store [110, §1]. Fuel removed from the reactor was stored in the carousel beneath the fuel transfer tunnel in B70 before being transferred to the primary storage holes. Constructed in a concrete lined and filled pit roughly 5 m below ground level in B78, the primary mortuary hole system comprises vertical galvanised mild steel tubes [110, §2.1] with a wall thickness of 13 mm (as calculated from dimensions given in Table **3.38**).

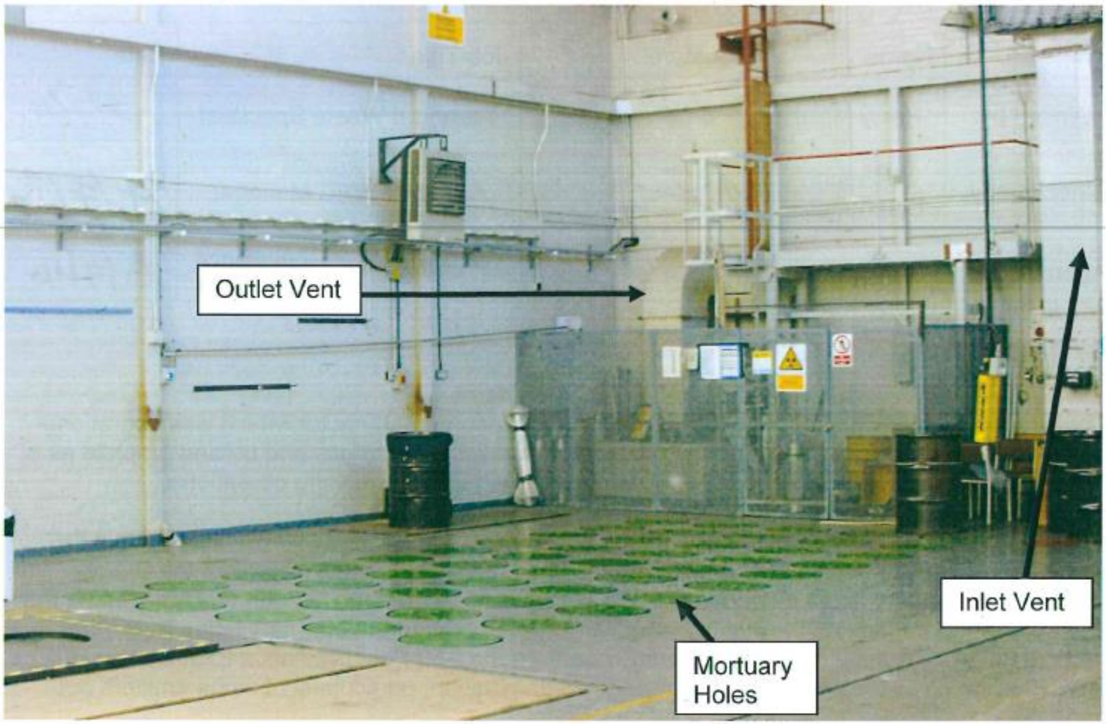


Figure 3.6: Primary mortuary holes and ventilation system (fenced compound now removed) [117, Fig.1].

3.2 Proposed End State

295 In-situ disposal of the Dragon Reactor bioshield and sub-surface concrete structures in B70, and the Dragon Fuel Mortuary Hole Structure in B78, has been identified as a credible option for the Winfrith site end state [104, §10] and subsequently as the

preferred option [111]. In-situ disposal of the B78 floor slab has also subsequently been identified as the preferred option [112].

296 Following final decommissioning and removal of plant and accessible metal, it is expected that all that will remain in the B70 Dragon Reactor building will be the outer annulus (reinforced concrete Walls A and B), the below-ground portion of the inner steel containment shell, Wall C (reinforced concrete) and the bioshield (reinforced concrete) [108, §2]. The steel thermal shield tanks and the RPV will be removed so that the only metal left in-situ will be the inaccessible inner steel containment shell[16] and steel reinforcement in the concrete walls, bioshield and base slab.

297 Although the details have not been finalised (INV-DRAGON-002), the concept design that will be applied to the disposal [111] involves decommissioning and removal of the Dragon Reactor core, demolishing the reactor building and associated structures to ground level [112][17] and then re-profiling the ground in this area and/or adding additional material to ensure at least 1 m of radiologically clean cover material is emplaced over the remaining below-ground structures. The concrete and masonry waste from demolition of the above-ground portion of the Dragon Reactor and Dragon Fuel Storage buildings will be used to backfill the below-ground voids left by demolition of the Dragon Reactor building; any remaining voids will be filled using material from the existing site stockpiles. An engineered cap will be emplaced over the below-ground Dragon structure to slow infiltration and leaching (the specific design is to be determined through detailed assessment) [104, §1.1]. A single cap is assumed to cover the Dragon Reactor building and the B78 floor slab (which includes the mortuary holes) [21, Fig.606/17].

298 All non-concrete and masonry items within B78 and the building structure itself will be removed; the demolished concrete and masonry will be used to backfill below-ground voids in the B70 building as noted above. The 40 new fuel storage holes are constructed in steel cassettes of five that can be removed relatively easily from their pit and so these will be extracted, leaving only the 50 primary mortuary holes [113]. The primary mortuary holes and remaining ventilation system (the lateral cuboid sections and curved structures – see Section 3.9) will be stabilised via infilling with clean grout to create a monolith structure [104, ES; 108, §2], and at least 1 m of radiologically clean cover material will be emplaced over the primary mortuary holes and floor slab, along with an engineered cap.

299 Illustrative cross-sectional and plan views of the proposed Dragon complex end state are provided in Figure **3.7** and Figure **3.8**.

300 Four inventory sets are therefore needed to model the Dragon Reactor complex: one for the below-ground B70 Dragon Reactor building structure potentially remaining in-situ;

[16] Removing the below-ground portion of the inner steel containment shell could undermine the integrity of the underground structure. Additionally, to remove it would also mean demolishing and removing parts of the concrete walls on either side of the steel shell (Walls B and C), which would not be proportionate given the small magnitude of the estimated associated inventory.

[17] Demolition to ground level is pragmatic because the top of the metal mortuary hole structure is at ground level [21] and the floor slab into which it is set spans both B78 and B70. The effort involved in removing the floor slab and demolishing to a lower level is considered to be disproportionate to any benefits.

one for material used as the B70 backfill; one for the below-ground Dragon Mortuary Hole Structure; and one for the B78 floor slab potentially remaining in-situ.

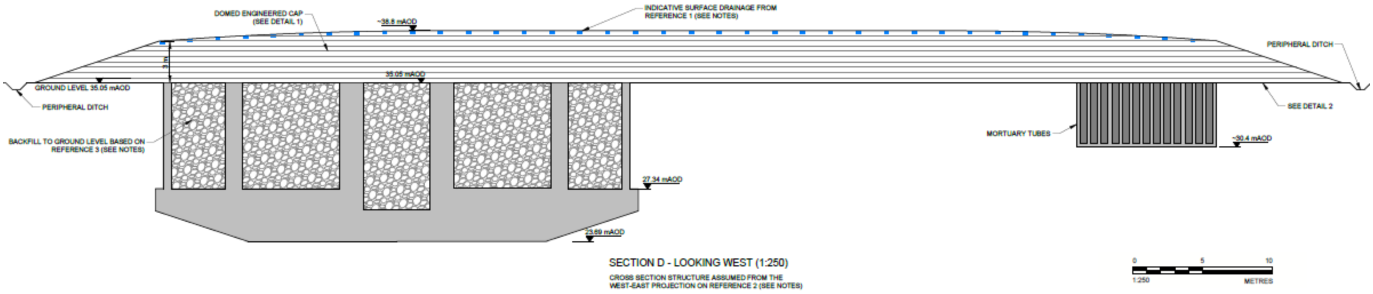


Figure 3.7: Illustrative cross-section of the in-situ below-ground Dragon Reactor building and Mortuary Hole Structure with a single cap covering both structures [114]. The floor slab connecting B78 and B70, which is now anticipated to be left in-situ, is not shown.

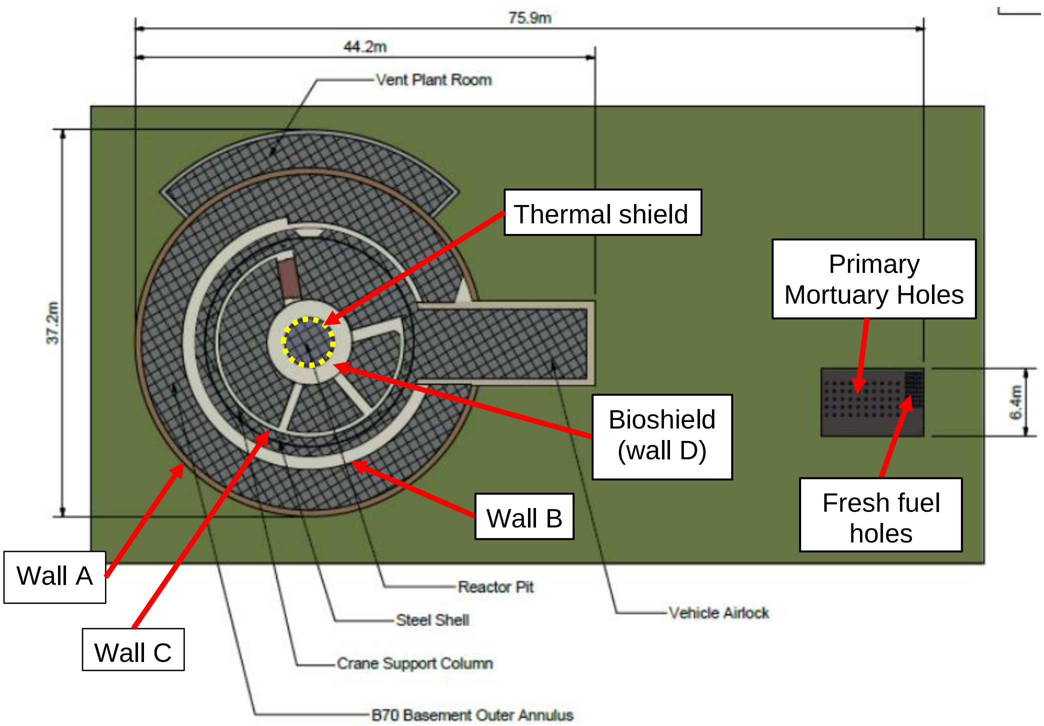


Figure 3.8: Plan of the key subterranean parts of the Dragon Reactor building (B70) and Mortuary Hole Structure [104, Fig.3]. The grey hatched area is below-ground void space to be infilled. It is currently assumed that the 40 fresh fuel hole structure will be removed. The floor slab connecting B78 and B70, which is now anticipated to be left in-situ, is not shown.

3.3 Sources of the Dragon Reactor Complex Radioactive Inventory

301 The majority of the radiological inventory present in the Dragon Reactor complex in-situ disposal at the end state is expected to be concentrated in the bioshield, which is mildly activated. However, due to the experimental nature of the reactor operation and the additional shielding provided by graphite blocks within the RPV, the bioshield

concrete is only lightly activated when compared with a Magnox power station reactor bioshield [106, §6.2.1; 104, §6.2.1].

302 The remaining inventory is associated with low-level contamination in the building paint, walls and floors of the B70 and B78 building structure. In B70, this contamination, much of which is already below OoS levels, derives from a number of sources:

- Operational activities during the lifetime of the facility – ^{137}Cs is a common contamination product [122, §3.1].
- Historically, ^{3}H dials were stored at the -25' level in the outer annulus, the leaking of which led to some contamination [109, §2.1.4].
- There is patchy contamination (^{3}H, ^{137}Cs and ^{60}Co) elsewhere in the facility from decommissioning, found primarily in the paint layer [109, §2.1.4]. The radionuclides ^{137}Cs and ^{60}Co are not believed to have penetrated into the concrete, but there is some evidence of ^{3}H migration into the concrete in higher activity areas [109, §2.1.4]. Some fission product contamination is also present.
- Historic decommissioning activities, and those to be undertaken, have the potential to redistribute some contamination within the facility since they involve remote drilling, sawing and laser cutting [109, §2.1]. The degree of contamination cannot be predicted, but it is assumed that this will be decontaminated as appropriate.
- During a lifting operation in the cathedral area on the 22 March 2021 to transfer the PGPC into a bespoke shielded container, contaminated water spilled onto the concrete floor [115]. Characterisation and clean-up of the spill is currently ongoing and it is expected to be largely decontaminated [116].

303 There is the potential for some low-level actinide contamination beneath the fuel carousel and fission product contamination in the steel-lined sump beneath the reactor [109, §2.1]; these areas will be characterised once they are accessible, but it is assumed here that they will be decontaminated as appropriate and so are not included in the end state inventory estimate (INV-DRAGON-001). These areas would be expected to have a different contamination fingerprint to that of the general building structure.

304 As well as containing the spent and fresh fuel stores, the B78 building has been used more generally for decommissioning activities and waste packaging prior to dispatch off site. General contamination in the B78 building is assumed to have a similar source to those listed above for B70 (with the exception of the ^{3}H dial storage and the PGPC spill in B70).

305 The primary mortuary holes were used to store spent fuel and are therefore expected to be contaminated with actinide and fission product radionuclides. Following defueling of the Dragon Reactor, the mortuary holes were also used to house various items from the on-site Post Irradiation Examination (PIE) facility (A59), which gives the potential for increased contamination, particularly of alpha emitters not normally associated with Dragon in significant quantities [108, §2; 117, §1]. The PIE facility examined a variety of fuel assemblies and their structural components, including fuel from other on-site reactors and also from off-site facilities [117, §2].

306 It is assumed that the remaining plant and structures comprising the Dragon reactor complex are either radiologically uncontaminated, OoS of RSR, or will be decontaminated prior to their demolition and removal from site (INV-DRAGON-001); there is no expectation that any other Dragon reactor complex below-ground concrete structures will be left in-situ at the end state. Similarly, no inventory associated with external areas of the Dragon reactor complex is captured in this report. It is assumed that any such contamination, if present, will be removed or is OoS (INV-DRAGON-001).

307 For the purposes of this assessment, the components associated with the disposal inventory are grouped into the following features:

- Dragon Reactor bioshield;
- Dragon Reactor (B70) building general surface contamination and additional ^3H ingress into the structure;
- Residual contamination from the Dragon Reactor building PGPC contaminated water spill;
- Dragon Fuel Storage (B78) building general surface contamination and additional ^3H ingress into the structure;
- Dragon Reactor structure backfill; and
- Primary Mortuary Hole Structure.

3.4 The Dragon Reactor Bioshield

3.4.1 Feature Description

308 The bioshield (Wall D in Figure **3.4**) is composed of reinforced concrete extending from the steel base plate; as of 2024 the bioshield had been removed down to the +18' level. The majority of the bioshield was shielded from significant activation by the thermal shields, but mild activation within the bioshield concrete and rebar has occurred [108, §3.3].

309 Higher levels of activation are expected in the region where the PGPC unit extended out from the reactor into the cathedral, penetrating the thermal shields and potentially creating a pathway for neutrons [108, §3.3]. The PGPC unit was attached to the lower section of the RPV and protruded into the cathedral room located at the -16' level (see Figure **3.5**); the PGPC was removed from its in-situ position in January 2018 (but remained within the Dragon Reactor building; see Section 3.6).

310 The bioshield, which forms a cylinder around the RPV and thermal shield tanks, is 5'9" (1.75 m) thick at its widest point [118; 107, §2.1] and then narrows slightly (with a larger inner diameter) towards the top of the reactor chamber. Table **3.1** summarises key dimensions and elevations associated with the bioshield.

Table 3.1: Dimensions of the Dragon Reactor bioshield. Data from [118] and extracted by NRS from the AutoCAD Inventor model file "Dragon Current State Model Issue 3".

Parameter	Value [m]
Wall thickness	1.75
Internal diameter	4.72
Total height (as of 2024)	12.57
Below-ground height	7.17
Above-ground height	5.4

3.4.2 Origin and Constraints on Radiological Inventory

311 The Dragon bioshield inventory is based on three main sources:

- Radiological characterisation data of six cores taken through the bioshield.
- Fingerprints derived for Dragon concrete blocks and the mild steel baseplate.
- Analogy with the neutron activation modelling of the concrete and rebar in the SGHWR bioshield, noting that the SGHWR activation data can only give a general indication of possible activation in the Dragon bioshield, and that direct comparison of total activities is not appropriate (see further discussion later in this section on page 162).

Characterisation Data

312 Current characterisation data for the bioshield comprises data from six cores taken in 2005, 2013 and 2017.

313 As part of a concrete coring campaign two cores were taken through the bioshield in 2005 at two different locations to determine whether the concrete and its associated rebar had been activated [119[18]; 120]. The core taken through the bioshield at the Ground Floor 0° radial was chosen to represent the intersection of the horizontal plane that goes through the reactor centre with the bioshield (i.e. the maximum neutron flux location), and the core taken at the +18' level was to determine whether the concrete had been activated at the furthest point from the maximum flux position [119, §2.2].

314 The 2005 cores were sub-sampled throughout their length to determine the activation profile for the concrete, resulting in 23 separate concrete samples. A single rebar sample was analysed from the Ground Floor core and two from the +18' core. The core taken from the maximum neutron flux location (Ground Floor 0°) showed slight activation of the concrete to a depth of 750 mm and also activation of the rebar situated at a depth of 100 mm from the surface of the bioshield [119, §1]. The activation products ^{14}C, ^{60}Co, ^{63}Ni and $^{152/154}$Eu were identified at very low levels with the dominant isotope being tritium (maximum 9.7 Bq/g). The +18' level core was also sub-

[18] Reference 119 is used in the absence of any other suitable source, but the validity of this reference is questionable – it is an undated draft project report, believed to have been developed around 2006/07.

sampled throughout its depth but showed very limited activation of the concrete and no activation of the rebar.

315 Three cores were taken in 2013 through the bioshield, asbestos layer and thermal shield metal layers, primarily to obtain material samples of the RPV and Side Thermal Shield Tanks at multiple elevations [121, §2.2]. The heights of the cores were set nominally at +6' -3", +1' -0' and -5' 0", all at 315^o radial, as indicated in Figure **3.9**, but the actual drill heights were slightly different.

316 Only samples from either end of the 2013 bioshield concrete cores were analysed, with results obtained for the samples from both ends of two cores and only the outer end of the third core. No rebar samples were obtained. Most results were at the LOD, but tritium was detected in three samples (maximum 0.56 Bq/g), and slight activation was demonstrated through measurement of ^{14}C in one sample (0.15 Bq/g), ^{60}Co in one sample (0.024 Bq/g), and ^{152}Eu in two samples (maximum 0.48 Bq/g). The fission product ^{137}Cs was also detected in four of the five samples (maximum 0.69 Bq/g), but this is assumed to be due to contamination on the external bioshield surfaces (INV-DRAGON-004).

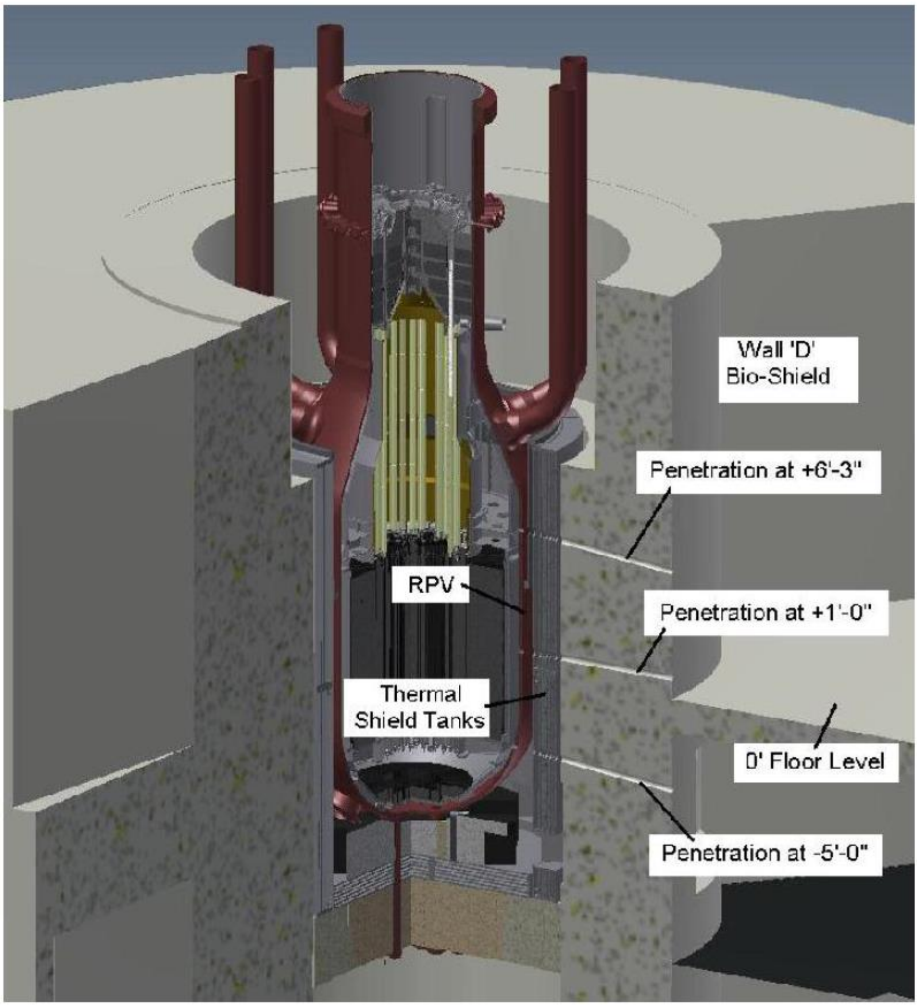


Figure 3.9: 2013 core nominal drilling locations through the bioshield [121, Fig.2].

317 In order to decommission the PGPC, a 2000 mm core was made in 2017 from the Helium Clean Up Plant (HCUP) room through the bioshield and thermal shields, to allow the cutting laser access to the upper section of the PGPC close to where it attached to the RPV [122, §1]. The approximate location of the core is shown in Figure **3.10**, which was taken at about the -5' level at ~85^{o} [122, §2]. Bioshield activation in this region is expected to be greater than elsewhere due to the neutron pathway offered by the PGPC.

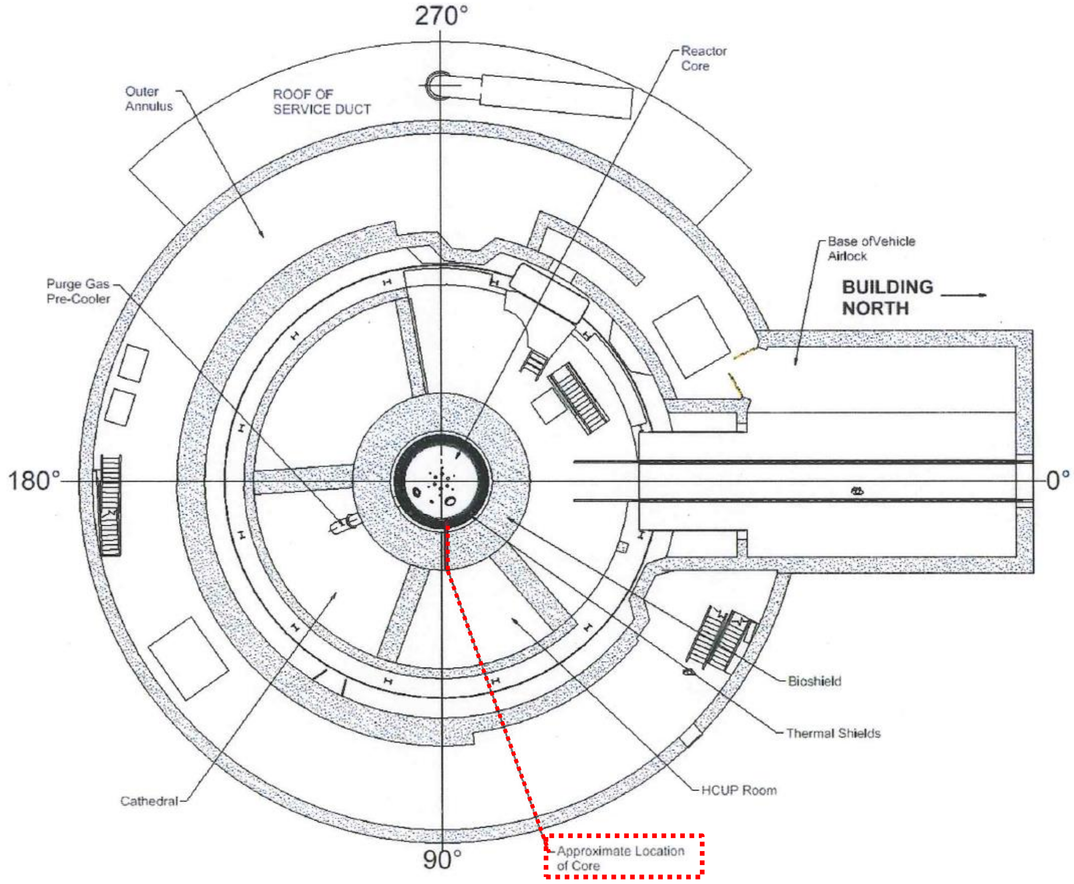


Figure 3.10: Drawing of the Dragon -5' level showing the approximate location of the core taken in 2017 (red dashed line) [122, Fig.1].

318 The stated core length is greater than the thickness of the bioshield (~2000 mm compared with 1750 mm) as the core was drilled at an angle through the bioshield. The bioshield core was "sliced", with sub-samples taken from some of the slices; starting from the outermost (HCUP) end, a 20 mm slice sub-sample was taken out of the start of each 120 mm length all the way along the core up until the final 300 mm closest to the reactor [122, §3.1.1]. The final 300 mm length was cut into 15 x 20 mm slices, each then forming the basis of a sub-sample [122, §3.1.1]. This resulted in a total of 29 concrete samples for analysis. All the concrete sub-samples were analysed by gamma spectrometry and the two samples closest to the reactor core were also analysed for activation products commonly found in concrete (^{3}H, ^{14}C, ^{41}Ca, ^{55}Fe and ^{63}Ni) [122, §3.1.1].

319 Of the rebar sections removed from the concrete, the piece of rebar with the highest activity was used to form a sample for analysis, with the sub-sample of this in the form of swarf [122, §3.1.4; 123, p.4]. The same determinands were analysed for as for the concrete samples, although analysis of ^{36}Cl was requested instead of ^{41}Ca. It was not recorded from where in the bioshield core the rebar sample was obtained [122, §2] (INV-DRAGON-004).

320 The sample analysis was reported in 2018 [123]. The gamma spectrometry results show that ^{152}Eu was present in all of the samples up to 550 mm from the inner bioshield surface (ranging 0.01-2.3 Bq/g), and ^{60}Co was also present in all but two of these samples (ranging 0.01-0.098 Bq/g). A few above LOD results for ^{63}Ni and ^{154}Eu were also observed. Tritium activities of 14-25 Bq/g were measured in the five analysed samples (also within 550 mm of the inner surface), but ^{41}Ca was beneath the LOD in the two samples considered. The fission product ^{137}Cs was measured above LOD in two samples (maximum 0.002 Bq/g), but this is assumed to be due to cross-contamination of the samples (INV-DRAGON-004). Only two results were above LOD for the rebar sample – ^{55}Fe (1.6 Bq/g) and ^{137}Cs (0.0012 Bq/g).

321 The raw sample analysis results for all six cores are reported in the accompanying Dragon inventory spreadsheet [15], decayed to a common analysis date of 05/04/2018. Those above LOD sample results for this characterisation data are provided in Table **3.2**.

Table 3.2: Dragon bioshield concrete and rebar sample analysis results for the six cores taken in 2005, 2013 and 2017 [119-123]. The rebar samples are shaded green. Sample distance from the inner bioshield surface conservatively corresponds to the innermost side of the sample. Blank cells indicate the determinand was not requested to be analysed; a dash indicates that the sample was analysed for, but reported at LOD. Radionuclides where every result was at LOD, and above LOD results for the naturally-occurring radionuclides (^{40}K, ^{208}Tl, ^{210}Pb, ^{212}Pb, ^{214}Pb, ^{226}Ra, ^{228}Ac, ^{232}Th, ^{234}Th, ^{234}U, ^{238}U), are reported in the accompanying Dragon inventory spreadsheet [15]. All results are decayed to the date of the most recent sample, 05/04/2018.

GAU ID	Magnox ID	Distance from Inner Surface	^{3}H	^{14}C	^{55}Fe	^{60}Co	^{63}Ni	^{137}Cs	^{152}Eu	^{154}Eu
		[mm]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]
2017 core at the -5 level										
GAU3875/40	DRA/SAMP/009A	1870				-		-	-	-
GAU3875/1	DRA/SAMP/9BB	1850				-		-	-	-
GAU3875/2	DRA/SAMP/009B	1730				-		-	-	-
GAU3875/3	DRA/SAMP/010	1610				-		-	-	-
GAU3875/4	DRA/SAMP/011	1490				-		-	-	-
GAU3875/5	DRA/SAMP/012	1370				-		-	-	-
GAU3875/6	DRA/SAMP/013	1150				-		-	-	-
GAU3875/7	DRA/SAMP/014	1130				-		-	-	-
GAU3875/8	DRA/SAMP/015	1010				-		-	-	-
GAU3875/9	DRA/SAMP/016	890				-		-	-	-
GAU3875/10	DRA/SAMP/017	770				-		-	-	-
GAU3875/11	DRA/SAMP/018	650				-		0.0009	-	-
GAU3875/12	DRA/SAMP/019	530				-		-	0.014	-
GAU3875/13	DRA/SAMP/020	410				-		-	0.051	-
GAU3875/14	DRA/SAMP/021	290				0.011		-	0.28	-
GAU3875/15	DRA/SAMP/022	270				0.014		-	0.3	-
GAU3875/16	DRA/SAMP/023	250				0.015		0.002	0.4	-
GAU3875/17	DRA/SAMP/024	230				0.019		-	0.4	-
GAU3875/18	DRA/SAMP/025	210				0.024		-	0.66	-
GAU3875/19	DRA/SAMP/026	190				0.035		-	0.82	-
GAU3875/20	DRA/SAMP/027	170	12			0.036		-	0.77	-
GAU3875/21	DRA/SAMP/028	150				0.055		-	1.05	-
GAU3875/22	DRA/SAMP/029	130				0.042		-	1.52	-
GAU3875/23	DRA/SAMP/030	110				0.053		-	2.3	0.07
GAU3875/24	DRA/SAMP/031	90	25			0.049		-	1.52	-
GAU3875/25	DRA/SAMP/032	70				0.063		-	1.48	0.049
GAU3875/26	DRA/SAMP/033	50	18			0.054		-	1.8	0.07
GAU3875/27	DRA/SAMP/034	30	19	-	-	0.098	-	-	2	0.048

GAU ID	Magnox ID	Distance from Inner Surface	^{3}H	^{14}C	^{55}Fe	^{60}Co	^{63}Ni	^{137}Cs	^{152}Eu	^{154}Eu
GAU3875/28	DRA/SAMP/035	0	14	-	-	0.074	0.24	-	1.9	-
GAU3875/39	DRA/SAMP/044			-	1.6	-	-	0.0012	-	-
2013 core nominally at +6'-3", 315o Radial										
GAU2995/9	C1k	1870	0.11	-		-		0.27	-	-
2013 core nominally at +1'-0", 315o Radial										
GAU2995/8	C2i	0	0.44	-		-		0.62	0.056	-
GAU2999/6	C2j	1870			0.34	-	-	-	-	
2013 core nominally at -5'-0", 315o Radial										
GAU2999/8	C3i	0			-	0.014	-	0.025	0.385	
GAU2995/10	C3j	1870	0.14	0.15		-		0.120	-	-
2005 core at ground floor, 0o Radial										
GAU777/1	WA/SAMPLE/1132	1700	3.01	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	1650	4.03	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	1600	4.95	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	1500	0.56	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	1250	2.40	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	1000	0.56	-	-	-	-	-	-	-
GAU777/1	WA/SAMPLE/1132	750	1.33			-		-	-	-
GAU777/1	WA/SAMPLE/1132	500	1.53	-	-	-	0.009	-	-	-
GAU777/1	WA/SAMPLE/1132	250	3.68	-	-	-	0.008	-	0.004	-
GAU777/1	WA/SAMPLE/1132	150	1.99	-	-	0.0019	0.011	-	0.029	-
GAU777/1	WA/SAMPLE/1132	100	4.29	-	-	0.0033	0.014	-	0.054	-
GAU777/1	WA/SAMPLE/1132	50	2.91	-	-	0.0067	0.023	-	0.076	0.003
GAU777/1	WA/SAMPLE/1132	0	2.71	0.07	-	0.0025	0.005	-	0.035	0.002
GAU777/1	WA/SAMPLE/1132	0	0.00	-	0.03	0.123	0.285	-	-	-
2005 core at +18' level										
GAU777/2	WA/SAMPLE/1134	815	4.70	-	-	-	-	-	-	-
GAU777/2	WA/SAMPLE/1134	765	0.17	-	-	-	-	-	-	-
GAU777/2	WA/SAMPLE/1134	715	1.07	-	-	-	0.009	-	-	-
GAU777/2	WA/SAMPLE/1134	615	0.51	-	-	-	-	-	-	-
GAU777/2	WA/SAMPLE/1134	415	0.71	-	-	-	-	-	-	-
GAU777/2	WA/SAMPLE/1134	250	1.28	-	-	-	-	-	-	-
GAU777/2	WA/SAMPLE/1134	150	0.77			-		-	-	-
GAU777/2	WA/SAMPLE/1134	100	2.76			-		-	-	-
GAU777/2	WA/SAMPLE/1134	50	4.70			-		-	-	-
GAU777/2	WA/SAMPLE/1134	0	3.68			-		-	-	-
GAU777/2	WA/SAMPLE/1134	0	0.36			-		-	-	-
GAU777/2	WA/SAMPLE/1134	0	0.14			-		-	-	-

Fingerprints

322 Two previously derived fingerprints have been considered in order to provide additional indicative values through scaling to the available characterisation data.

323 In 2013 a fingerprint was derived for concrete bricks from the Upper Support Ring (USR) above the RPV [124]. The results indicated the presence of ^{60}Co, $^{152/154}Eu$, ^{22}Na, ^{3}H and ^{41}Ca, demonstrating that sample activity was as a result of activation. Comparison of bulk concrete and surface smear sample results for ^{3}H showed that the ^{3}H activity was as a result of activation, not surface contamination.

324 In deriving the USR fingerprint, reference [124] excluded above-LOD measurements of ^{238}U, ^{232}Th and their daughters on the basis that they are naturally present in concrete, typically in the range of 30-40 Bq/kg [31]. Potassium-40 was also excluded on the basis that it is naturally-occurring in concrete (typically ~400 Bq/kg). Reference [124] excluded ^{137}Cs and ^{241}Am from the fingerprint derivation because the results indicated that they are present in insignificant quantities.

325 The 2013 USR concrete fingerprint, decayed and normalised to a common characterisation date of 5/04/2018, is provided in Table **3.3**. The radionuclide ^{63}Ni was not reported in the fingerprint, but this would be expected to be present in activated concrete (INV-DRAGON-004). The stated ^{41}Ca concentration is regarded here to be an indicative value as the ^{41}Ca measurement is likely to have been referenced against a similar low-energy beta emitter and not directly against a ^{41}Ca instrument standard, as certified standards are not readily available [125]. Similarly, ^{22}Na is commonly misidentified in gamma spectroscopy and, as a short-lived (2.6 y) product of fast neutron activation/proton bombardment of Al or Mg, is unlikely to be present in notable quantities (indeed, as discussed later in this section, ^{22}Na is excluded from the derived fingerprint for the concrete bioshield).

326 The second fingerprint considered is that derived for the mild steel baseplate above the RPV in 2014 [126], which was generated using six mild steel samples from comparable areas around the Dragon Reactor: two from the Primary Heat Exchanger thermal shield, two from the Main Shield Plug (MSP), one from the thermal shield and one from the Control Rod Drive. Due to the proximity of the baseplate to the reactor core, it is believed to be both activated and lightly contaminated.

327 The sample results showed activation products found in steel (^{3}H, ^{14}C, ^{63}Ni and ^{55}Fe) and the fission product ^{137}Cs [126, §2]. Alpha contamination (^{239}Pu, ^{240}Pu and ^{241}Am) from fuel contamination on one of the MSP samples was excluded from the fingerprint because this was not applicable for the steel baseplate. A number of results were inferred using the ratio to ^{60}Co; this was particularly needed for ^{36}Cl where the activity was inferred for five of the six samples (INV-DRAGON-004).

328 The 2014 mild steel baseplate fingerprint, decayed and normalised to a common characterisation date of 5/04/2018, is also provided in Table **3.3**. The ^{137}Cs content is as a result of surface contamination. The lack of the steel activation product ^{59}Ni and relatively high proportions of ^{14}C and ^{36}Cl in the fingerprint are noted (INV-DRAGON-004). It is possible that the sole ^{36}Cl above LOD result may be a false positive, although this would lead to a conservative over-estimate of the ^{36}Cl when using the baseplate fingerprint to scale measured samples (see later discussion).

Table 3.3: USR concrete brick and Dragon mild steel baseplate fingerprints [124; 126], decayed and normalised to the Dragon bioshield core common characterisation date of 05/04/2018 [15].

Radionuclide	USR concrete brick FP [%]	Mild steel baseplate FP [%]
^{3}H	82.11	4.26
^{14}C	0.71	2.68
^{22}Na	0.06	
^{36}Cl		0.65
^{41}Ca	1.61	
^{55}Fe		9.25
^{60}Co	0.92	43.87
^{63}Ni		36.59
^{137}Cs		2.70
^{152}Eu	14.02	
^{154}Eu	0.57	
Total	**100.00**	**100.00**

Neutron Activation Modelling for Primary Bioshield

329 Activation modelling of the Dragon bioshield has not been undertaken. Therefore, activation modelling results produced for the SGHWR have been used to inform derivation of the Dragon bioshield fingerprint. The SGHWR activation modelling study is discussed in Section 2.10.2.

330 The SGHWR is a different reactor type to that of Dragon, used different fuel and operated at different energies and over different periods. As noted previously, the type of concrete used in the activation model derives from a US Nuclear Regulatory Commission specification, so is not specific to Winfrith, and the rebar steel is a generic specification. The activation data are also calculated for 16 years after reactor shutdown (1990 to 2006), whereas it is now over 40 years since Dragon was shut down. Therefore, the activation data can only be considered to give a general indication of possible activation in the Dragon bioshield (INV-DRAGON-005). Given these considerations, only the average flux data [44, Tab.7-29 and Tab.7-31] have been considered (the activity calculated using the maximum flux uses the maximum flux as occurring in the 11th vertical compartment of the SGHWR activation model; given the different reactor design, the maximum flux calculated is not appropriate for comparison with Dragon).

331 In order to improve the correlation between the SGHWR activation modelling and the Dragon bioshield, the modelled activation activity using the average flux has been decayed – as Dragon has been shut down for 43 years from 1975 to the analysis date of the 2017 core (April 2018), rather than the modelled 16 years, the additional decay should be accounted for. Therefore, the modelled activity was decayed for a further 27 years (43 minus 16 years).

332 Over 200 isotopes were predicted to be formed by activation and subsequent decay in the concrete and rebar, although a number are stable and many have half-lives of less

than one year. The radionuclides modelled were reduced to a short-list of nine for concrete and 15 for rebar by requiring an activity contribution greater than 0.1% and a half-life greater than one year. Changing this requirement from 0.1% to 0.001% increases the number of radionuclides to be considered without significantly increasing the total calculated activity (the total activity increases from 99.8% and 9 radionuclides, to 99.9% and 21 radionuclides for concrete, and from 99.5% and 15 radionuclides to 99.8% and 33 radionuclides for rebar). In addition, the characterisation data show that the Dragon bioshield is lightly activated and the corresponding impact to the total Dragon activity would be very small. The resulting short-list of radionuclides along with the SGHWR average activation activity decayed for comparison with the Dragon bioshield is provided in Table **3.4** and presented graphically in Figure **3.11**.

333 As the activation modelling is for the SGHWR bioshield, it is not appropriate to compare the total activities. However, its value lies in considering the radionuclide activity proportions and ratios. For example, for the first modelled 6” of the SGHWR concrete bioshield, the ^{3}H:^{152}Eu ratio is 10.9 and the average ratio for the 2017 Dragon bioshield core samples is 11.78. The same comparison for the ^{154}Eu:^{152}Eu ratio is 0.034 and 0.032. This comparison is surprisingly close given the modelling assumptions and difference in the reactors and their operation. However, the comparison is less positive for the older core samples, with the average ^{3}H:^{152}Eu and ^{154}Eu:^{152}Eu ratios for the 2005 cores being 261.4 and 0.059, respectively (although this may be a reflection of changing sample analysis quality and ^{3}H variability and mobility).

334 In the following inventory estimate, the SGHWR activation analysis informs the development of the bioshield fingerprints but is not used directly. For the bioshield concrete fingerprint, it is only used to estimate the contribution of ^{154}Eu in one set of 2013 core samples, and for the ^{133}Ba and ^{151}Sm content. These three radionuclides only form 0.28%, 0.21% and 0.25%, respectively, of the resulting fingerprint, and it is not expected that use of Dragon activation analysis (if it were available) would lead to significant changes in the inventory. For the bioshield rebar fingerprint, activation analysis is explored as a possible basis for the fingerprint but is ultimately not used in its derivation (see discussion on page 179).

Table 3.4: Modelled SGHWR average bioshield concrete and rebar activation activity [44] decayed for an additional 27 years to represent the Dragon shutdown period, and then filtered to require greater than 0.1% contribution to total activity and a half-life greater than 1 year. The process followed is defined in the accompanying Dragon inventory spreadsheet [15]. The short-list also included ^{39}Ar and ^{85}Kr, but these were excluded on the basis that they will be generated as a gas and will not remain in the concrete.

Radionuclide	**Activity in Bioshield Concrete (Average Flux) [Bq/kg]**				**Activity in Bioshield Rebar (Average Flux) [Bq/kg]**			
Radial section	0-6”	6-12”	12-18”	18-38.28”	0-6”	6-12”	12-18”	18-38.28”
^{133}Ba	1.11E+05	4.36E+04	1.63E+04	2.31E+03				
^{14}C	1.64E+05	6.19E+04	2.38E+04	3.73E+03	9.01E+02	3.41E+02	1.32E+02	2.06E+01
^{41}Ca	3.34E+05	1.27E+05	4.90E+04	7.68E+03				
^{113m}Cd					4.56E+03	1.89E+03	6.98E+02	9.31E+01
^{60}Co					1.18E+04	4.65E+03	1.77E+03	2.61E+02
^{152}Eu	4.14E+06	1.70E+06	6.75E+05	1.06E+05	3.74E+03	1.53E+03	6.07E+02	9.51E+01
^{154}Eu	1.42E+05	5.98E+04	2.33E+04	3.33E+03				
^{55}Fe					1.76E+05	6.69E+04	2.59E+04	4.04E+03
^{3}H	4.53E+07	1.71E+07	6.66E+06	1.04E+06	7.31E+03	2.00E+03	6.46E+02	9.15E+01
^{93m}Nb								
^{59}Ni					4.47E+02	1.70E+02	6.57E+01	1.03E+01
^{63}Ni	2.70E+05	1.03E+05	3.96E+04	6.22E+03	3.96E+04	1.49E+04	5.76E+03	9.04E+02
^{193}Pt					3.95E+02	1.49E+02	5.16E+01	6.14E+00
^{151}Sm	1.34E+05	4.79E+04	1.75E+04	2.45E+03	1.47E+03	5.39E+02	1.97E+02	2.69E+01
^{121m}Sn					4.71E+02	1.76E+02	5.90E+01	7.25E+00
^{157}Tb					3.66E+02	1.69E+02	6.65E+01	8.68E+00
^{204}Tl					2.05E+03	8.21E+02	3.04E+02	4.23E+01

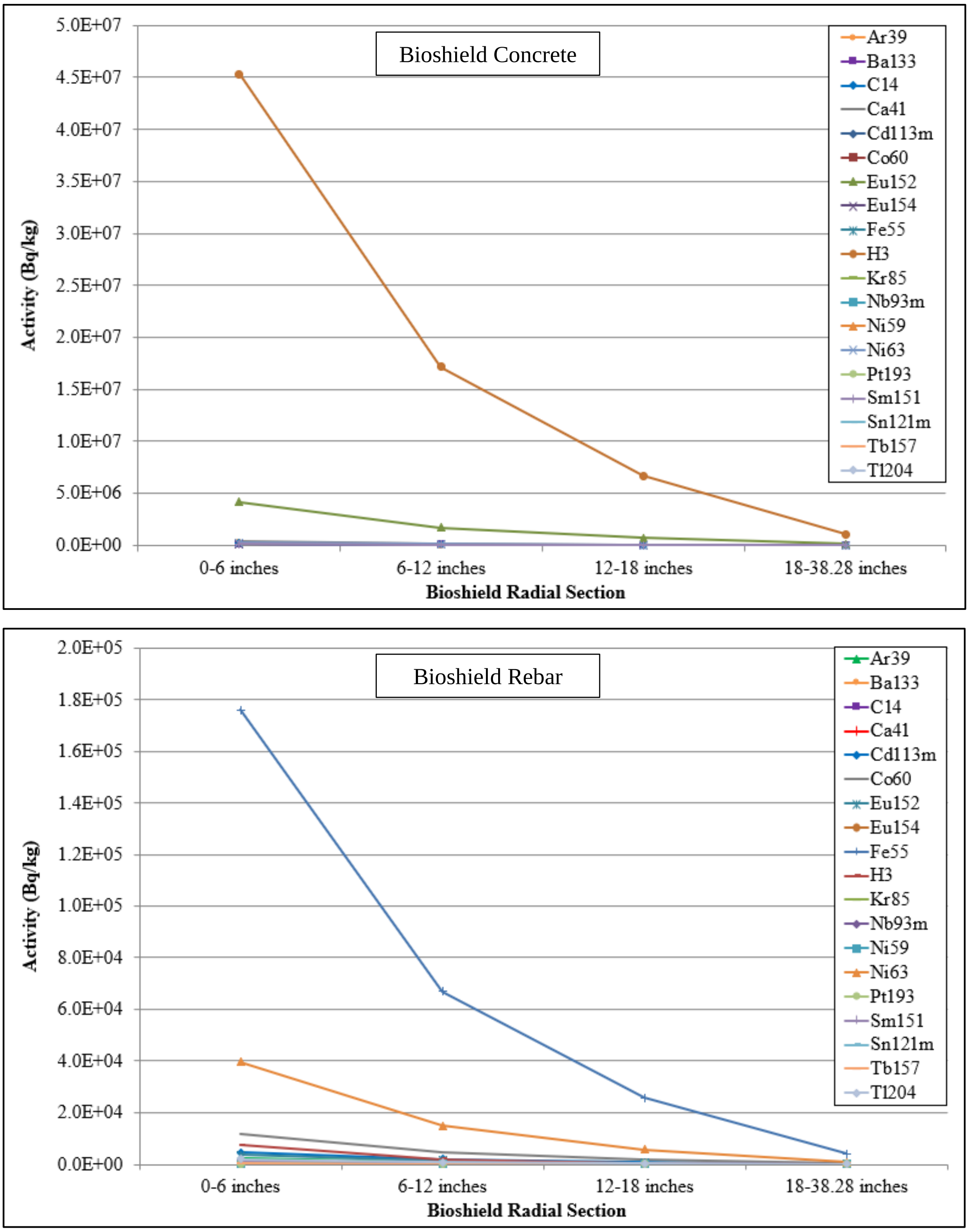


Figure 3.11: Modelled SGHWR average bioshield concrete (top) and rebar (bottom) activation activity [44] decayed for an additional 27 years to represent the Dragon shutdown period, and then filtered to require greater than 0.1% contribution to total activity and a half-life greater than one year [15].

Derivation of Bioshield Concrete Fingerprint

335 A fingerprint for the Dragon bioshield concrete has been developed from the characterisation data and using the specified fingerprints and indicative activation analysis to infer results where results for individual radionuclides were either not requested or were below the LOD. The approach taken is summarised by radionuclide as follows.

^{3}H

336 The measured above-LOD results for ^{3}H (decayed to 5/04/18) are as follows:

- 2017 core samples: 5 measurements made, all above LOD (12-25 Bq/g).
- 2013 core samples: 3 measurements made, all above LOD (0.1-0.4 Bq/g).
- 2005 core samples: 23 measurements made, all above LOD (0.2-5.0 Bq/g).

337 The measured ^{3}H activities are clearly not consistent between the datasets. The 2017 core is located in the area of the bioshield expected to have the least shielding and so subject to high neutron flux. In addition, the 2017 core samples are from the reactor-end of the core, whereas two of the 2013 samples are from the outer end of the core (the 0.4 Bq/g value is from the inner end of the core sample). The 2005 ground floor core is stated to have been taken from the maximum flux position and has higher results than the 2013 core samples, but lower than the 2017 core.

338 Consistent with [108, §3.3], an estimate for the ^{3}H activity in the other 2017 core samples has been calculated by assessing the ^{3}H:^{152}Eu ratio (^{152}Eu is used because it is also present as an activation product, it has a similar half-life, it was analysed for in every sample, and it is the radionuclide with the highest measured activity after ^{3}H, although it is noted that it does not behave chemically in the same way as ^{3}H). The average of the five calculated ratios is 11.8, but the maximum occurs for sample GAU3875/24 at a distance of 90-110 mm from the inner surface, indicating there was 16.5 times more ^{3}H activity than ^{152}Eu. This is illustrated graphically in Figure **3.12**.

339 Using the ratio of ^{3}H:^{152}Eu does assume that both radionuclides have a constant activity profile along the core and that the ratio between the profiles for both radionuclides is constant. As can be seen from Figure **3.12**, ^{3}H appears to peak approximately 100 mm in to the core before tailing off, whereas ^{152}Eu appears to gradually reduce throughout the core length (however, it is emphasised that there are very few data points on which to base this statement). Logarithmically (Figure **3.13**) the difference is not so significant. The difference in behaviour is also supported by Figure **3.11** for decayed activated SGHWR concrete, which shows similar behaviour to that in Figure **3.12**. Therefore, it is considered appropriate to infer the missing 2017 core ^{3}H results using the ^{3}H:^{152}Eu ratio and, conservatively, the maximum ^{3}H:^{152}Eu ratio value is used.

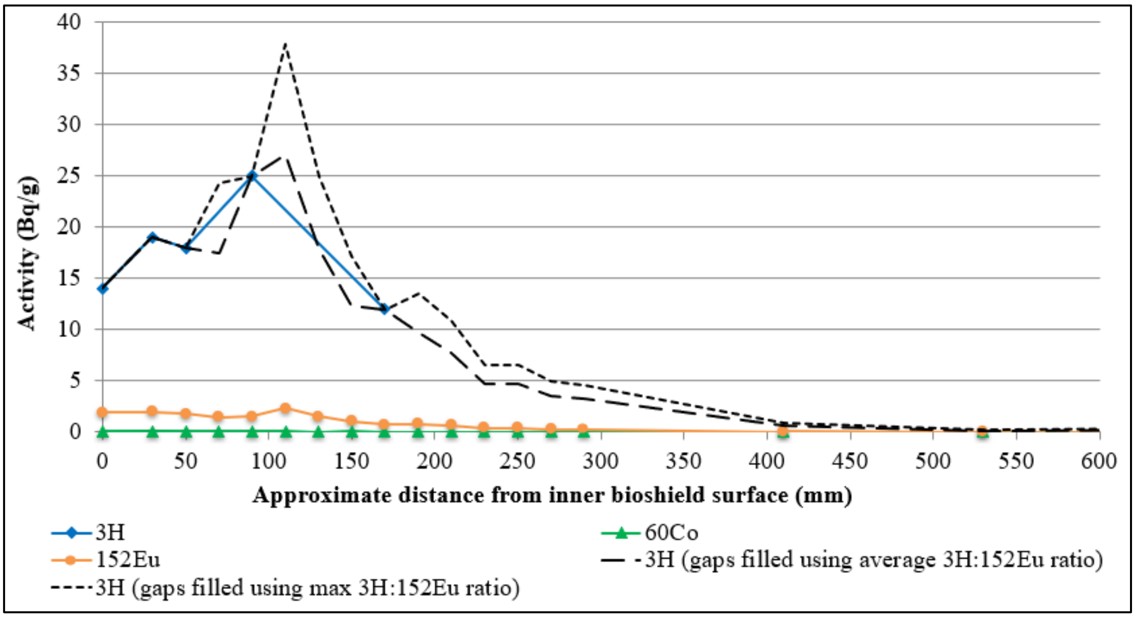


Figure 3.12: Radionuclide activity profile in the innermost 600 mm length of the 2017 bioshield concrete core. Measured results (coloured symbols) are from [123] and the two black dashed lines indicate where ^{3}H activities have been estimated using ^{3}H:^{152}Eu average and maximum ratios.

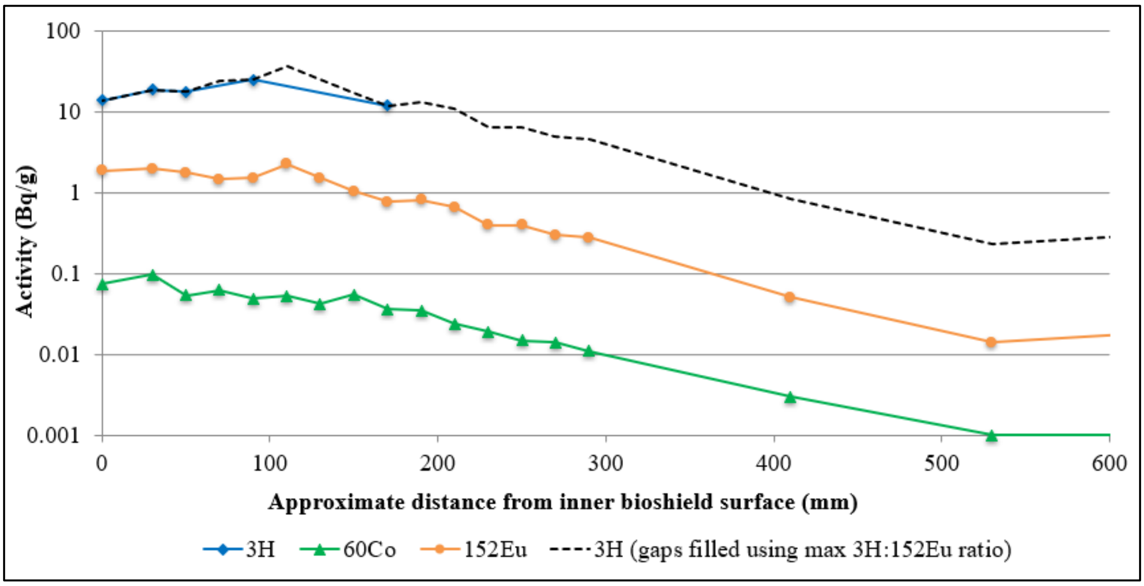


Figure 3.13: Repeat of Figure **3.12**, but activity is plotted logarithmically to compare the radionuclide activity profile in the core length.

340 The same approach is applied for the 2013 core data. For two of the three 2013 ^{3}H measurements the associated ^{152}Eu value was at the LOD, so the ^{3}H:^{152}Eu ratio of 8.0 is calculated for a single sample, GAU2995/8, where both the ^{3}H and ^{152}Eu values are above LOD and therefore this ratio is used to infer the ^{3}H content for the missing results.

341 There was no need to infer the ^{3}H content for any of the 2005 samples since all reported above LOD results.

^{14}C

342 Of the six cores, 23 samples were analysed for ^{14}C and only two were above the LOD – 0.15 Bq/g for the outer end of the 2013 -5' core and 0.07 Bq/g for the inner surface of the 2005 ground floor core. Given its long half-life (5,700 y) and that ^{14}C is expected to be present in activated concrete, the ^{14}C content was inferred by ratio.

343 Ratios using the two measured ^{14}C results were considered. For the 2013 sample, GAU2995/10, the associated ^{152}Eu result was at the LOD and the ratio to ^{3}H was quite high at 0.83. For the 2005 sample, GAU777/1, the ratios to ^{152}Eu and ^{3}H were 1.08 and 0.01, respectively – the former result suggesting a larger ^{14}C content than ^{152}Eu. Therefore, alternative methods were assessed for calculating the ratio. The radionuclide ratios in the concrete brick USR fingerprint [15; 124] were 0.051 for ^{14}C:^{152}Eu and 0.009 for ^{14}C:^{3}H at 5/04/2018; the same ratios in the decayed SGHWR activation analysis first 6” section [44; 15] are 0.040 and 0.004, respectively. As the concrete brick fingerprint ratios are slightly bounding of those suggested by the activation analysis, and they are based on material obtained from Dragon, the fingerprint ratios were used to infer the ^{14}C activity.

344 The ^{14}C activity is scaled from the ^{152}Eu values. Europium-152 is selected because: (1) it is the next most dominant radionuclide in the results after ^{3}H (which is mainly a calculated estimate for the 2013 and 2017 cores); (2) it is a less mobile and variable activation product than ^{3}H (and therefore likely to be a better correlation to the ^{14}C content); and (3) a ^{152}Eu value is available for the majority of samples. There are more above-LOD ^{3}H results for the 2005 core samples than ^{152}Eu, but it is considered more appropriate to scale to the less mobile radionuclide.

^{22}Na

345 None of the core samples were analysed for ^{22}Na. The concrete brick fingerprint [124] suggests that ^{22}Na could be present in the Dragon bioshield concrete, but the core results show that the bioshield is lightly activated so a substantial original inventory would not be expected. In addition, ^{22}Na has a half-life of only 2.6 y; the Dragon Reactor was shut down in September 1975 (45+ years ago) so over 17 half-lives have lapsed. On this basis, any ^{22}Na present in the bioshield is considered to be negligible and the radionuclide is excluded from the derived bioshield concrete fingerprint.

^{36}Cl

346 None of the concrete 2013 and 2017 core samples were analysed for ^{36}Cl. Of the 2005 core samples, 18 were analysed for ^{36}Cl and all were at the LOD (0.01-0.02 Bq/g). Review of the decayed SGHWR activation analysis [44; 15] indicates that the ^{36}Cl content in the concrete bioshield is expected to contribute less than 0.02%. On the basis of this expected small contribution, and the lack of any positive detections, ^{36}Cl is excluded from the Dragon bioshield concrete fingerprint.

^{41}Ca

347 Of the six cores, 23 measurements were made for ^{41}Ca but all were reported at the LOD (<1 Bq/g for the 2017 core, <0.1 Bq/g for the 2013 cores and <0.4 Bq/g for the 2005 cores). Given its long half-life (102,000 y) and that ^{41}Ca is expected to be present in activated concrete, the ^{41}Ca was inferred by ratio. The radionuclide ratios in the concrete brick USR fingerprint [15; 124] were 0.115 for ^{41}Ca:^{152}Eu and 0.020 for

^{41}Ca:^{3}H at 5/04/2018; the same ratios in the decayed SGHWR activation analysis first 6” section [44; 15] are 0.081 and 0.007, respectively. It was previously noted that the ^{41}Ca concentration in the concrete brick USR fingerprint should only be considered to be an indicative value, but as the fingerprint ratios are bounding of those suggested by the activation analysis, and they are based on material obtained from Dragon, the fingerprint ratios have been used to infer the ^{41}Ca activity. As for ^{14}C, the ^{41}Ca activities were scaled from ^{152}Eu.

^{55}Fe

348 Of the six cores, 22 concrete samples were analysed for ^{55}Fe – all were reported at the LOD (0.01-0.3 Bq/g), bar one of the 2013 core samples with 1 Bq/g at the outer end of the core. As every other result for this sample, GAU2999/6, was at the LOD, and the sample is for the outer end of the core, the positive result is likely to be from surface contamination rather than activation, and therefore not suitable for inclusion in the bioshield activated concrete fingerprint.

349 Iron-55 is a common activation product, but the results show that the bioshield is lightly activated and, with a half-life of just 2.74 y, more than 16 half-lives have lapsed since the Dragon Reactor was shut down in 1975. On this basis, any ^{55}Fe present in the bioshield would be expected to be negligible. This is supported by the decayed SGHWR activation analysis [44; 15], which indicates that ^{55}Fe is expected to contribute less than 0.003% to the total concrete bioshield activity, and by the fact that the activated concrete brick USR fingerprint [15] does not include ^{55}Fe. Therefore, ^{55}Fe is excluded from the Dragon bioshield concrete fingerprint.

^{60}Co

350 The measured above LOD results for ^{60}Co (decayed to 5/04/18) are as follows:

- 2017 core samples: All 29 samples were measured for ^{60}Co, with 15 above LOD (0.01-0.1 Bq/g).
- 2013 core samples: All 5 samples were analysed for ^{60}Co, with one above the LOD (0.014 Bq/g).
- 2005 core samples: All 23 samples were analysed for ^{60}Co, with four above the LOD (0.002-0.007 Bq/g).

351 As previously, the gaps have been inferred by scaling to ^{152}Eu. Conservatively, the maximum ^{60}Co:^{152}Eu ratio for the 2017 (0.052), 2013 (0.050) and 2005 (0.229) cores have been used (note that no above LOD ^{60}Co or ^{152}Eu measurements were made for the 2005 core at the +18’ level, so the maximum ratio for the 2005 core at ground level was used for both).

^{59}Ni

352 The core samples were not analysed for ^{59}Ni, a typical activation product. The radionuclide cannot be excluded from the fingerprint on the basis of its half-life, since it is long-lived (101,000 y). However, it is not included in the concrete brick USR fingerprint [15] and the decayed SGHWR activation analysis [44; 15] shows that ^{59}Ni is expected to contribute less than 0.007% to the total concrete bioshield activity. Therefore, this minor contributor to the Dragon bioshield concrete fingerprint is excluded.

^{63}Ni

353 The measured above LOD results for ^{63}Ni (decayed to 5/04/18) are as follows:

- 2017 core samples: The two inner-most samples were analysed, with one above the LOD (0.24 Bq/g).
- 2013 core samples: Two samples were analysed for ^{63}Ni, with both at the LOD.
- 2005 core samples: 18 samples were analysed, with seven above the LOD (0.005-0.023 Bq/g).

354 The maximum ^{63}Ni:^{60}Co ratio for each core has been used to infer the ^{63}Ni content as the chemical behaviour of ^{60}Co is more similar to that of ^{63}Ni than ^{152}Eu and there are almost as many above LOD values for ^{60}Co as there are for ^{152}Eu. As the activation product ^{63}Ni is not included in the concrete brick USR fingerprint [15; 124] no comparison could be made, but comparison to the decayed SGHWR activation analysis [44; 15] found that the sample-derived ratios are bounding and therefore should be used.

^{137}Cs

355 Caesium-137 is a typical fission product and as such would not be expected to be present in any notable quantity in the samples from within the bioshield, although ^{137}Cs surface contamination is present within the facility. On this basis, the Dragon Project Manager [108, §3.3] regarded the two above LOD values in the 2017 core samples (0.001 Bq/g for GAU3875/11 and 0.002 Bq/g for GAU3875/16) as accidental contamination during the extraction process and excluded them. Caesium-137 was also measured in four of the five 2013 core samples; however, these samples were from either end of the cores and therefore would also include surface contamination. All 23 samples from the 2005 cores reported ^{137}Cs at the LOD. This is further supported by the decayed SGHWR activation analysis [44; 15], which shows that ^{137}Cs as a result of activation is expected to contribute less than 0.00001% to the total activity. Therefore, ^{137}Cs is not included in the bioshield concrete activation fingerprint.

^{133}Ba and ^{151}Sm

356 The core samples were not analysed for the activation products ^{133}Ba and ^{151}Sm, and there is no estimate for their presence in the concrete brick USR fingerprint [15]. The decayed SGHWR activation analysis [44; 15] indicates that both radionuclides are expected to be present in activated concrete in small quantities, but contributing more than the 0.1% threshold used to filter the radionuclides. Therefore, ^{133}Ba and ^{151}Sm have been included in the Dragon bioshield concrete activation fingerprint in the proportions suggested by the activation analysis. As for the scaling of other radionuclides, the ^{133}Ba and ^{151}Sm activity has been scaled from the ^{152}Eu values.

^{152}Eu

357 All the samples were analysed for ^{152}Eu with 17 of the 2017 samples above the LOD (0.01-2.30 Bq/g), two of the 2013 samples above LOD (0.06 and 0.39 Bq/g), and five of the 2005 samples above LOD (0.004-0.076 Bq/g) (all stated at 5/04/2018).

^{154}Eu

358 The measured above LOD results for ^{154}Eu (decayed to 5/04/18) are as follows:

- 2017 core samples: All samples were analysed, with four above the LOD (0.05-0.07 Bq/g).

- 2013 core samples: Three samples were analysed, with all three at the LOD.
- 2005 core samples: All samples were analysed, with two above the LOD (0.002-0.003 Bq/g).

359 The decayed SGHWR activation analysis [44; 15] suggests a ^{154}Eu:^{152}Eu ratio of 0.0329. The average ratio using the 2017 core sample data is 0.032, and 0.059 for the 2005 core sample data. Therefore, the datasets are consistent (less so for 2005 but the same order of magnitude), and so the missing values for the 2013 core samples are inferred using the average activation analysis ratio to ^{152}Eu. The maximum ^{154}Eu:^{152}Eu ratios for the 2017 and 2013 core samples (0.039 and 0.062, respectively) have been used to infer the missing ^{154}Eu results in these datasets.

Naturally-occurring Radionuclides

360 The gamma spectroscopy analysis for all cores observed the presence of naturally-occurring radionuclides, including ^{40}K, and those from the ^{238}U and ^{232}Th decay chains. These radionuclides are excluded from the derived bioshield concrete fingerprint on the basis that they are naturally occurring and consistent with typical values quoted for concrete [31], and because they would not be expected to be produced via activation:

- The maximum ^{40}K values of 170 Bq/kg for the 2017 core and 100 Bq/kg for the 2013 core are less than the typical natural concentration of 400 Bq/kg [31, Tab.1].
- Uranium series: The average activity for ^{234}U and ^{238}U is 0.005 Bq/g for both, implying they are in secular equilibrium and therefore from the same natural source.
- Thorium series: ^{232}Th is not reported in the 2017 and 2013 core data, but the reported daughter radionuclides in this chain are all short-lived and have activities less than, or consistent with, the typical ^{232}Th content for concrete (30 Bq/kg [31, Tab.1]). The ^{232}Th results from 2005 support this also.

Activation Profile

361 The 2017 core sample results show no artificial activity in the outermost section of the bioshield (more than 650 mm from the inner bioshield surface) with low LOD values given. For the 2005 ground floor core, ^{63}Ni is consistently recorded in all samples out to about 750 mm, and ^{152}Eu to 500 mm.

362 There are two 2013 outermost sample results above LOD, for ^{14}C and for ^{3}H. For ^{14}C the value is high (0.15 Bq/g), particularly when compared with the other two ^{14}C LOD results of 0.02 Bq/g and the average ^{14}C activity inferred for the first 650 mm of 0.04 Bq/g. Given the lack of above LOD values for the other radionuclides at the outer end of the 2013 bioshield cores (other than ^{3}H discussed below), it is assumed that this result arises due to surface contamination or error.

363 Two 2013 core sample results for the outer end of the core measured very low levels of ^{3}H; given the decreasing trend in the 2017 sample results towards the outer surface of the bioshield, it is assumed that these two results derive from surface contamination rather than activation. However, ^{3}H above the LOD is measured in every core section sample for both 2005 cores. Nonetheless, given the high mobility of ^{3}H, the difficulty in detection, and the fact that the other key activation products (^{60}Co, ^{63}Ni and ^{152}Eu) show a decreasing trend from the inner bioshield surface outwards but the ^{3}H appears

random, all suggest that the results have arisen due to contamination rather than actual activation throughout the entire length of the bioshield cores.

364 Given the above, it is assumed here that the outer section of the bioshield (conservatively assumed to be greater than 750 mm) is not activated and that the derived Dragon bioshield concrete activation fingerprint applies only to the first 750 mm from the inner bioshield surface.

Derived Fingerprint

365 The resulting dataset of measured and inferred Dragon bioshield concrete samples is presented in Table **3.5**. As per the above discussion, other than some ^{3}H values and one ^{14}C sample result, all samples that are at least 750 mm from the inner bioshield surface are reported at the LOD or are calculated from a LOD result (indicated by the red font). Therefore, the fingerprint for the bioshield concrete is calculated from the average of the above-LOD values for samples that are less than 750 mm from the bioshield inner surface (at the bottom of Table **3.5** and in Table **3.6**).

366 Table **3.5** also reports the average radionuclide concentration when considering only the above-LOD values for 2017 core samples that are less than 750 mm from the bioshield inner surface. The 2017 core is believed to be located in the area of the bioshield with a potential neutron pathway along the PGPC (meaning there is potentially less shielding and therefore greater activation), while the 2005 ground floor core was stated to be in the area of highest neutron flux. The measured activities from the 2017 core are generally higher (by ~x10), which suggests that the PGPC provided an effective neutron pathway that compensated for being in a position of lower flux than for the 2005 ground floor core. Including the 2005 and 2013 core data in the derivation reduces the average total activity from 15.6 Bq/g to 9.5 Bq/g, although this is dominated by the ^{3}H estimate – if ^{3}H is excluded then there is minimal difference in the total average activity (1.5 Bq/g for 2017 core data alone compared with 1.1 Bq/g for all six cores). The maximum activity is the same in both datasets since it is dominated by the 2017 sample results. It would be conservative to derive the concrete bioshield fingerprint just using the 2017 data, but this is not considered appropriate for a fingerprint that is to be applied over the whole inner bioshield concrete volume. Given that the activation will be lower towards the top and bottom of the bioshield, and that most of the bioshield has more shielding than in the area of the PGPC, it is a more realistic representation to derive the fingerprint using the average of all the core samples. However, further characterisation to confirm that there are no areas of higher activation in the bioshield (particularly in the regions where barytes concrete was used; see later discussion) would support this approach (INV-DRAGON-004).

Table 3.5: Dragon bioshield concrete sample analysis results for the six cores taken in 2005 (GAU3875 samples), 2013 (GAU2995 and GAU2999 samples) and 2017 (GAU777 samples) [119-123], with results inferred as discussed above if results for individual radionuclides were either not requested or were below the LOD. Red font indicates the result was reported as LOD or is calculated from an LOD value, blue shading indicates a calculated/inferred result and green shading indicates a directly measured value that has been decayed to the common date, and no shading with black font indicates an above LOD value reported at 05/04/2018 (i.e. no calculation has been applied). The full analysis is recorded in the accompanying Dragon inventory spreadsheet [15]. The average and maximum values exclude LOD results and are for samples located <750 mm from the inner bioshield surface (indicated by the red dashed line). Presented at the date of the most recent sample, 05/04/2018.

Sample ID	Distance from Inner Surface	^{3}H	^{14}C	^{41}Ca	^{60}Co	^{63}Ni	^{133}Ba	^{151}Sm	^{152}Eu	^{154}Eu
[-]	[mm]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]	[Bq/g]
GAU2995/9	1870	0.11	0.002	0.005	0.003	0.037	0.001	0.001	0.040	0.014
GAU2999/6	1870	0.50	0.003	0.007	0.003	0.078	0.001	0.002	0.064	0.002
GAU2995/10	1870	0.14	0.150	0.005	0.002	0.030	0.001	0.001	0.040	0.014
GAU3875/40	1870	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.007
GAU3875/1	1850	0.49	0.002	0.003	0.002	0.006	0.001	0.001	0.03	0.009
GAU3875/2	1730	0.49	0.002	0.003	0.002	0.006	0.001	0.001	0.03	0.007
GAU777/1	1700	3.01	0.0003	0.0006	0.0005	0.008	0.0001	0.0002	0.005	0.002
GAU777/1	1650	4.03	0.0005	0.0012	0.0010	0.008	0.0002	0.0004	0.011	0.003
GAU3875/3	1610	0.49	0.002	0.003	0.003	0.010	0.001	0.001	0.03	0.008
GAU777/1	1600	4.95	0.0005	0.0012	0.0010	0.006	0.0002	0.0004	0.011	0.002
GAU777/1	1500	0.56	0.0005	0.0012	0.0010	0.007	0.0002	0.0004	0.011	0.003
GAU3875/4	1490	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.005
GAU3875/5	1370	0.33	0.001	0.002	0.001	0.003	0.0004	0.0005	0.02	0.005
GAU777/1	1250	2.40	0.0003	0.0006	0.0005	0.008	0.0001	0.0002	0.005	0.002
GAU3875/6	1150	0.16	0.001	0.001	0.001	0.003	0.0002	0.0002	0.01	0.003
GAU3875/7	1130	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.006
GAU3875/8	1010	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.008
GAU777/1	1000	0.56	0.0003	0.0006	0.0005	0.007	0.0001	0.0002	0.005	0.002
GAU3875/9	890	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.007
GAU777/2	815	4.70	0.0005	0.0012	0.0010	0.009	0.0002	0.0004	0.011	0.004
GAU3875/10	770	0.33	0.001	0.002	0.002	0.006	0.0004	0.0005	0.02	0.005
GAU777/2	765	0.17	0.0005	0.0012	0.0010	0.009	0.0002	0.0004	0.011	0.003
GAU777/1	750	1.33	0.0005	0.0012	0.0010	0.004	0.0002	0.0004	0.011	0.003
GAU777/2	715	1.07	0.0003	0.0006	0.0005	0.009	0.0001	0.0002	0.005	0.002
GAU3875/11	650	0.33	0.001	0.002	0.001	0.003	0.0004	0.0005	0.02	0.005
GAU777/2	615	0.51	0.0003	0.0006	0.0005	0.009	0.0001	0.0002	0.005	0.002
GAU3875/12	530	0.23	0.001	0.002	0.001	0.003	0.0003	0.0003	0.014	0.001
GAU777/1	500	1.53	0.0003	0.0006	0.0005	0.009	0.0001	0.0002	0.005	0.002

Sample ID	Distance from Inner Surface	^{3}H	^{14}C	^{41}Ca	^{60}Co	^{63}Ni	^{133}Ba	^{151}Sm	^{152}Eu	^{154}Eu
[-]	**[mm]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**	**[Bq/g]**
GAU777/2	415	0.71	0.0005	0.0012	0.0010	0.009	0.0002	0.0005	0.011	0.003
GAU3875/13	410	0.84	0.003	0.006	0.003	0.010	0.001	0.001	0.051	0.002
GAU3875/14	290	4.61	0.014	0.032	0.011	0.036	0.007	0.008	0.28	0.011
GAU3875/15	270	4.93	0.015	0.034	0.014	0.045	0.008	0.008	0.3	0.012
GAU777/1	250	3.68	0.0002	0.0004	0.0003	0.008	0.0001	0.0002	0.004	0.0002
GAU777/2	250	1.28	0.0005	0.0012	0.0010	0.009	0.0002	0.0005	0.011	0.003
GAU3875/16	250	6.58	0.020	0.046	0.015	0.049	0.010	0.011	0.4	0.016
GAU3875/17	230	6.58	0.020	0.046	0.019	0.062	0.010	0.011	0.4	0.016
GAU3875/18	210	10.86	0.033	0.076	0.024	0.078	0.017	0.019	0.66	0.026
GAU3875/19	190	13.49	0.042	0.094	0.035	0.114	0.021	0.023	0.82	0.032
GAU3875/20	170	12	0.039	0.088	0.036	0.117	0.020	0.022	0.77	0.030
GAU777/1	150	1.99	0.001	0.003	0.0019	0.011	0.001	0.002	0.029	0.001
GAU777/2	150	0.77	0.0005	0.0012	0.0010	0.004	0.0002	0.0006	0.011	0.003
GAU3875/21	150	17.27	0.053	0.120	0.055	0.178	0.028	0.034	1.05	0.041
GAU3875/22	130	25.00	0.077	0.174	0.042	0.136	0.041	0.049	1.52	0.059
GAU3875/23	110	37.83	0.117	0.263	0.053	0.172	0.062	0.074	2.3	0.07
GAU777/1	100	4.29	0.003	0.006	0.0033	0.014	0.001	0.003	0.054	0.002
GAU777/2	100	2.76	0.0005	0.0012	0.0010	0.004	0.0002	0.0006	0.011	0.003
GAU3875/24	90	25	0.077	0.174	0.049	0.159	0.041	0.049	1.52	0.059
GAU3875/25	70	24.34	0.075	0.169	0.063	0.204	0.040	0.048	1.48	0.049
GAU777/1	50	2.91	0.004	0.009	0.0067	0.023	0.002	0.004	0.076	0.003
GAU777/2	50	4.70	0.0005	0.0012	0.0010	0.006	0.0002	0.0006	0.011	0.004
GAU3875/26	50	18	0.091	0.206	0.054	0.175	0.048	0.058	1.8	0.07
GAU3875/27	30	19	0.101	0.229	0.098	0.318	0.053	0.065	2	0.048
GAU3875/28	0	14	0.096	0.218	0.074	0.24	0.051	0.061	1.9	0.074
GAU2995/8	0	0.44	0.003	0.006	0.002	0.037	0.001	0.002	0.056	0.002
GAU2999/8	0	3.01	0.019	0.044	0.014	0.049	0.010	0.015	0.385	0.012
GAU777/1	0	2.71	0.07	0.004	0.0025	0.005	0.001	0.002	0.035	0.002
GAU777/2	0	3.68	0.0005	0.0012	0.0010	0.004	0.0002	0.0006	0.011	0.003
All core data, <750 mm	**Average**	**8.38**	**0.04**	**0.09**	**0.03**	**0.10**	**0.02**	**0.02**	**0.75**	**0.03**
	Maximum	**37.83**	**0.12**	**0.26**	**0.10**	**0.32**	**0.06**	**0.07**	**2.30**	**0.07**
2017 core data only, <750 mm	**Average**	**14.15**	**0.05**	**0.12**	**0.04**	**0.14**	**0.03**	**0.03**	**1.02**	**0.04**
	Maximum	**37.83**	**0.12**	**0.26**	**0.10**	**0.32**	**0.06**	**0.07**	**2.30**	**0.07**

Table 3.6: Dragon bioshield concrete fingerprint derived using average concrete sample results from the 2005, 2013 and 2017 cores (Table **3.2**), and scaling from the concrete brick USR fingerprint ([15], Table **3.3**) and decayed SGHWR activation analysis ([44; 15], Table **3.4**). Fingerprint presented at 05/04/2018 [15].

Radionuclide	Half-life	Average Activity [Bq/g]	Average [%]	Maximum Activity [Bq/g]
^{3}H	1.23E+01	8.381	88.693	37.829
^{14}C	5.70E+03	0.041	0.430	0.117
^{41}Ca	1.02E+05	0.085	0.904	0.263
^{60}Co	5.27E+00	0.028	0.298	0.098
^{63}Ni	1.00E+02	0.098	1.040	0.318
^{133}Ba	1.05E+01	0.020	0.209	0.062
^{151}Sm	9.00E+01	0.024	0.252	0.074
^{152}Eu	1.35E+01	0.746	7.895	2.300
^{154}Eu	8.59E+00	0.026	0.280	0.074
Total		**9.45**	**100.00**	**41.13**

Derivation of Bioshield Barytes Concrete Fingerprint

367 Reinforced barytes concrete has been identified in the Dragon facility. The extent and composition are not known (INV-DRAGON-006), although its presence is indicated on some drawings (e.g. AE149323; Figure **3.14**), generally around penetrations. The drawings suggest that barytes concrete was used either to provide a greater level of shielding (suggesting that higher activation was expected), or to provide a similar level of shielding over a smaller concrete thickness. The significantly higher Ba content in such concrete will result in different activation proportions as Ba is only a small constituent of ordinary concrete. The Ba content in heavy (barytes) concrete could be as much as 30-50% [127; 128] which, on a simple scaling basis, would substantially increase the ^{133}Ba inventory from that in ordinary concrete. Žagar and Ravnik [129] measured neutron activation in ordinary high-density concrete and barytes concrete. Ordinary concrete contains mainly limestone ($CaCO_3$) and silicon oxides (SiO_2). The aggregate phase in barytes concrete is composed mainly from barytes minerals (barite $BaSO_4$ or witherite $BaCO_3$). Both types of concrete considered by Žagar and Ravnik were made with Portland cement [129, §2.1]. The study concluded that the residual radioactivity in ordinary concrete is predominantly due to the presence of trace elements, with key isotopes being ^{46}Sc, ^{65}Zn, ^{54}Mn, ^{60}Co, and ^{152}Eu, but the residual gamma radioactivity in barytes concrete was predominantly due to ^{133}Ba.

368 It has not been possible to identify what barytes concrete composition was used when Dragon was constructed and no samples of barytes concrete from Dragon have been analysed. Therefore, indicative composition information on barytes concrete has been sought. Two studies comparing ordinary and heavy/barytes nuclear concrete were obtained. The first, by García *et al.* [127, Tab.1], a neutron activation study, included composition information for an ITER-facility-like concrete (assumed to be a Ca-based

concrete although the proportion is not as high as in other ordinary concrete compositions) and a barytes concrete – the Ca and Ba components were 8.28 wt% and 0 wt% in the former and 5.02 wt% and 46.34 wt% in the latter. The second study identified, by Tefelski *et al.* [128] on neutron shielding concretes, included composition information for ordinary, borated, heavy, borated heavy and reinforced concretes used in the following reactors: Ulysse de Saclay and the Réacteur Universitaire de Strasburg. The composition data are presented in Table **3.7**.

As a rough indicative estimate, the average reduction in the proportion of Ca content and the increase in Ba content for the three shielding concretes identified is assumed to scale directly to a change in the activation inventory for the bioshield concrete. No attempt was made to account for the neutron absorption characteristics of Ba versus Ca. Table **3.8** summarises the Ca and Ba in each pair of ordinary and barytes concrete. Using the resulting average data, the Dragon bioshield activity associated with Ca has been assumed to correspond to 12.4 wt% Ca content in the Ca-based concrete composition, which reduces to 7.8 wt% in the Ba-based concrete. Similarly, the activity associated with Ba in the Ca-based Dragon bioshield concrete fingerprint (Table **3.6**) is assumed to correspond to an original Ba content of 0.5 wt%, which increases to 40.2 wt% in a Ba-based concrete. The fingerprint proportions for ^{41}Ca and ^{133}Ba activities were then scaled by these proportions. Changes in trace and other element proportions are neglected, as are any differences in neutron transport properties and concrete activation. It is emphasised that this is a very indicative estimate for the inventory associated with barytes concrete activation in Dragon (INV-DRAGON-006), especially given the simplifications made in this fingerprint estimate and the barytes concrete volume estimate (see Section 3.4.3). The resulting Dragon bioshield barytes concrete fingerprint is presented in Table **3.9**.

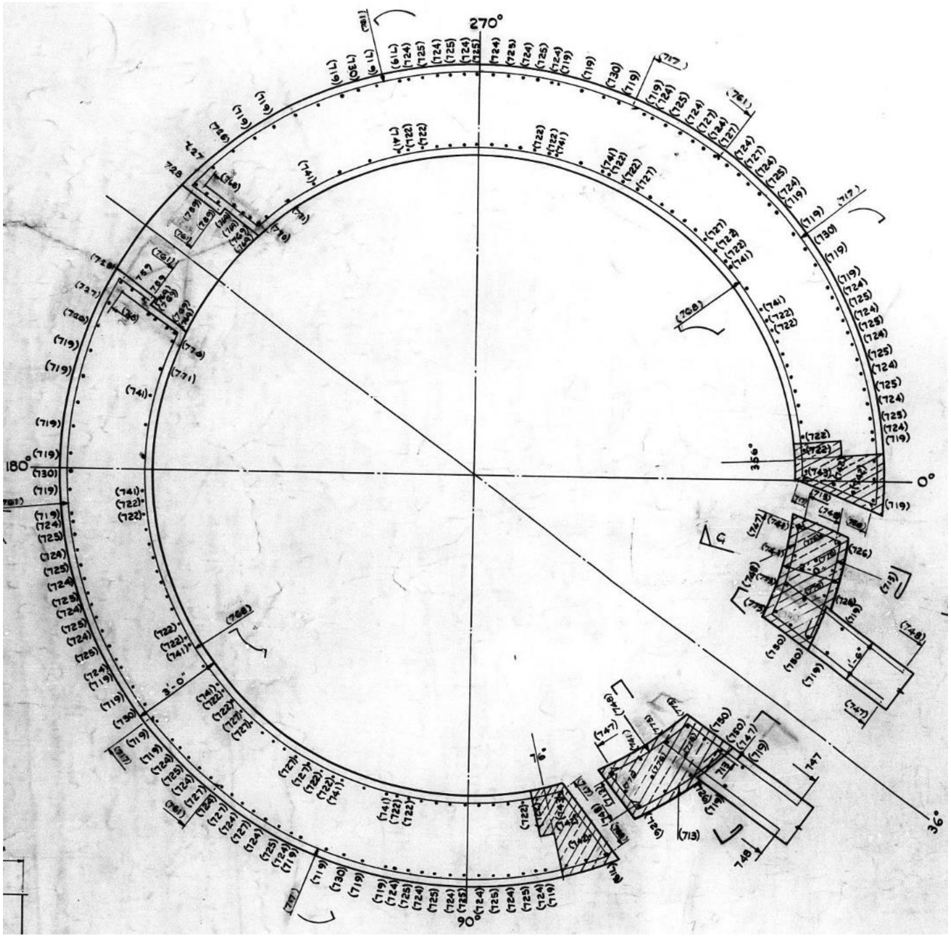


Figure 3.14: Extract from Drawing AE149323 (UKAEA, Version E, June 1961), a plan view of the Dragon bioshield, where the hatched area indicates the presence of barytes concrete.

Table 3.7: Concrete composition information for the Ulysse de Saclay and the Réacteur Universitaire de Strasburg reactors [128, Tab.1 and Tab.2]. The abbreviation used are: OC – ordinary concrete, BC – borated concrete, HC – heavy concrete, BHC – borated heavy concrete, RC – reinforced concrete. "U" and "R" designators mean Ulysse de Saclay or Réacteur Universitaire de Strasburg.

Table 1

Concrete mass composition of Ulysse de Saclay decommissioned reactor (Ulysse). Units in ppm

Concrete type:	OC-U	BC-U	HC-U	BHC-U
Density g/cm^3:	2.63	2.58	3.20	3.20
H	3502	3505	1114	1119
B	16	20042	6	21983
C	37190	37218	18553	18623
O	529086	514765	306846	285852
F	0	0	51559	51756
Na	920	921	487	489
Mg	2776	2779	793	796
Al	14563	14574	4751	4769
Si	217718	211824	71586	68227
P	262	262	95	96
S	3402	3404	92899	93253
K	5153	5157	3459	3472
Ca	174094	174224	109248	109665
Ti	961	961	197	198
Mn	252	252	1102	1107
Fe	9645	9652	2025	2033
Sr	328	328	9557	9593
Ba	132	132	325722	326972

Table 2

Concrete mass composition of Strasburg University decommissioned reactor (RUS). Units in ppm

Concrete type:	OC-R	BC-R	HC-R	RC-R
Density g/cm^3:	2.63	2.58	3.20	4.42
H	1648	8851	1180	2498
B	22	18314	5	5167
C	25168	27411	2499	7734
O	512453	520201	282393	146713
F	0	0	58286	0
Na	6406	5631	574	1589
Mg	2083	7652	1216	2159
Al	24083	22180	2589	6258
Si	278133	210148	40100	59296
P	385	348	44	99
S	4441	2513	105975	709
K	4096	9103	1050	2570
Ca	114096	149616	73982	43065
Ti	948	821	120	232
Mn	495	1599	49	451
Fe	11350	13134	1440	720663
Sr	671	1645	10597	553
Ba	13522	834	417901	244

Table 3.8: Comparison of Ca and Ba proportions in three pairs of ordinary and heavy/barytes concrete (OC and HC, respectively).

Data Source	Ca wt% reduction		Ba wt% increase	
	wt% in OC	wt% in HC	wt% in OC	wt% in HC
ITER-facility-like and Barite concrete [127, Tab.1]	8.3	5.0	0.00	46.3
Ulysse de Saclay ordinary and heavy concrete [128, Tab.1]	17.4	10.9	0.01	32.6
Réacteur Universitaire de Strasburg ordinary and heavy concrete [128, Tab.2]	11.4	7.4	1.35	41.8
Average	**12.4**	**7.8**	**0.5**	**40.2**

Table 3.9: Indicative Dragon bioshield barytes concrete fingerprint derived by scaling the Ca and Ba proportions in the Dragon bioshield ordinary concrete fingerprint (Table **3.6**) using indicative concrete composition information from [127; 128]. Fingerprint presented at 05/04/2018 [15]. Only the activities of ^{41}Ca and ^{133}Ba (shaded) are changed from the ordinary concrete fingerprint.

Radionuclide	Average Activity [Bq/g]	Average [%]	Maximum Activity [Bq/g]
^{3}H	8.381	75.234	37.829
^{14}C	0.041	0.365	0.117
^{41}Ca	0.054	0.483	0.166
^{60}Co	0.028	0.253	0.098
^{63}Ni	0.098	0.882	0.318
^{133}Ba	1.742	15.636	5.437
^{151}Sm	0.024	0.214	0.074
^{152}Eu	0.746	6.697	2.300
^{154}Eu	0.026	0.238	0.074
Total	**11.14**	**100.00**	**46.41**

Derivation of Bioshield Rebar Fingerprint

369 As indicated in Table **3.2**, the rebar data set is limited - there are four rebar samples and only two of these provide a few above-LOD results. Review of the mild steel baseplate fingerprint [126] and the thermal shield sample results (where available from the 2017 core) suggest significantly different ratios for a number of radionuclides (particularly ^{55}Fe, ^{60}Co and ^{63}Ni) compared with that calculated using the decayed SGHWR rebar activation analysis [44; 15]. While the SGHWR is an entirely different reactor and the activation analysis is not directly comparable, review of measured and predicted ratios for radionuclides in activated Dragon bioshield concrete were found to be surprisingly similar. Therefore, it is considered that the significant difference between activation

analysis for rebar and the baseplate fingerprint arises due to assumptions about the composition of the reinforcing steel. The baseplate fingerprint [126] is based on samples from mild steel components near the Dragon Reactor core, whereas steel reinforcing bar typically has a wide range of trace contaminants and the steel grade may be less controlled. Review of the activation analysis [44] shows a two order of magnitude difference between the assumed Co content for mild steel and for rebar in the activation model. The exact composition of rebar used in the Dragon bioshield is not known (INV-DRAGON-006).

370 Therefore, the Dragon bioshield rebar fingerprint was derived by considering both the baseplate fingerprint and the SGHWR activation analysis to calculate two separate fingerprints and then, due to a lack of any further information on the Dragon rebar, the most conservative of the two was selected (INV-DRAGON-006). Scaling to the 2017 thermal shield samples was not applied since measurements for a thermal shield are included in the baseplate fingerprint [126].

371 Considering the 2017 rebar sample alone, the two alternative fingerprint options were calculated by scaling to the measured ^{55}Fe and ^{137}Cs values. The average of the SGHWR flux activation data (see Table **3.4**) was used as there is no knowledge from where the single rebar sample measurement was taken. The calculated fingerprint using the steel baseplate fingerprint substantially bounds the total activity concentration derived using the SGHWR activation analysis (16.8 Bq/g compared with 2.2 Bq/g) and also includes a higher proportion of ^{14}C, which is long-lived and of more relevance to the long-term safety assessment than shorter-lived radionuclides or those that that are not as mobile or that have lower dose coefficients. The radionuclides present in the activation analysis short-list (Table **3.4**) that are not in the baseplate fingerprint comprise ~5% of the activation analysis total activity. Therefore, in the absence of any information on the composition of the bioshield rebar, the fingerprint derived using the mild steel baseplate fingerprint to fill the data gaps has been selected to produce the Dragon bioshield rebar inventory estimate.

372 Inclusion of the 2005 rebar sample data (measured above LOD and inferred using the baseplate fingerprint) slightly increases the proportion of ^{3}H, ^{14}C, and ^{36}Cl by a few percent, increases ^{55}Fe by ~13%, increases ^{63}Ni by ~18%, and decreases ^{60}Co by ~40%. This is a substantial difference compared to what is (bar a few data points) the mild steel baseplate fingerprint. An increase in the ^{55}Fe proportion is consistent with the SGHWR activation analysis, although ~70% is expected in that (compared with 25% in the derived fingerprint), and a smaller proportion of ^{63}Ni.

373 The resulting Dragon bioshield rebar fingerprint is presented in Table **3.10**.

Table 3.10: Dragon bioshield rebar fingerprint derived using average rebar sample results from the 2017 and 2005 cores (Table **3.2**) and scaling from the mild steel baseplate fingerprint ([126], Table **3.3**). Fingerprint presented at 05/04/2018 [15].

Radionuclide	Half-life	Average Activity [Bq/g]	Average [%]	Maximum Activity [Bq/g]
^{3}H	1.23E+01	0.21	7.68	0.36
^{14}C	5.70E+03	0.16	5.73	0.23
^{36}Cl	3.01E+05	0.05	1.89	0.11
^{55}Fe	2.74E+00	0.68	24.56	1.60
^{60}Co	5.27E+00	0.12	4.43	0.12
^{63}Ni	1.00E+02	1.54	55.67	3.11
^{137}Cs	3.02E+01	0.0012	0.04	0.001
Total		**2.77**	**100.00**	**5.54**

3.4.3 Inventory Estimate

374 In developing a cautious inventory, a uniform activity profile has been assumed through the Dragon bioshield, but only applied to the first 750 mm due to the lack of artificial activity in the outermost section of the bioshield. The inventory calculated here only includes that due to activation inside the concrete bioshield; contamination on the surface is accounted for separately (see Section 3.4.4).

375 The density of the bioshield concrete has been determined from bioshield samples to be 2,437.3 kg/m^3 [15; 130]. Using the dimensions in Table **3.1**, the volume of concrete in the first 750 mm (minus that associated with the barytes concrete) is 1.36E+02 m^3 with mass 3.31E+05 kg [15].

376 The extent of the barytes concrete is not known (INV-DRAGON-006). An approximate volume of barytes concrete has been estimated by scaling from drawing AE149323 and conservatively assuming that the indicated barytes concrete region extends through the full height of the bioshield, whereas in reality it is believed to have only been used selectively around penetrations. The barytes volume in the first 750 mm bioshield thickness is thus estimated to be 26.3 m^3 [15]. Using a barytes concrete density of 3,650 kg/m^3 [131, Appendix II, Tab.A.2.5], the barytes mass is estimated to be 9.59E+04 kg.

377 The Dragon Project Manager [108, p.10] has previously estimated that there is 150 kg of rebar per m^3 of bioshield concrete. It has subsequently been identified that a minimum of 1.5% steel within the bioshield was planned prior to construction [131, Section 2.3.A]. Based on the density of the bioshield, this equates to a minimum of ~36 kg/m³. However, 150 kg/m^3 (~6% steel) is assumed to pessimistically estimate the inventory, which then also makes some allowance for any additional steel above the specified minimum and for any steel activation past the 750 mm depth demonstrated for the bioshield concrete. Thus, the mass of rebar in the activated bioshield section is estimated to be 2.43E+04 kg.

378 The height of the bioshield extends above ground level by 5.4 m (Table **3.1**); this top portion of the bioshield will be demolished and used as part of the backfill for the

Dragon voids. Based on this height and assuming constant cross-sectional area and uniform activation contamination, 43% of the calculated bioshield inventory will contribute to the backfill inventory. Thus, the inventory presented in this section is the 57% attributed to the in-situ portion of the bioshield; the other 43% of the inventory is recorded in Section 3.8. The assumed contaminated volume and mass of in-situ bioshield Portland concrete (i.e. below ground level) are 7.75E+01 m^3 and 1.89E+05 kg, respectively. The assumed contaminated mass and volume of in-situ barytes concrete are 1.50E+01 m^3 and 5.47E+04 kg, and the mass of rebar in the in-situ activated bioshield section is estimated to be 1.39E+04 kg.

379 The maximum and average activity concentrations derived in the three Dragon bioshield fingerprints have been decayed to the inventory reference date and are presented in Table **3.11**. The decay calculations have been undertaken using the GoldSim-RT software package [132; 133; 134] and modelling the decay chains as specified in the PA approach report [135, §6].

380 An estimate of the radioactive inventory is presented in Table **3.12**, based on the data and approach described in Section 3.4.2 and [15], using the average fingerprint values for each of the three bioshield materials. The resulting average activity concentration across the activated bioshield volume is also presented (accounting for the proportions of ordinary concrete, barytes concrete and rebar). The maximum activity concentration is given as that for the bioshield barytes concrete component as this is the highest of the three material activity concentrations (see Table **3.11**).

Table 3.11: Derived Dragon bioshield fingerprints and calculated average and maximum activity concentrations presented at 01/01/2027. ^{148}Sm and ^{152}Gd are included in the fingerprint for completeness, as they were included in the modelled decay chains when decaying the data from 5/04/2018 to 01/01/2027, but their impact will remain negligible unless a significantly longer decay period is assumed.

Radionuclide	Bioshield Concrete - Portland			Bioshield Concrete - Barytes			Bioshield Rebar		
	Average [Bq/g]	Average [%]	Maximum [Bq/g]	Average [Bq/g]	Average [%]	Maximum [Bq/g]	Average [Bq/g]	Average [%]	Maximum [Bq/g]
^{3}H	5.125	87.225	23.134	5.125	75.234	23.134	0.130	6.821	0.222
^{14}C	0.041	0.691	0.116	0.041	0.596	0.116	0.159	8.320	0.228
^{36}Cl							0.052	2.747	0.112
^{41}Ca	0.085	1.454	0.263	0.054	0.789	0.166			
^{55}Fe							0.074	3.901	0.175
^{60}Co	0.009	0.152	0.031	0.009	0.131	0.031	0.039	2.037	0.039
^{63}Ni	0.092	1.574	0.299	0.092	1.357	0.299	1.452	76.122	2.931
^{137}Cs							0.001	0.051	0.001
^{133}Ba	0.011	0.189	0.035	0.979	14.374	3.057			
^{148}Sm	4.38E-30	7.45E-29	1.35E-29	4.38E-30	6.42E-29	1.35E-29			
^{151}Sm	0.022	0.379	0.070	0.022	0.327	0.070			
^{152}Gd	9.41E-15	1.60E-13	2.90E-14	9.41E-15	1.38E-13	2.90E-14			
^{152}Eu	0.477	8.115	1.470	0.477	7.000	1.470			
^{154}Eu	0.013	0.223	0.037	0.013	0.192	0.037			
Total	**5.876**	**100.000**	**25.455**	**6.813**	**100.000**	**28.380**	**1.908**	**100.000**	**3.708**

Table 3.12: Estimated Dragon bioshield in-situ (i.e. below ground level cutline) disposal inventory, with the inventory based on the average fingerprint values for each material, and a maximum activity concentration drawn from that for the bioshield barytes concrete component, presented for an inventory reference date of 01/01/2027.

Radionuclide	Bioshield Concrete – Portland [MBq]	Bioshield Concrete – Barytes [MBq]	Bioshield Rebar [MBq]	Total Disposal Inventory [MBq]	Average Activity [Bq/g]	Maximum (Barytes) Activity [Bq/g]
^{3}H	9.68E+02	2.80E+02	1.81E+00	1.25E+03	4.856	23.134
^{14}C	7.67E+00	2.22E+00	2.20E+00	1.21E+01	0.047	0.116
^{36}Cl			7.27E-01	7.27E-01	0.003	
^{41}Ca	1.61E+01	2.94E+00		1.91E+01	0.074	0.166
^{55}Fe			1.03E+00	1.03E+00	0.004	
^{60}Co	1.69E+00	4.88E-01	5.39E-01	2.71E+00	0.011	0.031
^{63}Ni	1.75E+01	5.06E+00	2.01E+01	4.27E+01	0.166	0.299
^{137}Cs			1.36E-02	1.36E-02	5.29E-05	
^{133}Ba	2.09E+00	5.36E+01		5.57E+01	0.216	3.057
^{148}Sm	8.27E-28	2.39E-28		1.07E-27	4.14E-30	1.35E-29
^{151}Sm	4.20E+00	1.22E+00		5.42E+00	0.021	0.070
^{152}Gd	1.78E-12	5.15E-13		2.29E-12	8.91E-15	2.90E-14
^{152}Eu	9.01E+01	2.61E+01		1.16E+02	0.451	1.470
^{154}Eu	2.47E+00	7.16E-01		3.19E+00	0.012	0.037
Total	**1.11E+03**	**3.73E+02**	**2.65E+01**	**1.51E+03**	**5.861**	**28.380**

3.4.4 Sensitivity Analysis and Further Characterisation

381 Although the bioshield inventory estimate is supported by a substantial dataset derived from core samples, there is some remaining uncertainty, particularly regarding the representativeness of the fingerprint and whether it accounts for potentially high activation areas indicated by barytes concrete (INV-DRAGON-004) and the fact that no activation modelling has been undertaken for the Dragon Reactor bioshield (INV-DRAGON-005). This uncertainty is accounted for by making alternative assumptions and exploring the effect on the calculated inventory.

382 Calculating the inventory based on the maximum, rather than average, fingerprint values for each material gives an alternative total disposal inventory as shown in Table **3.13**. The maximum fingerprint values are themselves conservative because they are derived from the maximum for each radionuclide from all samples, rather than the sample with the highest total activity; it is highly unlikely that a single sample would contain the maximum for every radionuclide.

Table 3.13: Alternative estimated Dragon bioshield in-situ (i.e. below ground level cutline) disposal inventory, with the inventory based on the maximum (rather than average) fingerprint values for each material, presented for an inventory reference date of 01/01/2027.

Radionuclide	Bioshield Concrete – Portland [MBq]	Bioshield Concrete – Barytes [MBq]	Bioshield Rebar [MBq]	Total Disposal Inventory [MBq]	Activity Concentration [Bq/g]
^{3}H	4.37E+03	1.27E+03	3.08E+00	5.64E+03	21.9
^{14}C	2.20E+01	6.37E+00	3.16E+00	3.15E+01	0.122
^{36}Cl			1.56E+00	1.56E+00	0.006
^{41}Ca	4.97E+01	9.07E+00		5.88E+01	0.228
^{55}Fe			2.43E+00	2.43E+00	0.009
^{60}Co	5.86E+00	1.70E+00	5.39E-01	8.10E+00	0.031
^{63}Ni	5.65E+01	1.64E+01	4.07E+01	1.14E+02	0.441
^{137}Cs			1.36E-02	1.36E-02	5.29E-05
^{133}Ba	6.53E+00	1.67E+02		1.74E+02	0.675
^{148}Sm	2.55E-27	7.38E-28		3.29E-27	1.28E-29
^{151}Sm	1.31E+01	3.81E+00		1.70E+01	0.066
^{152}Gd	5.48E-12	1.59E-12		7.07E-12	2.75E-14
^{152}Eu	2.78E+02	8.04E+01		3.58E+02	1.391
^{154}Eu	6.90E+00	2.00E+00		8.89E+00	0.035
Total	**4.81E+03**	**1.55E+03**	**5.14E+01**	**6.41E+03**	**24.905**

383 Comparing these values with those in Table **3.12**, the total inventories (MBq) for bioshield Portland concrete and barytes concrete are approximately four times higher, the total inventory for rebar is approximately two times higher, and the average activity concentration (Bq/g) is approximately four times higher. The activity concentration is equivalent to 0.025 GBq/tonne, still substantially below the upper limit for LLW in the UK (4 GBq/tonne alpha and 12 GBq/tonne beta/gamma).

384 A different way of accounting for the remaining uncertainty would be to uplift the total inventory by a simple scaling factor of, say, ten, but this would not take into account maximum fingerprint values for individual radionuclides that may be more than ten times higher than the average value. In the case of the bioshield, using the maximum fingerprint values results in less than an order of magnitude increase in the total inventories and average activity concentration. As there is relatively high confidence in the underpinning dataset, it is considered that these alternative estimates sufficiently cover the remaining uncertainties and any uplift beyond them would be unnecessarily conservative.

385 It may be possible in the future to obtain some radiological and chemical samples from known barytes concrete locations in order to characterise the precise specification of barytes concrete used and to determine whether these were higher activation areas (and incorporate this into the inventory estimate if so). However, it is expected that any new bioshield inventory estimate underpinned by such data would be bounded by the alternative estimates presented above. Both the reference and alternative inventories will be considered in the radiological PA.

3.5 Dragon Reactor Building (B70) General Contamination

3.5.1 Feature Description

386 The Dragon Project Manager [108, §3.2] notes that there are two main sources of contamination within the building structure itself:

- Surface contamination of the walls, floors and ceilings being exposed to the general Dragon area atmosphere throughout operation and decommissioning, where contaminants may have been in the airflow or generated as a result of operations/decommissioning; and
- Tritium ingress into the concrete structures from the storage of millions of Gas Tritium Luminescent Devices (GTLDs, also known as Betalites), which were recovered from old Trimphones pending the completion of a safe recovery process for the tritium [136, §1] and were stored on the -25' level of B70 in the late 1980s.

387 The surface area that may have been exposed to contaminants has been calculated using data supplied from the Dragon Project Team and sourced from the “Dragon Start Condition.iam” CAD Model [15]. This includes all concrete / brick surfaces for the entire building, including the walls, floors and ceilings as well as the vent plant room, as shown in Figure **3.4**, that are expected to remain at the end state, whether in-situ below ground or as part of the demolished above-ground portion that will be emplaced as backfill. The steel shell surface area has also been included, but only for the section below ground level (which is proposed to remain in-situ due to accessibility challenges between the two concrete walls[16]), as the section above is expected to be removed. The total surface area is calculated to be 16,883 m^2 [15]; note that this surface area excludes that for building B78, which is accounted for separately in Section 3.7.

388 The surface area of the below-ground part of the structure to remain in-situ has been estimated by: (1) accounting for the surface area of those features and components with known proportions above/below ground level (e.g. the bioshield, steel shell); and (2) for the rest of the below-ground structure, the total surface area of all levels in the

building has been proportioned according to the ratio of the building height to the ground level and assuming that there is equal surface area on every floor of the building (INV-DRAGON-011). That is, the base of the first main floor in the structure is at -25' (neglecting the reactor pit at -29') and the base of the roof area is at +61', so 38% of the building is below ground. The total below-ground surface area is estimated to be 6,458 m^2 [15].

389 The surface area of the Betalite store area (area 121 in Figure **3.15**) has been estimated by approximating the red dashed area as a rectangle (there are no internal walls or other surfaces within this area). The length of the rectangle is assumed to form 1/6th of the Wall A inner circumference, the width is the difference between the Wall A inner and Wall B outer radii, and the height is 10' (the difference between -25' and -15' levels). This approach results in an approximate surface area of 258 m^2 for the Betalite store (INV-DRAGON-011), which is entirely below ground [15].

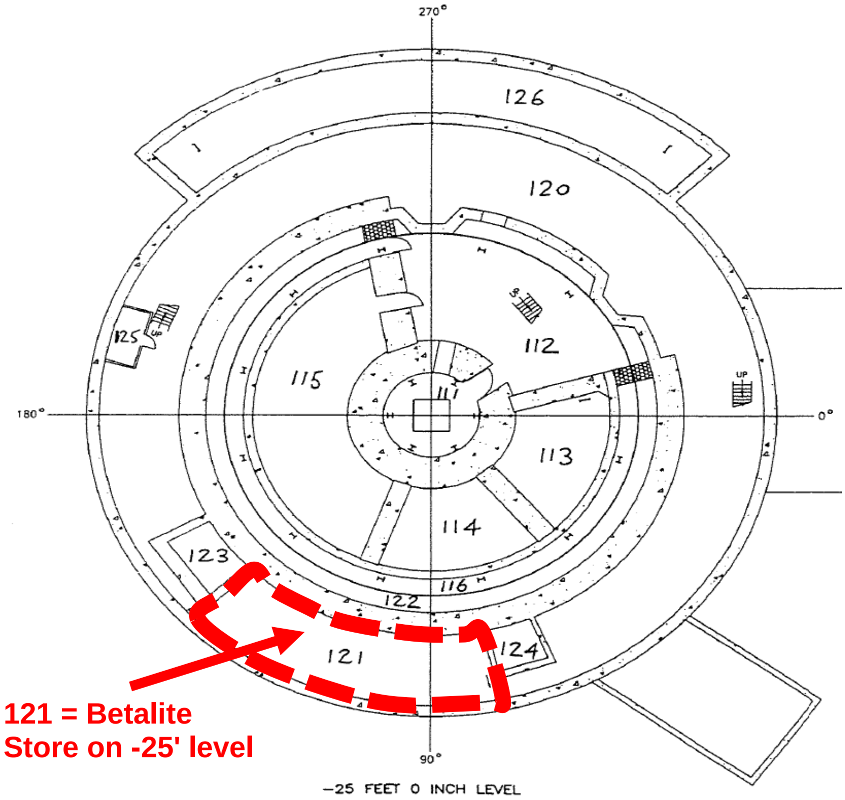


Figure 3.15: Plan view of the B70 Dragon Reactor building -25' level indicating the location and scope of the Betalite store area (edited from [137]).

3.5.2 Origin and Constraints on Radiological Inventory

390 The Dragon building general surface contamination inventory is based on three main sources:

- Radiological characterisation data from ten sampling datasets (giving a total of 99 surface samples) taken at various locations throughout the Dragon Complex, upon which a contamination fingerprint is derived.

- An in-situ sampling campaign of the Dragon building in March 2018, leading to identification of a hotspot activity upon which to scale the fingerprint.
- A probe response calibration and activity conversion using a standardised NRS procedure.

391 The ^{3}H ingress inventory is based on two sampling datasets from the ten used for the surface contamination fingerprint, which were for over 40 concrete cores sub-sampled along their length giving a total of 165 “depth” concrete samples (excluding the paint surface layer).

Characterisation Data and General Building Contamination Fingerprint

392 Various sampling campaigns have been carried out around the Dragon facility since operations ceased and the Betalites were disposed of, and a number of general area contamination fingerprints have been developed, some of which use results for items subsequently removed and disposed of (these existing fingerprints have not been used in this inventory assessment). A new general area fingerprint has been derived in this inventory assessment specifically for application to the exposed building structure that will remain on site, that incorporates all available relevant sampling data obtained over the last few years and makes use of historic sampling data that can be verified by contemporary reports.

393 A number of sampling and analysis campaigns of the building incorporating samples from the various levels and radials of B70, including some of the equipment, and locations in B78, have been undertaken. Results from ten sampling campaigns have been used to derive the general building contamination fingerprint, as described in Table **3.14**.

394 An updated version of the official Dragon general area FP, FP002, was issued in October 2022 [138]. This considered several hundred samples taken since 2006 and was derived for the purpose of characterising waste consigned off site during ongoing decommissioning. The underlying dataset was reviewed to determine whether any new results should be added to the FP derived for the purposes of this inventory estimate. However, all samples not already included are from metal and other infrastructure that will be removed, and therefore do not provide useful constraints on the inventory for on-site disposal.

Table 3.14: The ten sampling datasets used to derive the Dragon general building contamination fingerprint. The two sample sets coloured green were used to estimate the ^{3}H ingress inventory. It is assumed that any surface metal and wood samples will not remain on site at the end state and so results from all such samples (nine in total) were excluded from the fingerprint.

Sample reference and source report	Description
REPE1112 dated 14/10/99, revised in 2002 [139]	4 smear samples from the south, north and west walls in B78, and the floor.
REPE1463 dated 21/03/1999, revised 15/05/00 [140]	5 core samples from unknown locations in B78.

Sample reference and source report	Description
REPG1316 dated 17/12/2001 [141]	2 core samples, a paint sample from the B70 outer containment area at -25' and a concrete sample from the B70 inner containment at +18'.
REPE1126 (revised) dated 18/01/02 [142]	10 surface samples from the B70 inner and outer containment at various building levels.
RA13861 dated 13/02/02 [143]	1 wood sample and 1 concrete sample from unknown locations in B70 (assumed to be from the inner containment).
L060152, dated 11 April 2006 [144]	31 concrete core samples divided to give a total of 151 concrete and paint samples, taken from the inner and outer containment at various levels in B70, as well as in the Betalite store area and in B78. Results provided at depths through the cores.
GAU748 date 15 February 2006 and GAU800, dated 24 April 2006 [145; 146]	10 paint and 8 metal samples taken from the vehicle airlock between B70 and B78.
GAU3558, dated 22 August 2016 [147]	2 paint samples from either side of the blast door separating B70 and B78 (the door is on the B78 side of the vehicle airlock). The sampling plan is documented in [148].
GAU3296, dated 1 April 2015 [149]	1 paint sample from approximately 10' in one of the corners of B78 and is of interest due to the ^{60}Co levels that were found, which was attributed to decommissioning works such as size reduction that have been carried out in B78 [150].
Samples WA/SAMP/0339 to WA/SAMP/0358, assumed date 30/05/2003 [136]	20 paint and core samples taken to determine whether tritium had permeated the concrete walls – samples taken from the inner and outer walls at -25', 0', +18' and +36'. Results provided at depths through the cores.

395 A table of the raw sample data is too large to sensibly include in this report but is provided in the accompanying spreadsheet [15]. However, Table **3.15** summarises the available surface samples characterisation data for selected radionuclides. It is necessary to assess if there are distinct sub-groups within the characterisation dataset or if it is appropriate to average across all surface samples from such a wide range of areas. Therefore, the data presented in Table **3.15** have been grouped and analysed according to the area of the Dragon Complex in which the sample was taken and also by the building level of the sample.

396 Of the surface samples, the radionuclides reported with the most above-LOD results are ^{3}H (68), ^{137}Cs (58), and ^{60}Co (26), and therefore the ratios between these radionuclides were considered to see if there are significant differences. The surface samples from the Betalite store area only report ^{3}H values, so the ratios cannot be calculated for this area.

- Considering the activity ratio by building area, the ^{137}Cs:^{3}H ratio suggests the outer (0.09), vehicle airlock (0.13) and B78 (0.08) sample results are similar. The ratio for the inner wall sample group is higher (0.43) but probably still within reason. When considering the same ratio across B70 building levels there is not much variation (0.06-0.25).
- The ^{137}Cs:^{60}Co ratio suggests the inner (10.70), outer (8.76) and B78 (11.33) sample groups are similar. The vehicle airlock is a bit higher (19.40), but it is uncertain why. There is also a bigger variation in B70 level ratios (5.42-17.87), particularly for -25’ (6.98) and -18’ (5.42), but these ratios are typically calculated from only one to three samples.

397 There is always a reasonable amount of variation in reported ^{3}H activity, but the Betalite area average is substantially higher (1,366 Bq/g based on five results, whereas the next highest average is 42.7 Bq/g for two B78 samples). There is one very high ^{3}H paint sample result (6,600 Bq/g, sample WA/SAMPLE/1164 / L060152-161 from the floor in the Betalite area) that biases the whole ^{3}H dataset. The 6,600 Bq/g result is assumed to either be a true result that is highly localised, or to be in error as this is substantially higher than the next highest ^{3}H measurement across all samples (226 Bq/g, from the -25' level, but not from the Betalite area - the next highest ^{3}H result in the Betalite area is only 100 Bq/g). Comparison of the percentage of ^{3}H in the paint to the subsequent concrete layer behind it for different samples in the Betalite area shows that the concrete ^{3}H is 0.04% of that in the paint for the high floor sample, but the percentage is more than two orders of magnitude higher for the inner and outer wall samples in the Betalite area (17% for WA/SAMPLE/1161 inner wall and 12% for WA/SAMPLE/1162). Given this uncertainty, for this inventory assessment the anomalously high sample (WA/SAMPLE/1164) has been excluded from the dataset for the reference inventory estimate. The impact of this exclusion is to reduce the average ^{3}H activity for the Betalite area from 1,366 Bq/g to 56.9 Bq/g based on the four remaining sample results. The possibility that the sample represents a real, albeit localised, area of very high ^{3}H contamination (INV-DRAGON-007) is explored further in sensitivity analysis in Section 3.5.4.

398 Following review of the results, it was decided to group and use the samples for the B70 inner and outer wall areas, the vehicle airlock and B78 to derive the general building surface contamination fingerprint. However, to avoid unduly over-estimating the ^{3}H surface contamination inventory, it was decided to apply the Betalite area results in a separate fingerprint that uses the ^{3}H results from the Betalite area and the combined results for the other radionuclides.

399 In developing the general building surface contamination fingerprint two aspects are noted. First, for many radionuclides in the characterisation dataset, the measured activities are very low and could be considered to be at the level of noise in the results rather than being statistically meaningful. In addition, other than for ^{3}H, ^{137}Cs and ^{60}Co, there are insufficient results for the identified radionuclides to draw statistically meaningful conclusions (regarding activity or distribution across the facility), particularly for application of a general fingerprint across the whole Dragon facility (INV-DRAGON-004 and INV-DRAGON-007). Further characterisation to reduce this uncertainty may be undertaken as decommissioning proceeds. Nonetheless, the

approach taken in developing the fingerprint and assumptions made regarding its application to the entire building are generally conservative.

400 The approach taken to inclusion/exclusion of radionuclides from the general surface contamination fingerprint is summarised as follows:

- Given that the samples have been taken from a wide range of areas and that there are few pairs of radionuclides from individual samples in each building area to calculate radionuclide ratios for scaling, results have not been inferred where results for individual radionuclides were either not requested or were below the LOD.
- The following radionuclides were excluded because no sample results are above the LOD: ^{57}Co, ^{58}Co, ^{129}I, ^{134}Cs, ^{236}U, ^{65}Zn, ^{54}Mn, ^{154}Eu, and ^{155}Eu.
- ^{22}Na is excluded as it is commonly misidentified and only noted in four results from 1999 and, from a long-term PA viewpoint, it is also short-lived (2.6 y) so it is conservative to maximise other longer-lived radionuclides in the fingerprint.
- ^{40}K, measured above the LOD in 92 surface and depth samples, is not included in the fingerprint because it occurs naturally [124].
- ^{152}Eu is observed in two metal airlock samples from 2006 (GAU748/11 and GAU748/18), but it was at the LOD in another 151 paint/concrete samples (also from 2006), many with very low LODs. This radionuclide is expected to only be present via contamination though operations/decommissioning. It is not included in the original general area FP (FP-002) [108; 151], and was not reported in recent Dragon transfer flask [152] or B78 ventilation system [117] fingerprints. As it is relatively short-lived (13 y half-life), it is not a significant radionuclide for long-term PAs. The measured activities are low, comparable to those of the reported LOD values and, as it is conservative to maximise longer-lived radionuclides in the fingerprint (as it is applied to a closed system in terms of total activity rather than scaling individual radionuclides ratios - see later discussion), ^{152}Eu is excluded from the derived fingerprint. However, it may be appropriate for additional ^{152}Eu samples to be taken in the B78 and vehicle airlock areas in the future and to consider separating out the B78 and vehicle airlock areas from the Dragon general fingerprint, as ^{152}Eu may well only arise in areas where spent fuel was able to easily contaminate surfaces.
- The following short-lived (<1 y half-life) naturally-occurring radionuclides are not considered further: ^{208}Tl, ^{212}Pb, ^{214}Pb, ^{212}Bi, ^{214}Bi, ^{228}Ac, and ^{234}Th.
- ^{232}U was analysed for in two samples with a single above-LOD result (sample G1317 / WA/DRG/C/B70-B78/001, a concrete core from the inner wall area at +18'); the other results for this sample seem reasonable. Of the two results, the LOD is 0.002 Bq/g and the above LOD result is also 0.002 Bq/g, so the single positive result is low and possibly questionable. None of the other isotopes in this decay chain were analysed for in this sample, but ~5 samples do report the short-lived daughter ^{212}Pb, although this is also a daughter of naturally-occurring ^{232}Th (see below). ^{232}U arises through neutron capture on Th/U/Pa or decay from ^{232}Pa or ^{236}Pu; a mechanism for production of ^{232}U at Dragon has not been identified. This isotope was excluded from the original general area fingerprint [108; 151] and it is not present in the B78 Ventilation System [117]

and Dragon fuel transfer flask [152] fingerprints. Therefore, consistent with the fingerprints mentioned, and due to the questionable sole result and lack of identified production mechanism, ^{232}U has been excluded from the derived fingerprint.

- Apart from one concrete sample, all the samples that contain ^{232}Th are from paint. Fonseca and Pecequilo [153] show that for ThO_2 used in white paint, ^{232}Th content can range from 2-26 Bq/kg. Of the six sample results, the maximum is 6 Bq/kg so it could be argued that the ^{232}Th is present naturally and therefore exclude it from the fingerprint (assuming that similar paint has been used at Winfrith). However, ^{232}Th was included (in small proportions) in the original general area (FP002) [108; 151] and Dragon transfer flask [152] fingerprints. As it is long-lived (1.41E+10 y) it is also conservative to include it in the inventory for the PA, so it is retained in the fingerprint derived here.
- Uranium decay chain radionuclides (^{238}U, ^{234}U, ^{230}Th, ^{226}Ra, ^{210}Pb) are observed in a number of samples.
  - ^{210}Pb: Seven of the 21 analysed samples were above the LOD, consisting of one metal, one wood and five paint samples (0.03-0.16 Bq/g).
  - ^{226}Ra: 25 samples were analysed, with two paint samples from the vehicle airlock above the LOD (0.10-0.15 Bq/g).
  - ^{230}Th: Four paint samples from the vehicle airlock (GAU800/1-/4) were analysed, all returning above LOD results (0.001-0.004 Bq/g).
  - ^{238}U and ^{234}U: 11 samples paint and concrete samples from the vehicle airlock and B78 were analysed, all returning above the LOD results (0.004-2.210 Bq/g ^{234}U and 0.002-2.450 Bq/g ^{238}U).

  ^{238}U and ^{234}U are included in the original fingerprint, the B78 ventilation system and the Dragon fuel transfer flask fingerprints [108; 151; 117; 152], but ^{230}Th, ^{226}Ra and ^{210}Pb are not. Of the 25 samples analysed for ^{226}Ra, only two paint samples were above LOD, equivalent to 100-150 Bq/kg, which is higher than the range given by Fonseca and Pecequilo [153] for naturally-occurring ^{226}Ra in paint and the typical value of 40 Bq/kg given by the European Commission [31, Tab.1] for ^{226}Ra in concrete (although the maximum concrete value in [31] is 240 Bq/kg). Naturally occurring ^{226}Ra should be in secular equilibrium with the parent ^{238}U, but the average ^{238}U activity is 0.46 Bq/g and the ^{226}Ra activity is 0.12 Bq/g (note that only one sample reports both ^{226}Ra and ^{238}U, so this can only be an indicative comparison). The greater proportion of ^{238}U would suggest that there is more than a natural source of ^{238}U present if it could be shown that the ^{226}Ra was naturally occurring. As there is no clear justification for excluding these radionuclides, they make small contributions to the derived fingerprint, and because their long-lived nature means it is conservative to include them in the PA, they have been retained in the fingerprint.

401 The resulting derived Dragon general building area surface contamination fingerprint and the fingerprint for the Betalite store area are presented in Table **3.16**.

Table 3.15: Summary of selected characterisation data considered in derivation of the Dragon general building surface contamination fingerprint [15]. Only the results from surface samples are included in this analysis, with the sample activity presented at a common date of 11/04/2006. The results are grouped according to the Dragon Complex area in which the samples were taken, and by the building level from which they were taken. See text for discussion of data and exclusions. The table continues with additional radionuclides on the next page. *(This page is set to print on A3.)*

	Total No. samples	No. surface samples	Ratio ^{137}Cs:^{3}H		Ratio ^{137}Cs:^{60}Co		^{3}H			^{14}C			^{55}Fe			^{60}Co			^{63}Ni			^{90}Sr			^{137}Cs		
			No. Results	Av. ratio	No. Results	Av. ratio	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]	No. Samples >LOD	Av. [Bq/g]	Max. [Bq/g]
Breakdown by Dragon Complex area (surface samples only)																											
Inner wall up to bioshield	80	24	4	0.41	6	12.20	17	11.10	35.74	2	0.31	0.40	0		0.00	6	0.10	0.18	0		0.00	1	0.04	0.04	10	0.99	4.47
Outer wall up to the inner wall	103	31	10	0.09	4	8.76	25	20.99	226.00	0		0.00	0		0.00	4	0.07	0.10	0		0.00	0		0.00	15	0.42	1.07
Betalite store	29	5	0		0		4	56.89	100.34	0		0.00	0		0.00	0		0.00	0		0.00	0		0.00	0		0.00
B78 Fuel Store	29	16	2	0.08	8	11.33	2	42.67	85.00	0		0.00	0		0.00	8	2.14	5.30	0		0.00	4	7.98	22.00	14	14.90	109.65
Vehicle Airlock	23	23	11	0.19	4	17.43	11	1.33	2.68	10	0.21	0.51	0		0.00	4	0.05	0.10	7	0.18	0.76	0		0.00	11	0.28	1.79
Breakdown by Dragon building level (surface samples only)																											
-25'	77	20	1	0.10	3	6.98	17	37.91	226.00	0		0.00	0		0.00	3	0.09	0.14	0		0.00	0		0.00	3	0.50	0.87
-25' excluding Betalite store	48	15	1	0.10	3	6.98	13	32.07	226.00	0		0.00	0		0.00	3	0.09	0.14	0		0.00	0		0.00	3	0.50	0.87
-18'	2	2	0		2	5.42	0		0.00	0		0.00	0		0.00	2	0.08	0.10	0		0.00	0		0.00	2	0.48	0.67
0'	62	16	3	0.25	2	17.87	12	5.87	15.00	0		0.00	0		0.00	2	0.12	0.18	0		0.00	1	0.04	0.04	6	1.07	4.47
+18'	37	12	6	0.24	2	13.98	9	7.58	35.74	1	0.23	0.23	0		0.00	2	0.06	0.07	0		0.00	0		0.00	9	0.47	1.93
+36'	32	8	3	0.06	1	12.73	7	21.81	61.00	0		0.00	0		0.00	1	0.07	0.07	0		0.00	0		0.00	4	0.63	1.07
Vehicle Airlock	23	23	11	0.19	4	17.43	11	1.33	2.68	10	0.21	0.51	0		0.00	4	0.05	0.10	7	0.18	0.76	0		0.00	11	0.28	1.79
B78	29	16	2	0.08	8	11.33	2	42.67	85.00	0		0.00	0		0.00	8	2.14	5.30	0		0.00	4	7.98	22.00	14	14.90	109.65
Unknown	2	2	1	0.09	0		1	5.30	5.30	1	0.40	0.40	0		0.00	0		0.00	0		0.00	0		0.00	1	0.49	0.49
Sample results from all areas (surface samples only)																											
All areas excl. Betalite store	235	94	27	0.18	22	12.21	55	14.79	226.00	12	0.23	0.51	0		0.00	22	0.83	5.30	7	0.18	0.76	5	6.40	22.00	50	4.56	109.65
All areas	264	99	27	0.18	22	12.21	59	17.64	226.00	12	0.23	0.51	0		0.00	22	0.83	5.30	7	0.18	0.76	5	6.40	22.00	50	4.56	109.65

	^{241}Am			^{152}Eu			^{226}Ra			^{230}Th			^{232}Th			^{232}U			^{234}U			^{235}U			^{238}U		
	No. Samples >LOD	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**	**No. Samples >LOD**	**Av. [Bq/g]**	**Max. [Bq/g]**
Breakdown by Dragon Complex area																											
Inner wall up to bioshield	0			0			0			0			1	0.006	0.006	1	0.002	0.002	1	0.01	0.01	1	0.003	0.003	1	0.01	0.01
Outer wall up to the inner wall	0			0			0			0			1	0.003	0.003	0			1	0.02	0.02	0			1	0.01	0.01
Betalite store	0			0			0			0			0			0			0			0			0		
B78 Fuel Store	2	2.22	2.58	0			0			0			0			0			5	0.81	2.21	3	0.19	0.47	5	1.01	2.45
Vehicle Airlock	0			0			2	0.12	0.15	4	0.003	0.004	4	0.002	0.002	0			4	0.01	0.02	1	0.01	0.01	4	0.004	0.007
Breakdown by Dragon building level																											
-25'	0			0			0			0			1	0.003	0.003	0			1	0.02	0.02	0			1	0.01	0.01
-25' excluding Betalite store	0			0			0			0			1	0.003	0.003	0			1	0.02	0.02	0			1	0.01	0.01
-18'	0			0			0			0			0			0			0			0			0		
0'	0			0			0			0			0			0			0			0			0		
+18'	0			0			0			0			1	0.006	0.006	1	0.002	0.002	1	0.01	0.01	1	0.003	0.003	1	0.01	0.01
+36'	0			0			0			0			0			0			0			0			0		
Vehicle Airlock	0			0			2	0.12	0.15	4	0.003	0.004	4	0.002	0.002	0			4	0.01	0.02	1	0.01	0.01	4	0.004	0.007
B78	2	2.22	2.58	0			0			0			0			0			5	0.81	2.21	3	0.19	0.47	5	1.01	2.45
Unknown	0			0			0			0			0			0			0			0			0		
Sample results from all areas (surface samples only)																											
All areas excl. Betalite store	2	2.22	2.58	0			2	0.12	0.15	4	0.003	0.004	6	0.003	0.006	1	0.002	0.002	11	0.37	2.21	5	0.12	0.47	11	0.46	2.45
All areas	2	2.22	2.58	0			2	0.12	0.15	4	0.003	0.004	6	0.003	0.006	1	0.002	0.002	11	0.37	2.21	5	0.12	0.47	11	0.46	2.45

Table 3.16: Dragon general building and Betalite store area surface contamination fingerprints, and average and maximum activity concentrations, derived using 99 surface characterisation samples [136-139] and grouped as discussed above. Fingerprint presented at 11/04/2006 [15].

Radionuclide	General Building Area			Betalite Store Area		
	Average [Bq/g]	Maximum [Bq/g]	Average [%]	Average [Bq/g]	Maximum [Bq/g]	Average [%]
^{3}H	14.791	226.000	48.734	56.890	100.340	78.524
^{14}C	0.227	0.510	0.748	0.227	0.510	0.314
^{60}Co	0.828	5.299	2.727	0.828	5.299	1.142
^{63}Ni	0.183	0.759	0.602	0.183	0.759	0.252
^{90}Sr	6.395	21.997	21.072	6.395	21.997	8.827
^{137}Cs	4.557	109.651	15.015	4.557	109.651	6.290
^{241}Am	2.219	2.577	7.311	2.219	2.577	3.063
^{210}Pb	0.067	0.159	0.220	0.067	0.159	0.092
^{226}Ra	0.125	0.150	0.412	0.125	0.150	0.173
^{230}Th	0.003	0.004	0.010	0.003	0.004	0.004
^{232}Th	0.003	0.006	0.008	0.003	0.006	0.003
^{234}U	0.373	2.210	1.230	0.373	2.210	0.515
^{235}U	0.118	0.470	0.390	0.118	0.470	0.163
^{238}U	0.461	2.450	1.521	0.461	2.450	0.637
Total	**30.350**	**372.243**	**100.000**	**72.450**	**246.583**	**100.000**

402 Due to the wide range of areas and items from which samples were taken, a number of radionuclides not commonly found in the facility are contained within the resulting fingerprint, mainly in the form of alpha emitters. The potential for Pu isotopes to be included in the general fingerprint has been raised (e.g. in association with the high proportion of ^{241}Am), as observed in the B78 Ventilation System fingerprint [117]. This was considered by comparing ^{241}Am:U ratios in the B78 fingerprint with those derived for the Dragon general building area fingerprint (Table **3.17**), but it can be seen that the ratios are not comparable with differences of two to three orders of magnitude, implying distinctly different fingerprints. If this is ignored and the B78 fingerprint used to scale Pu from ^{241}Am, the estimated ^{241}Pu content in the general area fingerprint would be higher than the ^{3}H content (35.4% ^{241}Pu compared with 30.8% ^{3}H). The ^{241}Pu content is greater than ^{3}H in the B78 fingerprint, but it is unlikely that this would apply to surface contamination throughout the facility, especially in those areas with limited fuel handling. As there are no Pu sample measurements in the characterisation dataset and only two ^{241}Am results, the uncertainty in the considered approach is significant. Applying such a fingerprint across the entire facility would result in a significant Pu inventory that is not justified, especially as, where gross alpha/beta results are available for the characterisation data, there is no indication that a significant proportion of the activity is unaccounted for. Given this, the presence of Pu has not been inferred and it is excluded from the Dragon general building contamination fingerprint at this time (INV-DRAGON-007). See Section 3.5.4 for further analysis and discussion.

Table 3.17: Comparison of ^{241}Am:U ratios in the B78 Ventilation System Fingerprint [117] and the derived Dragon general building surface contamination fingerprint (Table **3.16**). Data presented at 11/04/2006 [15].

	Ratio to ^{241}Am - B78 Fingerprint	Ratio to ^{241}Am - Derived General Building Fingerprint
Radionuclide	**[-]**	**[-]**
^{234}U	0.00510	0.16827
^{235}U	0.00024	0.05336
^{238}U	0.00024	0.20799

ViridiScope Sampling Campaign

403 ViridiScope carried out a sampling campaign of the Dragon building on 8 and 12-15 March 2018, with 147 samples collected and analysed in-situ for alpha, beta and gamma radiation [154, p.13]. The survey involved the use of a remote laser sampling tool that was able to climb the walls and ablate areas of paint for analysis [108, §3.2]. The Remotely Operated Vehicle (ROV) was deployed to areas of higher activity as found in previous surveys at various levels around B70, and took samples from a number of locations, including Wall A inner +36', Wall B outer +36', and hotspots on Wall A inner -35' and Wall B outer -25' [108, §3.2; 154, Tab.1-Tab.6].

404 The highest activity of 100 cps was measured using a DP6 probe for a Wall B outer surface hotspot on the -25' level on 14 March 2018 [154, p.15, Tab.4]. Reference [154, p.15] stated that the gamma spectrum identified the activity to be ^{137}Cs (219 Bq/g), which shows good correlation with the total beta activity measurement of 239 Bq/g. The alpha result was a LOD value of 0.13 Bq/g.

405 As a bounding estimate, the approach applied to calculating the Dragon building surface contamination inventory has been to use the fingerprints derived above with the highest activity patch measured in the ViridiScope survey to calculate a surface activity in Bq/cm^2, which is assumed to apply to a proportion of the entire building surface. Given the low total inventory for Dragon (especially in comparison with SGHWR), it is expected that such a conservative approach will not challenge the PA and environmental safety case, but if a greater margin is required then a more detailed estimate may need to be developed and/or additional characterisation undertaken to use a more realistic inventory (INV-DRAGON-007). Furthermore, the derived general building surface contamination fingerprint contains ~11% actinides, which is considered to be pessimistic, particularly as the ViridiScope survey reported LODs for alpha activity.

Probe Response Correction and Activity Conversion

406 In order to use the 100 cps hot spot activity measured in the ViridiScope survey [154, Tab.4] to calculate an inventory estimate, a detection efficiency correction must be applied to the measured count rate.

407 As the probe used was a DP6, only the detectable beta emitters for this probe are used in the probe response calculation, with the alpha activity inferred from the beta response based upon the relevant fingerprint. A standard calculation spreadsheet with defined

DP6 probe efficiency values was supplied by NRS and has been incorporated into the accompanying Dragon inventory spreadsheet, combined with the derived Dragon building fingerprints [15]. The resulting efficiency calibration for both the general building and Betalite area fingerprints is presented in Table **3.18**.

408 The formula used to equate activity to a probe count rate measurement is specified by Magnox Ltd [155, Annex B]. The probe measurement in the ViridiScope survey was for fixed contamination and so the formula is simplified to the following:

$$Activity\ [Bq/cm^2] = \frac{Direct\ probe\ count\ rate\ [cps]}{Probe\ response\ [cps\ per\ Bq/cm^2]}$$

409 Using the above and the derived probe efficiencies in Table **3.18** for a 100 cps measurement equates to 10.80 Bq/cm^2 assuming the general building contamination fingerprint and 126.96 Bq/cm^2 for the Betalite fingerprint, both at 14/03/2018 [15].

Table 3.18: DP6 probe efficiency assuming the measured activity is of a hotspot with characteristics specified by the derived Dragon general building and Betalite store area surface contamination fingerprints (Table **3.16**). Efficiencies calculated for the 100 cps probe measurement date of 14/03/2018 [15; 154, Tab.4].

		General Building Fingerprint		**Betalite Area Fingerprint**	
Radionuclide	**Probe Efficiency [%]**	**Fingerprint activity scaled for ^{90}Y and strong beta [-]**	**Effective efficiency [%]**	**Fingerprint activity scaled for ^{90}Y and strong beta [-]**	**Effective efficiency [%]**
^{60}Co	11.17	0.0072	0.080	0.0037	0.0003
^{90}Sr	17.51	0.1772	3.102	0.0912	0.283
^{90}Y	21.35	0.1772	3.782	0.0912	0.345
^{106}Ru	0		0		0
^{106}Rh	22.62		0		0
^{125}Sb	6.21		0		0
^{134}Cs	13.49		0		0
^{137}Cs	16.97	0.1352	2.294	0.0696	0.160
^{147}Pm	0		0		0
^{151}Sm	0		0		0
^{152}Eu	8.56		0		0
^{154}Eu	13.49		0		0
^{155}Eu	0.41		0		0
^{14}C	0	0.0094	0	0.0049	0
^{36}Cl	18.580		0		0
^{99}Tc	5.84		0		0
^{35}S	0		0		0
		Total Efficiency:	**9.26**	**Total Efficiency:**	**0.79**

Tritium Ingress Profile

410 As discussed above, the surface sample results of the available Dragon characterisation survey data have been used to derive the general building and Betalite area surface contamination fingerprints. The results for the sub-surface samples in the characterisation survey data have been used to derive an estimate for tritium ingress into Dragon building surfaces.

411 Two of the characterisation datasets discussed above were for core samples that were sub-sampled [136; 144], giving rise to 165 sub-surface concrete samples at various depths. Only tritium was measured in the 44 concrete sub-samples from the 2003 dataset [136], all above the LOD (0.14-44.0 Bq/g at 11/04/2006). In addition to the 120 above-LOD ^{3}H results (0.04-7.20 Bq/g) for the 121 samples in the 2006 dataset [144], the gamma spectroscopy analysis considered $^{152, 154, 155}$Eu, ^{241}Am, $^{134, 137}$Cs, $^{57, 58, 60}$Co and ^{40}K; nine samples reported above-LOD results for ^{137}Cs (0.003-0.032 Bq/g) and 84 for naturally-occurring ^{40}K (0.05-0.42 Bq/g). As ^{137}Cs is relatively mobile and could be present as contamination during sampling and ^{40}K is present naturally, there is limited evidence for contamination ingress into the Dragon building structure surfaces by any radionuclide other than ^{3}H.

412 Table **3.19** summarises the available sub-surface sample characterisation data for ^{3}H. As for the surface samples, the data is also assessed for sub-groups within the characterisation dataset or if it is appropriate to average across all samples. As previously observed, the average inner and outer wall data groups show similar average activities (0.52 Bq/g and 0.45 Bq/g, respectively). The B78 results are lower, but comparable; no sub-surface samples were taken in the vehicle airlock. The results for the Betalite area are, unsurprisingly, higher (average 4.81 Bq/g). Assessment across B70 building depths also shows higher activity (2.27 Bq/g) on the -25’ level (where the Betalite store is located) compared with 0.36-0.65 Bq/g on the other building levels for which data is available. The nature of the two separate groups is illustrated in Figure **3.16**, with the ^{3}H results combined into all areas and the Betalite area, and demonstrating the reduction in activity with increasing depth into the core samples.

Table 3.19: Summary of selected characterisation data considered in derivation of the Dragon tritium ingress estimate [15]. Only the results from sub-surface samples are included in this analysis, with the sample activity presented at a common date of 11/04/2006. The results are grouped according to the Dragon Complex area in which the samples were taken, and by the building level from which they were taken. The actual sample depths have been allocated into depth groups using an approximate best fit.

	^{3}H activity			Average ^{3}H activity at 11/04/2006 of samples at specified depth [Bq/g]																	
	No. >LOD	Av. [Bq/g]	Max. [Bq/g]	No. >LOD	0-5cm	No. >LOD	5-10cm	No. >LOD	10-15cm	No. >LOD	15-30cm	No. >LOD	30-50cm	No. >LOD	50-65cm	No. >LOD	65-100cm	No. >LOD	100-135cm	No. >LOD	5cm bulk
Breakdown by Dragon Complex area																					
Inner wall up to bioshield	56	0.52	1.76	20	0.58	13	0.39	7	0.44	4	0.38	4	0.34	4	0.96	0		0		4	0.71
Outer wall up to the inner wall	72	0.45	1.69	27	0.49	27	0.43	14	0.37	0		0		0		0		0		4	0.54
Betalite store	23	4.81	43.96	9	8.74	5	3.46	3	3.59	1	1.42	0		1	0.04	0		4	0.61	0	
B78 Fuel Store	13	0.07	0.16	6	0.08	5	0.08	2	0.05	0		0		0		0		0		0	
Vehicle Airlock	0			0		0		0		0		0		0		0		0		0	
Breakdown by Dragon building level																					
-25'	56	2.27	43.96	22	3.93	14	1.51	8	1.59	3	0.79	0		1	0.04	0		4	0.61	4	0.49
-25' excl. Betalite store	33	0.50	1.33	13	0.61	9	0.42	5	0.39	2	0.47	0		0		0		0		4	0.49
-18'	0			0		0		0		0		0		0		0		0		0	
0'	46	0.44	1.76	20	0.51	15	0.37	6	0.31	1	0.17	2	0.21	2	1.12	0		0		0	
+18'	25	0.36	0.74	9	0.27	8	0.38	6	0.37	0		0		0		0		0		2	0.71
+36'	24	0.65	1.69	5	0.87	8	0.56	4	0.56	1	0.39	2	0.46	2	0.81	0		0		2	0.82
Vehicle Airlock	0			0		0		0		0		0		0		0		0		0	
B78	13	0.07	0.16	6	0.08	5	0.08	2	0.05	0		0		0		0		0		0	
Samples from all areas (sub-surface samples only)																					
All areas excl. Betalite	141	0.44	1.76	53	0.48	45	0.38	23	0.37	4	0.38	4	0.34	4	0.96	0		0		8	0.63
All areas	164	1.06	43.96	62	1.68	50	0.69	26	0.74	5	0.59	4	0.34	5	0.78	0		4	0.61	8	0.63

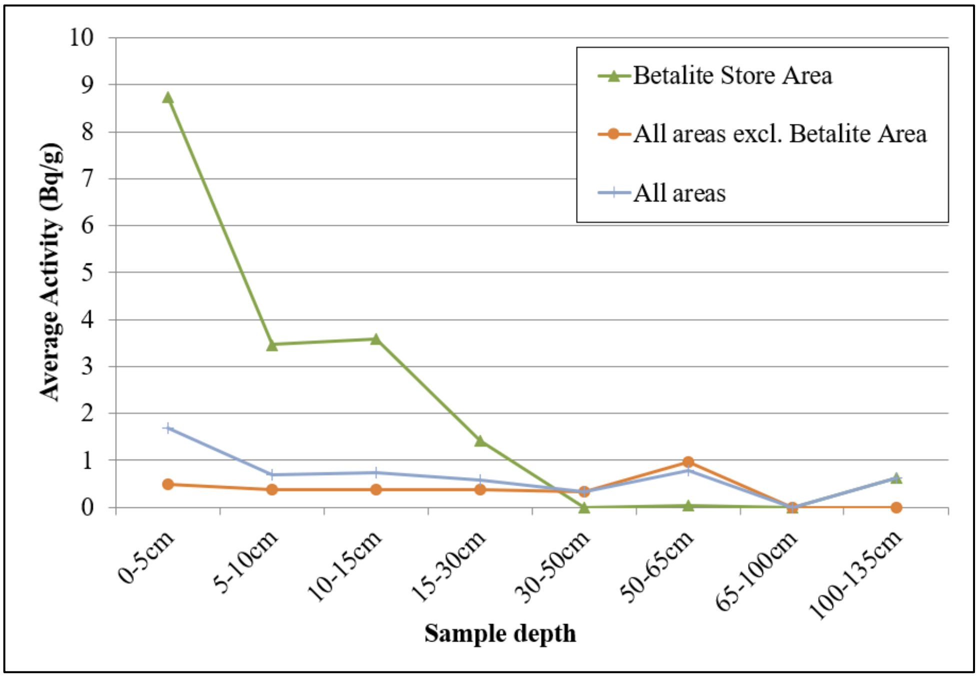


Figure 3.16: Profile of ^{3}H activity results according to sample core depth [15].

413 Figure **3.16** clearly shows that, consistent with the surface samples, the appropriate approach to grouping the sub-surface ^{3}H data is by combining the inner, outer and B78 groups into one set and separately considering the Betalite results to avoid substantial over-estimation of the ^{3}H activity.

414 The ^{3}H activity for the two groups and by depth is summarised in Table **3.20**. Looking at the general building area data, the activity decreases with depth to ~50 cm, but appears to increase in the single data point after this depth. However, the observed activities are less than 1 Bq/g and the measurement uncertainty is significant. For the Betalite area no samples were taken at 30-50 cm, there is one sample at 50-65 cm (0.04 Bq/g), the single sample was reported at the LOD for 65-100 cm, and then four samples were reported for 100-135 cm (0.22-0.82 Bq/g).

415 In estimating the ^{3}H ingress inventory, the depth of ingress to assume across the building must be determined. The calculation applied in this inventory assessment is approximate, simply taking the calculated contaminated building surface area and multiplying this by the depth of contamination, without accounting for the actual surface thickness. Whilst ^{3}H activity data is available at depths to 135 cm, all the walls/floors in the Dragon building are not this thick. Wall D (the bioshield) is the thickest at 1.75 m and Wall B (the next thickest) is 1.3 m thick [15]; however, the majority of other walls/surfaces are assumed to be less than this. Thus, it is proposed that as ^{3}H LOD values begin to be reported after 30 cm thick (for the Betalite area) and most structural walls are assumed to be this thick, that the calculation of ^{3}H ingress into Dragon surfaces should be limited to 30 cm. This is considered to be appropriate for this estimated inventory.

Table 3.20: Dragon building tritium ingress estimate for the general building and Betalite store area using 165 sub-surface characterisation samples [136; 144] for the activity estimates between 0-135 cm, and assuming the surface activity calculated in the general surface contamination fingerprints for the ^{3}H activity in the paint layer (Table **3.16**). Fingerprint presented at 11/04/2006 [15].

Layer	Average General Building Area [Bq/g]	Maximum General Building Area [Bq/g]	Average Betalite Store Area [Bq/g]	Maximum Betalite Store Area [Bq/g]
Paint layer	14.79	226.00	56.89	100.34
0-5cm	0.48	1.69	8.74	43.96
5-10cm	0.38	0.84	3.46	7.30
10-15cm	0.37	0.92	3.59	7.00
15-30cm	0.38	0.67	1.42	1.42
30-50cm	0.34	0.65		
50-65cm	0.96	1.76	0.04	0.04
65-100cm				
100-135cm			0.61	0.82

3.5.3 Inventory Estimate

416 As a bounding estimate, the approach applied to calculating the Dragon building surface contamination inventory has been to use the derived fingerprints with the measured highest activity patch, which is assumed to apply to a proportion of the entire building surface. The applicable surface areas and the resulting total activities are presented in Table **3.22**, calculated using the average activity concentrations presented in Table **3.21**. This is an extremely pessimistic approach as it applies the highest measured 100 cps hotspot surface contamination to the entire Dragon building. Therefore an arbitrary assumption is made that only 5% of the surface activity is present – this assumption is made on the basis that the building does not have any significant contamination (INV-DRAGON-007); see further analysis and discussion in Section 3.5.4.

Table 3.21: Derived Dragon general building and Betalite store area surface contamination fingerprints (Table **3.16**), and average and maximum activity concentrations, decayed to 01/01/2027. ^{227}Ac, ^{229}Th, ^{231}Pa, ^{233}U and ^{237}Np are included in the fingerprint for completeness, as they were included in the modelled decay chains when decaying the data from 11/04/2006 to 01/01/2027, but their impact will remain negligible unless a significantly longer decay period is assumed.

	General Building Area			**Betalite Store Area**		
Radionuclide	**Average [Bq/g]**	**Maximum [Bq/g]**	**Average %**	**Average [Bq/g]**	**Maximum [Bq/g]**	**Average %**
^{3}H	4.609	70.426	30.544	17.728	31.268	62.846
^{14}C	0.227	0.509	1.501	0.227	0.509	0.803
^{60}Co	0.054	0.347	0.360	0.054	0.347	0.192
^{63}Ni	0.158	0.658	1.049	0.158	0.658	0.561
^{90}Sr	3.883	13.356	25.733	3.883	13.356	13.765
^{137}Cs	2.831	68.110	18.758	2.831	68.110	10.034
^{210}Pb	0.094	0.154	0.625	0.094	0.154	0.334
^{226}Ra	0.124	0.149	0.821	0.124	0.149	0.439
^{228}Ra	0.002	0.006	0.015	0.002	0.006	0.008
^{227}Ac	1.39E-05	5.52E-05	9.21E-05	1.39E-05	5.52E-05	4.93E-05
^{228}Th	0.002	0.005	0.015	0.002	0.005	0.008
^{229}Th	4.27E-13	4.96E-13	2.83E-12	4.27E-13	4.96E-13	1.51E-12
^{230}Th	0.003	0.005	0.021	0.003	0.005	0.011
^{232}Th	0.003	0.006	0.017	0.003	0.006	0.009
^{231}Pa	5.19E-05	2.06E-04	3.44E-04	5.19E-05	2.06E-04	1.84E-04
^{233}U	6.57E-10	7.63E-10	4.35E-09	6.57E-10	7.63E-10	2.33E-09
^{234}U	0.373	2.210	2.474	0.373	2.210	1.324
^{235}U	0.118	0.470	0.785	0.118	0.470	0.420
^{238}U	0.461	2.450	3.058	0.461	2.450	1.636
^{237}Np	1.46E-05	1.70E-05	9.69E-05	1.46E-05	1.70E-05	5.18E-05
^{241}Am	2.146	2.493	14.224	2.146	2.493	7.609
Total	15.09	161.35	100.00	28.21	122.20	100.00

Table 3.22: Activity estimate for Dragon building surface contamination using the derived average Dragon general building and Betalite store area surface contamination fingerprints (Table **3.16**) and applying a hot spot equivalent surface activity measurement (9.91 Bq/cm^2 for the general building and 102.58 Bq/cm^2 for the Betalite area) to the applicable building surface area. The activity is calculated assuming the measured hot spot activity applies to the entire building surface area, but is also presented assuming only of 5% the building surface is contaminated. Only the below-ground building structure surface areas are considered, as the above-ground section forms the void backfill and is considered in Section 3.6. Data presented at 14/03/2018 [15].

Component	Surface Area [m^2]	Total Activity [Bq]	5% of Total Activity [Bq]
Below ground B70 general building contamination (excl. Betalite area)	6,200.0	6.144E+08	3.072E+07
Betalite store area building contamination	258.3	2.650E+08	1.325E+07
Total	**6,458.3**	**8.794E+08**	**4.397E+07**

417 To calculate the total ^3H ingress inventory the mass of the paint and concrete layers is required. These have been calculated using the surface areas in Table **3.22**, the paint and concrete density values cited in Table **3.23**, and using the tritium sample depth ranges in Table **3.20**. For each of the sections (e.g. paint layer, 0-5 cm sub-sample, etc.), the ^3H contamination volume was calculated by multiplying the surface area of the exposed surfaces within the building by the depth (up to 30 cm deep) and then multiplying by the average activity.

418 The Dragon surface area value includes the inner and outer bioshield surfaces. As a separate inventory estimate is calculated for the bioshield concrete, the ^3H ingress inventory is not applied to the bioshield concrete to avoid double-counting, but contamination in the paint layer on the bioshield surface is included in the estimate as that is not covered by the bioshield inventory estimate.

419 In using the entire contaminated building structure surface area value to estimate the ^3H inventory, it is implicitly assumed that all surfaces are at least 30 cm thick (if single-sided contamination) and that all surfaces are painted concrete. Whilst this may be correct for the majority, the calculated surface area value does include some metal surfaces (e.g. the metal floor slab) and the 1” steel wall; this is regarded as a conservative assumption used to produce an indicative inventory estimate.

420 The resulting ^3H ingress inventory estimate is presented in Table **3.24** for the original survey date of 11/04/2006 and at 01/01/2027.

Table 3.23: Paint and concrete parameter values used in estimating the ^{3}H ingress inventory.

Parameter	Value	Unit	Source
Paint thickness	0.001	m	Assumption, consistent with discussion in Section 2.9
Paint density	1,500	kg/m^3	Assumption consistent with value adopted for SGHWR (density of waterborne wall paint from [38], adopted in conjunction with an assumed paint thickness of 1 mm; Section 2.9)
Concrete density	2,400	kg/m^3	Assumption consistent with previous PAs and SGHWR (Table **2.6**)

Table 3.24: Estimated Dragon building ^{3}H ingress below-ground in-situ disposal inventory, presented for the characterisation date of 11/04/2006 and for the inventory reference date of 01/01/2027, calculated using the average activity.

	Activity [Bq] at 11/04/2006						**Activity [Bq] at 1/01/2027**
	Paint Layer	**0-5cm**	**5-10cm**	**10-15cm**	**15-30cm**	**Total**	**Total**
Below ground structure (excl. Betalite)	1.376E+08	3.488E+08	2.789E+08	2.676E+08	8.265E+08	1.859E+09	5.793E+08
Betalite area	2.204E+07	2.709E+08	1.072E+08	1.112E+08	1.320E+08	6.434E+08	2.005E+08
Total per layer	1.596E+08	6.197E+08	3.862E+08	3.788E+08	9.585E+08	2.503E+09	7.798E+08

421 Using the approach as set out above and calculated in the Dragon inventory spreadsheet [15], maximum and average activity concentrations and an estimate of the radioactive inventory for the Dragon Reactor building are presented in Table **3.25** (note this combines the average activity surface contamination and ^{3}H ingress inventory estimates). The average activity concentrations presented in Table **3.25** are calculated assuming that the inventory is distributed over the contaminated layer; the tritium ingress part of the inventory is calculated separately as this is over a much greater thickness than the surface contamination. The maximum activity concentrations presented in Table **3.25** are those derived for the general building surface contamination fingerprint; these are used to provide an indication of maximum activity concentration, but noting that they were derived for surface contamination only and exclude the Betalite area. The inventory associated with the below-ground portions of the Dragon building is presented in Table **3.25**, with the above-ground inventory recorded in Section 3.8. The decay calculations have been undertaken using the GoldSim-RT software package [132; 133; 134] and modelling the decay chains as specified in the PA approach report [135, §6].

Table 3.25: Estimated Dragon Reactor building general contamination in-situ disposal inventory, including maximum and average surface activity concentrations from the derived Dragon general building contamination fingerprint and inventory based on average activity concentrations and 5% surface contamination, presented for an inventory reference date of 01/01/2027.

Radio-nuclide	Below ground (excl. Betalite) [MBq]	Betalite area [MBq]	Total Disposal Inventory [MBq]	Average Activity (excl. Betalite) [Bq/g]	Average Activity (Betalite area) [Bq/g]	Maximum Activity [Bq/g]
^{3}H	5.87E+02	2.06E+02	7.93E+02	4.741	18.804	70.426
^{14}C	3.53E-01	7.28E-02	4.26E-01	0.227	0.227	0.509
^{60}Co	8.46E-02	1.74E-02	1.02E-01	0.054	0.054	0.347
^{63}Ni	2.47E-01	5.09E-02	2.98E-01	0.158	0.158	0.658
^{90}Sr	6.05E+00	1.25E+00	7.30E+00	3.883	3.883	13.356
^{137}Cs	4.41E+00	9.10E-01	5.32E+00	2.831	2.831	68.110
^{210}Pb	1.01E-01	2.08E-02	1.22E-01	0.065	0.065	0.154
^{226}Ra	1.93E-01	3.98E-02	2.33E-01	0.124	0.124	0.149
^{228}Ra	2.57E-03	5.29E-04	3.09E-03	0.002	0.002	0.006
^{227}Ac	4.39E-06	9.05E-07	5.30E-06	2.82E-06	2.82E-06	5.52E-05
^{228}Th	1.97E-03	4.06E-04	2.38E-03	0.001	0.001	0.005
^{229}Th	4.93E-14	1.02E-14	5.94E-14	0.000	0.000	4.96E-13
^{230}Th	4.92E-03	1.01E-03	5.93E-03	0.003	0.003	0.005
^{232}Th	3.92E-03	8.09E-04	4.73E-03	0.003	0.003	0.006
^{231}Pa	3.44E-05	7.08E-06	4.14E-05	2.20E-05	2.20E-05	2.06E-04
^{233}U	1.80E-10	3.71E-11	2.17E-10	1.15E-10	1.15E-10	7.63E-10
^{234}U	5.82E-01	1.20E-01	7.02E-01	0.373	0.373	2.210
^{235}U	1.85E-01	3.81E-02	2.23E-01	0.118	0.118	0.470
^{238}U	7.20E-01	1.48E-01	8.68E-01	0.461	0.461	2.450
^{237}Np	9.59E-06	1.98E-06	1.16E-05	6.15E-06	6.15E-06	1.70E-05
^{241}Am	3.35E+00	6.90E-01	4.04E+00	2.146	2.146	2.493
Total	**6.03E+02**	**2.10E+02**	**8.12E+02**	**1.52E+01**	**2.93E+01**	**1.61E+02**

3.5.4 Sensitivity Analysis and Further Characterisation

422 Four key areas of uncertainty remain within the inventory estimate for general contamination of the Dragon Reactor Building:

- The representativeness of the surface contamination characterisation dataset and extent of surface contamination (INV-DRAGON-004; INV-DRAGON-007).
- The representativeness of the ^{3}H ingress contamination dataset (INV-DRAGON-004).

- The potential for the presence of Pu isotopes (INV-DRAGON-007).
- The potential for very high tritium contamination within the Betalite store area (INV-DRAGON-007).

423 Each of these can be addressed by a sensitivity analysis, as discussed in the following sub-sections.

Data Representativeness and Extent of Contamination

424 Although there are a large number of samples underpinning the fingerprint used, a DQO process was not followed in their collection and so there remains uncertainty about the representativeness of the underlying dataset. Additionally, the assumption that only 5% of surface activity is present (introduced to reflect anecdotal understanding of the contamination level and to counteract the pessimistic use of the highest measured hotspot to scale the fingerprint) is not underpinned.

425 The uncertainty relating to surface contamination can be accounted for by assessing the impact of assuming that 100% of surface contamination, calculated using the highest measured hotspot activity, is present. Such a scenario is believed to be extremely pessimistic, since the building is not known to have any significant contamination, and can therefore be considered to cover all residual uncertainty relating to general data representativeness and contamination levels.

426 The uncertainty relating to ^{3}H ingress can be accounted for by calculating the inventory using the maximum rather than average activity concentrations, in the same way as was applied in the bioshield sensitivity analysis.

427 Table **3.26** presents the alternative inventory calculated in this way.

Table 3.26: Alternative Dragon Reactor (B70) Building general contamination inventory, calculated using the alternative assumption that 100% (rather than 5%) of surface contamination is present, and using maximum rather than average activity concentrations to calculate the ^{3}H ingress inventory, presented for an inventory reference date of 01/01/2027.

Radionuclide	Below cutline (excl. Betalite) [MBq]	Betalite area [MBq]	Total Disposal Inventory [MBq]
^{3}H	2.04E+03	7.30E+02	2.77E+03
^{14}C	7.07E+00	1.46E+00	8.52E+00
^{60}Co	1.69E+00	3.49E-01	2.04E+00
^{63}Ni	4.94E+00	1.02E+00	5.95E+00
^{90}Sr	1.21E+02	2.50E+01	1.46E+02
^{137}Cs	8.83E+01	1.82E+01	1.06E+02
^{210}Pb	2.02E+00	4.16E-01	2.44E+00
^{226}Ra	3.86E+00	7.96E-01	4.66E+00
^{228}Ra	5.13E-02	1.06E-02	6.19E-02
^{227}Ac	8.78E-05	1.81E-05	1.06E-04
^{228}Th	3.94E-02	8.12E-03	4.75E-02
^{229}Th	9.85E-13	2.03E-13	1.19E-12

Radionuclide	Below cutline (excl. Betalite) [MBq]	Betalite area [MBq]	Total Disposal Inventory [MBq]
^{230}Th	9.84E-02	2.03E-02	1.19E-01
^{232}Th	7.85E-02	1.62E-02	9.47E-02
^{231}Pa	6.87E-04	1.42E-04	8.29E-04
^{233}U	3.60E-09	7.41E-10	4.34E-09
^{234}U	1.16E+01	2.40E+00	1.40E+01
^{235}U	3.69E+00	7.61E-01	4.45E+00
^{238}U	1.44E+01	2.97E+00	1.74E+01
^{237}Np	1.92E-04	3.95E-05	2.31E-04
^{241}Am	6.69E+01	1.38E+01	8.07E+01
Total	**2.37E+03**	**7.97E+02**	**3.17E+03**

428 Comparing the values in Table **3.26** to those in Table **3.25**, it can be seen that the total inventories (MBq) for each component are approximately four times higher in the alternative inventory than in the reference inventory estimate. Assuming the equivalent alternative inventories for B70 and B78 (as they are based on the same datasets – see further discussion in Section 3.7) but no increase in the inventory of any of the other Dragon features, the inventory and proportional contribution of each component of the Dragon Reactor building to the overall Dragon inventory would change as shown in Table **3.27**.

Table 3.27: Contributions of Dragon Reactor (B70) Building components to overall Dragon inventory in the alternative inventory estimate compared to the reference inventory estimate (based on activities at 01/01/2027).

Component of Dragon Reactor (B70) Building contamination	Reference inventory estimate (5% surface contamination present and ^{3}H ingress calculated using average activity concentration)	Alternative (100% surface contamination present and ^{3}H ingress calculated using maximum activity concentration)
	MBq and % contribution to overall Dragon inventory	
General area – surface contamination	2.35E+01 (0.3%) in-situ; 3.95E+01 (0.5%) as backfill	4.70E+02 (3.5%) in-situ; 7.90E+02 (5.8%) as backfill
Betalite store – surface contamination	9.06E+00 (0.1%) in-situ	1.81E+02 (1.3%) in-situ
General area – ^{3}H ingress	5.79E+02 (8.0%) in-situ; 9.83E+03 (13.6%) as backfill	1.90E+03 (14.0%) in-situ; 3.21E+03 (23.6%) as backfill
Betalite store – ^{3}H ingress	2.00E+02 (2.8%) in-situ	6.16E+02 (4.5%) in-situ
Total	**8.12E+02 (11.2%) in-situ; 1.02E+03 (14.1%) as backfill**	**3.17E+03 (23.3%) in-situ; 4.00E+03 (29.4%) as backfill**

Potential Presence of Pu Isotopes

429 Section 3.5.2 (paragraph 402) discusses the potential for Pu isotopes to be included in the general fingerprint and a possible approach for doing so using the B78 fingerprint. This is subject to significant uncertainty and, since it results in a higher ^{241}Pu content than ^{3}H (which is not credible for the general Dragon building area), would result in a significant Pu inventory that is not justified, even in a bounding case.

430 The recently updated Dragon General Area fingerprint FP-002 [138] also contains Pu isotopes because they are present in smear samples from the B78 plenum chamber and venting, from the non-retractable viewing gallery and from the internal charge machine. Whilst this fingerprint is based on removed items, it shows that Pu contamination was present in Dragon and therefore cannot be ruled out in the residual building contamination, although there is no direct evidence for this. FP-002 offers an alternative approach for including Pu isotopes in the Dragon Reactor Building inventory. Using it to create a combined fingerprint results in the same issues as the approach using the B78 fingerprint; instead, FP-002 can be used in its entirety to calculate an alternative inventory including Pu, presented in Table **3.28**. Although FP-002 is based largely on items that will be removed (many non-concrete) and so is not suitable for the reference inventory estimate, it does give a more realistic indication of how much Pu may be present than any other available approach in this sensitivity analysis. Note that, because the inventory is still scaled to the highest hotspot activity (assuming 5% of surface contamination is present), the total MBq changes only very slightly compared to the reference inventory estimate.

Table 3.28: Alternative Dragon Reactor Building general contamination inventory, with that for the non-Betalite area calculated using the updated FP-002 fingerprint [138], presented for an inventory reference date of 01/01/2027.

Radio-nuclide	General area (exc. Betalite)				Betalite area	Total Disposal Inventory [MBq]
	Total activity [MBq]	Average % (Finger-print)	Average activity concentration [Bq/g]	Maximum activity concentration [Bq/g]	Total Activity [MBq]	
^{3}H	5.85E+02	26.270	4.03E+00	4.65E+01	2.06E+02	7.92E+02
^{14}C	5.72E-01	2.473	3.67E-01	4.38E+00	7.28E-02	6.45E-01
^{36}Cl	8.23E-01	3.556	5.28E-01	6.29E+00	0.00E+00	8.23E-01
^{55}Fe	4.62E-03	0.020	2.96E-03	3.53E-02		4.62E-03
^{60}Co	5.55E-02	0.240	3.56E-02	4.25E-01	1.74E-02	7.29E-02
^{63}Ni	1.47E+00	6.361	9.44E-01	1.13E+01	5.09E-02	1.52E+00
^{90}Sr	3.28E+00	14.175	2.10E+00	2.51E+01	1.25E+00	4.53E+00
^{137}Cs	8.76E+00	37.875	5.62E+00	6.70E+01	9.10E-01	9.67E+00
^{152}Eu	6.50E-01	2.808	4.17E-01	4.97E+00		6.50E-01
^{154}Eu	6.13E-02	0.265	3.93E-02	4.69E-01		6.13E-02
^{210}Pb	3.50E-10	0.000	2.25E-10	3.21E-08	2.08E-02	2.08E-02
^{226}Ra	4.11E-09	0.000	2.64E-09	1.74E-07	3.98E-02	3.98E-02

Radio-nuclide	General area (exc. Betalite)				Betalite area	Total Disposal Inventory [MBq]
	Total activity [MBq]	Average % (Finger-print)	Average activity concentration [Bq/g]	Maximum activity concentration [Bq/g]	Total Activity [MBq]	
^{228}Ra	4.53E-02	0.275	2.91E-02	4.87E-01	5.29E-04	4.58E-02
^{227}Ac	8.97E-07	0.000	5.75E-07	3.39E-05	9.05E-07	1.80E-06
^{228}Th	3.48E-02	0.263	2.23E-02	4.65E-01	4.06E-04	3.52E-02
^{229}Th	1.76E-15	0.000	1.13E-15	1.99E-13	1.02E-14	1.19E-14
^{230}Th	2.16E-06	0.000	1.38E-06	3.89E-05	1.01E-03	1.02E-03
^{232}Th	6.93E-02	0.299	4.44E-02	5.30E-01	8.09E-04	7.01E-02
^{231}Pa	7.02E-06	0.000	4.50E-06	1.26E-04	7.08E-06	1.41E-05
^{233}U	6.50E-12	0.000	4.17E-12	3.15E-10	3.71E-11	4.35E-11
^{234}U	2.67E-02	0.115	1.71E-02	2.04E-01	1.20E-01	1.47E-01
^{235}U	3.77E-02	0.163	2.42E-02	2.89E-01	3.81E-02	7.58E-02
^{238}U	4.68E-02	0.202	3.00E-02	3.58E-01	1.48E-01	1.95E-01
^{238}Pu	5.96E-02	0.258	3.82E-02	4.56E-01		5.96E-02
^{239}Pu	2.53E-02	0.110	1.63E-02	1.94E-01		2.53E-02
^{240}Pu	1.87E-02	0.081	1.20E-02	1.43E-01		1.87E-02
^{241}Pu	8.09E-01	3.496	5.19E-01	6.19E+00		8.09E-01
^{237}Np	3.53E-07	0.000	2.27E-07	7.40E-06	1.98E-06	2.33E-06
^{241}Am	1.30E-01	0.696	8.33E-02	1.23E+00	6.90E-01	8.20E-01
Total	**6.02E+02**	**100.00**	**1.49E+01**	**1.77E+02**	**2.10E+02**	**8.12E+02**

Potential for High Tritium in Betalite Store Area

431 In the reference inventory estimate, the single anomalously high ^{3}H result of 6,600 Bq/g from a paint layer sample from the Betalite store area is excluded from the fingerprint, as discussed and justified in Section 3.5.2. Subsequent discussion with site personnel indicated that there may have been a credible reason for localised very high tritium contamination in one area (for example, a leak on one wall due to the lids of the Betalites stored against it not being sealed); however, this anecdotal evidence is highly uncertain and not underpinned. Therefore, no change was made to the reference inventory estimate, but the possibility can be accounted for by the calculation of an alternative inventory including the anomalous result in the fingerprint. This change only affects the Betalite store area.

432 Including the very high ^{3}H sample increases the paint layer average for the Betalite store area from 56.9 Bq/g to 1,366 Bq/g, and the total activity of the paint layer across the Betalite store area from 2.20E+01 MBq to 5.29E+02 MBq (all at a date of 11/04/2006). Assuming no increase in the inventory of any of the other Dragon features, the inventory and proportional contribution of the Betalite store area to the overall Dragon inventory would change as shown in Table **3.29**.

433 If elevated tritium contamination were shown to be present in the Betalite store during future decommissioning, there would be options to either include it in the disposal or

to decontaminate or remove the worst-affected concrete. Optimisation would be undertaken at this stage to determine the best management route.

Table 3.29: Contributions of the Betalite store area to the overall Dragon inventory for low-^{3}H (excluding the anomalously high result; as per the reference inventory estimate) and high-^{3}H fingerprints (including it), based on activities at 01/01/2027.

Component of Betalite store area contamination	Reference inventory estimate (5% surface contamination present and ^{3}H ingress calculated using average activity concentration)	Alternative (100% surface contamination present and ^{3}H ingress calculated using maximum activity concentration)
	MBq and % contribution to overall Dragon inventory	
Surface contam. (low ^{3}H FP)	9.06E+00 (0.1%)	1.81E+02 (1.3%)
Surface contam. (high ^{3}H FP)	1.26E+02 (1.7%)	2.53E+03 (15.1%)
^{3}H ingress (low ^{3}H FP)	2.00E+02 (2.8%)	6.16E+02 (4.5%)
^{3}H ingress (high ^{3}H FP)	3.58E+02 (4.8%)	1.40E+03 (8.4%)
All (low ^{3}H FP)	**2.10E+02 (2.9%)**	**7.97E+02 (5.9%)**
All (high ^{3}H FP)	**4.85E+02 (6.5%)**	**3.93E+03 (23.5%)**

Further Characterisation

434 Further sample collection and analysis for the purpose of better constraining the Dragon Reactor Building inventory is not expected in the near term since a large number of samples have already been collected. Further characterisation may be undertaken as decommissioning proceeds, but this will be after the inventory freeze for assessments in support of the permit application.

435 However, it is considered that the sensitivity analyses described in this section adequately cover the remaining uncertainties, which will be further captured via uncertainty analysis as part of PA. The alternative inventories explored in this section, all of which are considered to be pessimistic, do not result in significant changes to the overall Dragon inventory (i.e. increasing the order of magnitude), or in activity concentration values that come close to the upper limit for UK LLW.

3.6 Residual Contamination from the Dragon Reactor Building (B70) Purge Gas Pre-Cooler Contaminated Water Spill

3.6.1 Feature Description

436 The PGPC's function was to cool the purge gases from the reactor core to approximately 100°C [156]. This was achieved by passing the gas over three interconnected cooling coils through which clean helium was circulated in a closed loop system. The PGPC consists of an outer shell of carbon steel and various internal components of stainless steel. The individual sampling lines (each with an outer diameter of a quarter of an inch) carried a very low flow and were cooled by their proximity to the outer shell.

437 As noted in Section 3.4.1, the PGPC was removed from its in-situ position (connecting the lower section of the RPV and the cathedral area at the -16’ level) in January 2018; however, it remained within the Dragon Reactor Building. During a lifting operation in the cathedral area on the 22 March 2021 to transfer the PGPC into a bespoke shielded container, contaminated water spilled from it onto the concrete floor at the base of the B70 reactor building (-25’ level) [115]. Characterisation and clean-up of the spill is currently ongoing.

438 It is currently intended that the contaminated concrete resulting from the spill will be decontaminated to 200 Bq/g [116], a level consistent with the optimisation threshold in the End State EAC. However, it is not clear whether this will be possible and so, to bound the impact of incomplete removal, an estimate has been derived for residual contamination that could remain on site. As the PGPC and its contents had been due to be fully removed off site, residual contamination from this contaminated water spill represents an additional inventory to that calculated in Section 3.4.4. The area in which the spill occurred is entirely below ground level, so any contamination remaining after clean-up needs to be included in the inventory that is anticipated to be left in-situ at the site end state.

3.6.2 Origin and Constraints on the Radiological Inventory

439 As characterisation and clean-up are ongoing, the inventory presented below is a first estimate made prior to the availability of comprehensive data. It is based on a number of assumptions and is subject to significant uncertainty (INV-DRAGON-010). However, it is believed to be a conservative inventory estimate, which will be refined by further characterisation data.

440 The surface area of the spill is estimated to be approximately 3.3 m^2 [115]. The total activity of the spill was estimated via MicroShield dose modelling using two different approaches: 1) treating the spill area as one large circular source; 2) modelling the spill as the sum of two separate sources, recognising the presence of the PGPC itself as well as the non-uniformity of the dose rates [115]. Neither model accounts for the dose contribution of the PGPC, which is conservative with regard to the characterisation of the spill since any inventory remaining in the PGPC will be removed as originally intended. This modelling gives an estimated range for the activity currently remaining in the contaminated floor region of between 14 GBq (Model 1) and 24 GBq (Model 2).

441 A smear sample taken from the PGPC shows contamination results that are closely correlated with the Dragon primary coolant fingerprint [156]. This is to be expected, since the items which were sampled during derivation of the primary coolant fingerprint were exposed to a similar contamination pathway as the PGPC in terms of the passing gases [156]. The primary coolant fingerprint, presented in Table **3.30**, is therefore considered to be representative of the PGPC spill.

Table 3.30: Dragon primary coolant fingerprint as originally derived and decayed to the date of the PGPC spill [156, Tab.4].

Radionuclide	% at 20/06/2012 [157]	% at 22/03/2021 (date of spill)
^{63}Ni	0.013	0.015
^{14}C	0.04	0.049
^{60}Co	0.031	0.012
^{137}Cs	98	98.38
^{3}H	1.27	0.953
^{90}Sr	0.591	0.588
Total	**99.9**	**100.0**

3.6.3 Inventory Estimate

442 The following assumptions (captured in INV-DRAGON-010) have been made in deriving the inventory estimate presented in Table **3.31** below:

- The total activity currently remaining in the contaminated floor region is assumed to be 24 GBq, the upper end of the range given in [115].
- Contamination is assumed to penetrate a depth of 10 mm into the concrete floor. Using the estimated spill area of 3.3 m^2, this gives an assumed contaminated concrete volume of 0.033 m^3. These assumptions only affect the activity concentration, not the total activity.
- The density of concrete is assumed to be 2400 kg/m^3, as for the rest of the Dragon Reactor Building. Using the assumed volume noted above, this gives a contaminated concrete mass of 79.2 kg. This assumption only affects the activity concentration, not the total activity.
- Although it is intended to decontaminate the contaminated concrete resulting from the spill to 200 Bq/g [116], the actual extent of clean-up that will be carried out is not currently known and is likely to depend on health physics requirements in addition to the results of PA modelling. Inventories are presented below for four cases: (1) that no decontamination is undertaken (extremely unlikely), that (2) 90% and (3) 99% of contamination is removed, and (4) that decontamination is undertaken to reduce the activity concentration to the upper limit of LLW (this corresponds to the removal of 95.5% of the contamination).

Table 3.31: Estimated additional Dragon building in-situ disposal inventory associated with residual contamination from the PGPC contaminated water spill, presented for an inventory reference date of 01/01/2027.

Radio-nuclide	No decontamination		90% of contamination removed		99% of contamination removed		Decontaminated to upper LLW limit (95.5% removed)	
	MBq	Bq/g	MBq	Bq/g	MBq	Bq/g	MBq	Bq/g
^{63}Ni	3.46E+00	4.37E+01	3.46E-01	4.37E+00	3.46E-02	4.37E-01	1.57E-01	1.98E+00
^{14}C	1.18E+01	1.49E+02	1.18E+00	1.49E+01	1.18E-01	1.49E+00	5.33E-01	6.73E+00
^{60}Co	1.35E+00	1.71E+01	1.35E-01	1.71E+00	1.35E-02	1.71E-01	6.12E-02	7.73E-01
^{137}Cs	2.07E+04	2.61E+05	2.07E+03	2.61E+04	2.07E+02	2.61E+03	9.37E+02	1.18E+04
^{3}H	1.65E+02	2.09E+03	1.65E+01	2.09E+02	1.65E+00	2.09E+01	7.48E+00	9.45E+01
^{90}Sr	1.23E+02	1.55E+03	1.23E+01	1.55E+02	1.23E+00	1.55E+01	5.56E+00	7.02E+01
Total	**2.10E+04**	**2.65E+05**	**2.10E+03**	**2.65E+04**	**2.10E+02**	**2.65E+03**	**9.50E+02**	**1.20E+04**

3.6.4 Sensitivity Analysis and Further Characterisation

443 As discussed above, there is substantial uncertainty surrounding the inventory of residual contamination from the PGPC contaminated water spill. Data collected during the ongoing clean-up is expected to reduce this significantly in the future. In general, conservative assumptions have been used to derive this initial estimate; however, an assumption that 99% of the contamination currently present will be removed during clean-up may not be bounding.

444 As shown in Table **3.31**, if a more cautious assumption that only 90% clean-up is achieved is taken forward, both the total activity and activity concentration of this feature would be an order of magnitude higher. It would have a much greater impact on the overall Dragon inventory (constituting approximately 25% compared to 3% for 99% clean-up), and its activity concentration (equivalent to 27 GBq/tonne) would then be above the upper limit for LLW in the UK (4 GBq/tonne alpha and 12 GBq/tonne beta/gamma) and more than two orders of magnitude higher than any other Dragon feature.

445 This sensitivity analysis shows that, based on current information, there will be a need to ensure that at least 95.5% clean-up is achieved, since NRS does not intend to dispose of ILW on site. Therefore, the 95.5% decontamination case has been taken forward as the reference inventory for inclusion in the total Winfrith end state radiological inventory, and hence in summary figures and tables in this report. However, additional data collected during clean-up may provide sufficient confidence to reduce the total activity estimate and hence decrease the level of required decontamination.

446 Because it is assumed that this feature will be decontaminated to the upper LLW limit and this will be confirmed by survey data, no alternative inventory is calculated. Similarly, because samples are yet to be taken, the maximum activity concentrations of the PGPC spill reported in subsequent sections are the same as the average activity concentration values, calculated as described above.

3.7 Dragon Fuel Storage Building (B78) General Contamination

3.7.1 Feature Description

447 As described in Section 3.1.2 and illustrated in Figure **3.17**, the B78 building is connected to the B70 building by a contiguous floor slab into which steel rail tracks are set. Rooms 312 and 321 in Figure **3.17** are assumed to correspond to the B70 vehicle airlock, and general contamination in them (including the floor) is included in the inventory estimate for B70 (Section 3.5). The mortuary hole structures are considered separately (Section 3.9) and are not included in the scope of discussions in this section.

448 Although some rooms in B78 are believed to be clean, there is limited evidence to support this, and it is conservatively assumed for the purpose of this inventory estimate that the entire building is contaminated (INV-DRAGON-004). External walls are assumed to be contaminated on the inside only, while internal walls are assumed to be contaminated on both sides.

449 The total surface area within B78 that could potentially be contaminated is needed in order to calculate an inventory. To estimate this, width and length dimensions were measured directly from engineering drawings [158] and cross-checked against dimensions taken from laser images (as supplied by NRS); where discrepancies occurred, an intermediate value was adopted. All height dimensions were taken from the laser images. Owing to lack of clarity in the drawings and laser images, and to reduce the number of calculations, a simplified pattern of internal walls was adopted (INV-DRAGON-011), which is assumed to be sufficiently accurate for the purpose of this inventory estimate (and overall a slight over-estimate, which is conservative). Full details of the surface area calculations and assumptions are included in the accompanying Dragon inventory spreadsheet [15]. Table **3.32** presents the calculated surface areas.

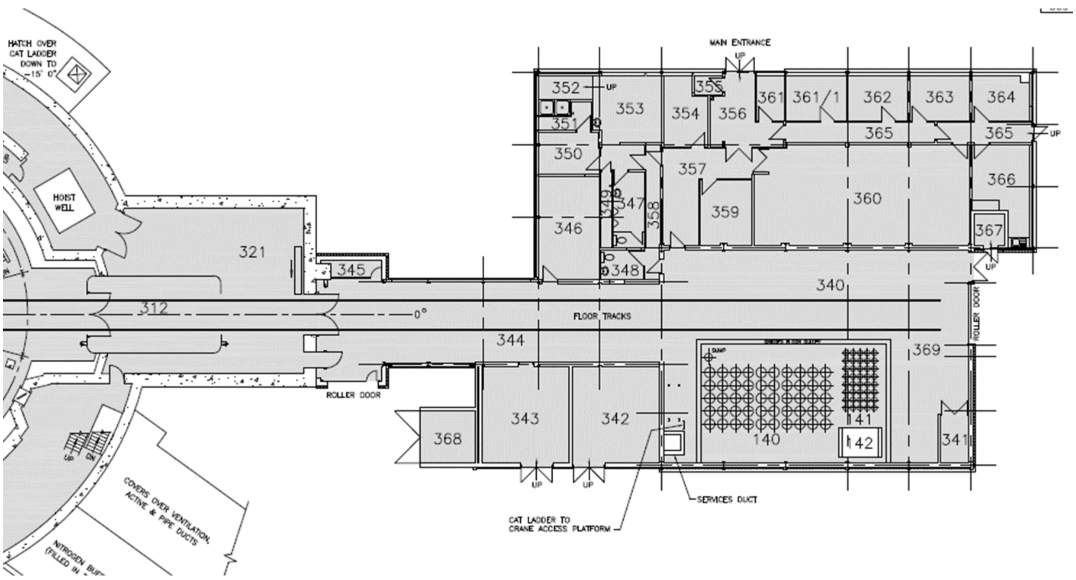


Figure 3.17: Plan view of the B78 building and its connection to the Dragon Reactor Building (B70) [extract from 158]. The primary and secondary mortuary hole structures are shown within B78, labelled 140 and 141 respectively.

Table 3.32: Calculated surface areas within the B78 that could potentially be contaminated.

Component of B78	**Surface area (m^2)**
Floor slab (to remain in-situ)	710
Above cutline (to be demolished and used in B70 backfill)	3368

3.7.2 Origin and Constraints on Radiological Inventory

450 No additional data (i.e. beyond that already reported in Section 3.5) relating to the fingerprint or contamination level in B78 are available. The existing dataset for general Dragon Reactor Building (B70) contamination (Section 3.5.2) already includes samples from B78, and as explained therein, ratios between ^{3}H, ^{137}Cs and ^{60}Co suggest that there is little difference between sample groups from B78, inner B70, outer B70 and the vehicle airlock. Therefore, it is considered appropriate to use the same fingerprint for B78 as for the B70 general building contamination (presented at 11/04/2006 in Table **3.16** and at 01/01/2027 in Table **3.21**).

451 The 2018 ViridiScope sampling survey [154], as used to determine the Dragon Reactor general building contamination level, was restricted to B70. Limited B78 sample data (listed in Section 3.5.2 with full details in the spreadsheet [15]) exist from between 1999 and 2006. The 1999 and 2000 samples show non-trivial contamination levels, the majority coming from ^{90}Sr and ^{137}Cs, some of which will have decayed since the samples were taken. The 2006 samples show minimal contamination. Given that the only sample data available is not recent, it is considered preferable to use the same approach as for B70, that is, based on the ViridiScope survey, using the highest count rate (100 cps) and a probe efficiency calibration to calculate a hot spot equivalent surface activity of 9.91 Bq/cm^2, and assuming that only 5% of the total surface activity calculated using this value is present (with other values used to calculate alternative inventories in Section 3.7.4). The sample data suggest that this approach is conservative.

452 As for B70, it is assumed that tritium has ingressed into the concrete structures of B78. No additional B78-specific information is available; therefore, the B70 general building (excluding the Betalite store area) tritium ingress profile is applied (Table **3.20**; this does include some samples from B78). Engineering drawings [158] suggest that walls in B78 are no more than 30 cm thick. To avoid double counting (as the majority of the walls are assumed to be contaminated from both sides), only ingress up to 15 cm depth is considered.

3.7.3 Inventory Estimate

453 Using the approach as set out above and as calculated in the Dragon inventory spreadsheet [15], maximum and average activity concentrations and an estimate of the radioactive inventory for the Dragon Fuel Storage building are presented in Table **3.33** (note this combines the average activity surface contamination and ^{3}H ingress inventory estimates). The average activity concentrations presented in Table **3.33** are calculated assuming that the inventory is distributed over the contaminated layer; this is calculated separately for the tritium ingress part of the inventory as this is over a much greater thickness than the surface contamination. The maximum activity concentrations presented in Table **3.33** are those derived for the general B70 building surface

contamination fingerprint; these are used to provide an indication of activity concentration but note that they were derived for surface contamination only. The inventory associated with the below-ground components of the B78 building (i.e. the floor slab) is presented in Table **3.33**, with the above-ground inventory recorded in Section 3.8. As for the B70 building general contamination, the decay calculations have been undertaken using the GoldSim-RT software package [132; 133; 134] and modelling the decay chains as specified in the PA approach report [135, §6].

Table 3.33: Estimated Dragon Fuel Storage building general contamination in-situ disposal inventory, including maximum and average surface activity concentrations from the derived B70 general building contamination fingerprint and inventory based on average activity concentrations, presented for an inventory reference date of 01/01/2027.

Radionuclide	B78 Floor Slab Total Activity [MBq]	Average Activity [Bq/g]	Maximum Activity [Bq/g]
^{3}H	3.82E+01	4.755	70.426
^{14}C	4.04E-02	0.227	0.509
^{60}Co	9.68E-03	0.054	0.347
^{63}Ni	2.82E-02	0.158	0.658
^{90}Sr	6.93E-01	3.883	13.356
^{137}Cs	5.05E-01	2.831	68.110
^{210}Pb	1.16E-02	0.065	0.154
^{226}Ra	2.21E-02	0.124	0.149
^{228}Ra	2.94E-04	0.002	0.006
^{227}Ac	5.03E-07	2.82E-06	5.52E-05
^{228}Th	2.25E-04	0.001	0.005
^{229}Th	5.64E-15	0.000	4.96E-13
^{230}Th	5.63E-04	0.003	0.005
^{232}Th	4.49E-04	0.003	0.006
^{231}Pa	3.93E-06	2.20E-05	2.06E-04
^{233}U	2.06E-11	1.15E-10	7.63E-10
^{234}U	6.66E-02	0.373	2.210
^{235}U	2.11E-02	0.118	0.470
^{238}U	8.23E-02	0.461	2.450
^{237}Np	1.10E-06	6.15E-06	1.70E-05
^{241}Am	3.83E-01	2.146	2.493
Total	**4.01E+01**	**1.52E+01**	**1.61E+02**

3.7.4 Sensitivity Analysis and Further Characterisation

454 As the inventory estimate for the B78 building is based on the same dataset as for the B70 reactor building, the same uncertainties (except for those relating to the Betalite store) apply. These are:

- The representativeness of the surface contamination characterisation dataset and extent of surface contamination (INV-DRAGON-004; INV-DRAGON-007).
- The representativeness of the ^{3}H ingress contamination dataset (INV-DRAGON-004).
- The potential for the presence of Pu isotopes (INV-DRAGON-007).

455 Further discussion on each of these uncertainties can be found in Section 3.5.4. There is additional uncertainty regarding the applicability of the dataset to B78, as the ViridiScope sampling survey used to determine contamination level did not include B78 (INV-DRAGON-004); however, as discussed above, this approach is believed to be conservative.

456 Table **3.37** presents equivalent alternative inventories for the B78 floor slab to those calculated for the B70 building: one assuming that 100% of surface contamination is present and using maximum rather than average concentrations to calculate ^{3}H ingress; and another using the Pu-containing fingerprint.

Table 3.34: Alternative B78 floor slab inventories: (i) calculated using the alternative assumption that 100% (rather than 5%) of surface contamination is present, and using maximum rather than average activity concentrations to calculate the ^{3}H ingress inventory, and (ii) using the Pu-containing FP-002 fingerprint [138]. All activities are presented for an inventory reference date of 01/01/2027.

Radionuclide	Alternative inventory (i) (100% surface contamination present; maximum ^{3}H ingress) [MBq]	Alternative inventory (ii) (Pu-containing fingerprint) [MBq]
^{3}H	1.83E+02	3.81E+01
^{14}C	8.09E-01	6.55E-02
^{36}Cl	-	9.41E-02
^{55}Fe	-	5.28E-04
^{60}Co	1.94E-01	6.35E-03
^{63}Ni	5.65E-01	1.68E-01
^{90}Sr	1.39E+01	3.75E-01
^{137}Cs	1.01E+01	1.00E+00
^{152}Eu	-	7.44E-02
^{154}Eu	-	7.01E-03
^{210}Pb	2.31E-01	4.01E-11
^{226}Ra	4.42E-01	4.70E-10
^{228}Ra	5.87E-03	5.18E-03
^{227}Ac	1.01E-05	1.03E-07
^{228}Th	4.51E-03	3.98E-03
^{229}Th	1.13E-13	2.02E-16
^{230}Th	1.13E-02	2.47E-07
^{232}Th	8.98E-03	7.93E-03

Radionuclide	Alternative inventory (i) (100% surface contamination present; maximum ^{3}H ingress) [MBq]	Alternative inventory (ii) (Pu-containing fingerprint) [MBq]
^{231}Pa	7.86E-05	8.03E-07
^{233}U	4.11E-10	7.44E-13
^{234}U	1.33E+00	3.05E-03
^{235}U	4.23E-01	4.32E-03
^{238}U	1.65E+00	5.36E-03
^{238}Pu	-	6.82E-03
^{239}Pu	-	2.90E-03
^{240}Pu	-	2.14E-03
^{241}Pu	-	9.25E-02
^{237}Np	2.19E-05	4.05E-08
^{241}Am	7.66E+00	1.49E-02
Total	**2.20E+02**	**4.00E+01**

457 It can be seen from comparing the total activity values in Table **3.33** and Table **3.34** that the first alternative inventory is 5.5 times higher than the reference inventory estimate. Assuming the equivalent alternative inventories for B70 and B78 (as they are based on the same datasets) but no increase in the inventory of any of the other Dragon features, the inventory and proportional contribution of each component of the B78 building to the overall Dragon inventory would change as shown in Table **3.35**.

Table 3.35: Contributions of Dragon Fuel Store (B78) Building components to overall Dragon inventory in the alternative inventory estimate compared to the reference inventory estimate (based on activities at 01/01/2027).

Component of B78 Building contamination	Reference inventory estimate (5% surface contamination present and ^{3}H ingress calculated using average activity concentration)	Alternative (100% surface contamination present and ^{3}H ingress calculated using maximum activity concentration)
	MBq and % contribution to overall Dragon inventory	
Surface contamination	2.69E+00 (0.04%) in-situ; 1.28E+01 (0.18%) as backfill	5.37E+01 (0.40%) in-situ; 2.55E+02 (1.88%) as backfill
^{3}H ingress	3.74E+01 (0.52%) in-situ; 1.78E+02 (2.45%) as backfill	1.66E+02 (1.22%) in-situ; 7.90E+02 (5.81%) as backfill
Total	**4.01E+01 (0.55%) in-situ; 1.90E+03 (2.63%) as backfill**	**2.20E+02 (1.62%) in-situ; 1.05E+03 (7.69%) as backfill**

Further Characterisation

458 Further sample collection and analysis for the purpose of better constraining the Dragon Fuel Store Building inventory is not expected in the near term. Further characterisation may be undertaken as decommissioning proceeds, but this will be after the inventory freeze for assessments in support of the permit application.

459 However, it is considered that the sensitivity analyses described in this section adequately cover the remaining uncertainties, which will be further captured via uncertainty analysis as part of PA. The alternative inventories explored in this section, all of which are considered to be pessimistic, do not result in significant changes to the overall Dragon inventory (i.e. increasing the order of magnitude), or in activity concentration values that come close to the upper limit for UK LLW.

3.8 Dragon Reactor Building (B70) Backfill

3.8.1 Feature Description

460 Any below-ground voids resulting from the demolition of the Dragon Reactor Building will be filled with material originating from the Winfrith site. As stated in the Conceptual Site Model [21], the concrete structures of both the Dragon Reactor Building (B70) and the Dragon Fuel Storage Building (B78) above the cutline (ground level) will be used to fill the Dragon void below the cutline. A remaining void space of 1,099 m^3 is anticipated, which will be filled with material from the existing D630 rubble stockpiles (INV-DRAGON-003). The backfill material is expected to be in the form of wireline-cut concrete blocks and demolition rubble that meets the defined EAC [17] (i.e. metal, wood, hazardous materials, etc., will be excluded from the backfill).

3.8.2 Origin and Constraints on Radiological Inventory

461 The contaminated backfill inventory comprises the above cutline portion of the Dragon bioshield (43% of the total bioshield activity) and the inventory associated with surface contamination and ^3H ingress into the above cutline portion of the B70 building structure (56% of the building surface contamination) and B78 building structure (83% of the building surface contamination). The origin and constraints on these inventory estimates are defined in Sections 3.4, 3.5 and 3.7, respectively. Uncertainties relating to these inventory estimates apply equally to the below cutline portion that will remain in-situ and the above cutline portion that will be demolished and used as backfill.

462 The above cutline portion of the bioshield inventory assigned to the backfill includes the inventory associated with the rebar. Whilst the EAC [17] mean that accessible metal will be excluded from the backfill, where the building is demolished by cutting into separate blocks, the rebar will be retained. If the above cutline building were broken into rubble, then accessible metal would be removed. As the exact demolition plans are still evolving and it is conservative to include the rebar inventory, it has been included in the backfill inventory (INV-DRAGON-003).

463 Preliminary calculations in the CSM [21, §2.6] indicate that the B70 Dragon void volume (below ground level) will be 6,544 m^3. It is currently anticipated that 400 m^3 of blocks wireline cut from the primary containment (all emplaced within Wall C) and 5,045 m^3 of conventional demolition arisings (the total volume expected to be produced from B70 and B78) are to be emplaced in this void, leaving a shortfall of 1,099 m^3. These volumes are initial estimates and are subject to significant uncertainty (INV-DRAGON-003). For the purposes of this inventory assessment, it is assumed that the difference will be met using material from the existing D630 rubble stockpiles (see discussion in Section 2.17.2). Since it is also assumed that the entirety of the stockpiled material will be emplaced into the SGHWR voids, this represents deliberate double counting for the purpose of inventory derivation; however, only a small volume is

estimated to be needed to fill the Dragon shortfall. In the event that not enough stockpile material is available to fill the shortfall in both the SGHWR and Dragon voids, additional clean material will be used. Assuming that rubble stockpile material will make up the entire shortfall is therefore conservative.

3.8.3 Inventory Estimate

464 Average and maximum activity concentrations and an estimate of the radioactive inventory for the Dragon above-ground structures are presented in Table **3.36**. The maximum activity concentration is that for the general building surface contamination fingerprint as this bounds that for the bioshield. In contrast, the average activity concentrations for the backfill are dependent on the demolition approach and backfill processing (i.e. blocks, rubble or compacted rubble), which is subject to significant uncertainty (INV-DRAGON-003). For the purposes of calculating average activity concentrations in this estimate, a density for the entire backfill of 1,967 kg/m^3 (equivalent to the value for compacted broken concrete adopted in SGHWR calculations; Table **2.6**) has been assumed.

Table 3.36: Dragon Reactor (B70) building backfill disposal inventory, including maximum and average activity concentrations and inventory based on average activity concentrations, presented for an inventory reference date of 01/01/2027.

Radio-nuclide	B70 surface contamination and ^{3}H ingress [MBq]	B78 surface contamination and ^{3}H ingress [MBq]	Bioshield Concrete - Portland [MBq]	Bioshield Concrete - Barytes [MBq]	Bioshield Rebar [MBq]	Stockpile rubble [MBq]	Total Disposal Inventory [MBq]	Average Activity [Bq/g]	Maximum Activity (building contamination) [Bq/g]
^{3}H	9.95E+02	1.81E+02	7.29E+02	2.11E+02	1.36E+00	5.66E+01	2.17E+03	1.69E-01	7.04E+01
^{14}C	5.94E-01	1.92E-01	5.78E+00	1.67E+00	1.66E+00	1.10E+01	2.09E+01	1.62E-03	5.09E-01
^{36}Cl					5.47E-01		5.47E-01	4.25E-05	
^{41}Ca			1.22E+01	2.22E+00			1.44E+01	1.12E-03	
^{55}Fe					7.77E-01		7.77E-01	6.04E-05	
^{60}Co	1.42E-01	4.59E-02	1.27E+00	3.68E-01	4.06E-01	2.78E+00	5.01E+00	3.89E-04	3.47E-01
^{63}Ni	4.15E-01	1.34E-01	1.32E+01	3.81E+00	1.52E+01	6.66E+01	9.93E+01	7.71E-03	6.58E-01
^{90}Sr	1.02E+01	3.29E+00				3.30E+02	3.43E+02	2.66E-02	1.34E+01
^{137}Cs	7.42E+00	2.40E+00			1.03E-02	9.18E+02	9.28E+02	7.21E-02	6.81E+01
^{133}Ba			1.58E+00	4.04E+01			4.19E+01	3.26E-03	
^{148}Sm			6.23E-28	1.80E-28			8.03E-28	6.24E-32	
^{151}Sm			3.16E+00	9.17E-01			4.08E+00	3.17E-04	
^{152}Gd			1.34E-12	3.88E-13			1.73E-12	1.34E-16	
^{152}Eu			6.78E+01	1.96E+01			8.75E+01	6.80E-03	
^{154}Eu			1.86E+00	5.39E-01			2.40E+00	1.86E-04	
^{210}Pb	1.70E-01	5.49E-02					2.25E-01	1.75E-05	1.54E-01
^{226}Ra	3.25E-01	1.05E-01					4.30E-01	3.34E-05	1.49E-01
^{228}Ra	4.31E-03	1.39E-03					5.71E-03	4.43E-07	5.51E-03
^{227}Ac	7.39E-06	2.39E-06					9.77E-06	7.59E-10	5.52E-05

Radio-nuclide	B70 surface contamination and ^{3}H ingress [MBq]	B78 surface contamination and ^{3}H ingress [MBq]	Bioshield Concrete - Portland [MBq]	Bioshield Concrete - Barytes [MBq]	Bioshield Rebar [MBq]	Stockpile rubble [MBq]	Total Disposal Inventory [MBq]	Average Activity [Bq/g]	Maximum Activity (building contamination) [Bq/g]
^{228}Th	3.31E-03	1.07E-03					4.38E-03	3.40E-07	5.26E-03
^{229}Th	8.28E-14	2.68E-14					1.10E-13	8.51E-18	4.96E-13
^{230}Th	8.27E-03	2.67E-03					1.09E-02	8.50E-07	4.82E-03
^{232}Th	6.60E-03	2.13E-03					8.73E-03	6.78E-07	6.00E-03
^{231}Pa	5.78E-05	1.87E-05					7.64E-05	5.94E-09	2.06E-04
^{233}U	3.02E-10	9.77E-11					4.00E-10	3.11E-14	7.63E-10
^{234}U	9.79E-01	3.16E-01				5.38E+00	6.68E+00	5.19E-04	2.21E+00
^{235}U	3.10E-01	1.00E-01				2.69E+00	3.10E+00	2.41E-04	4.70E-01
^{238}U	1.21E+00	3.91E-01				3.23E+01	3.39E+01	2.63E-03	2.45E+00
^{237}Np	1.61E-05	5.21E-06					2.13E-05	1.66E-09	1.70E-05
^{238}Pu						2.35E+00	2.35E+00	1.82E-04	
^{239}Pu						1.17E+01	1.17E+01	9.10E-04	
^{240}Pu						1.63E+01	1.63E+01	1.27E-03	
^{241}Pu						4.63E+01	4.63E+01	3.60E-03	
^{241}Am	5.63E+00	1.82E+00				2.46E+01	3.21E+01	2.49E-03	2.49E+00
^{244}Cm						8.13E+00	8.13E+00	6.31E-04	
Total	**1.02E+03**	**1.90E+02**	**8.36E+02**	**2.81E+02**	**1.99E+01**	**1.53E+03**	**3.884E+03**	**3.02E-01**	**1.61E+02**

3.8.4 Sensitivity Analysis and Further Characterisation

465 Uncertainties in the Dragon Reactor Building backfill inventory can be split into the following:

- Those relating to demolition and backfilling strategy (INV-DRAGON-002 and INV-DRAGON-003) rather than inherent radiological properties of backfill components, by which exact volumes, densities and locations of the different backfill components are not yet known (but will be known at the time of emplacement). Consideration of the impact of such options is not within the scope of this inventory report and will instead be covered in optimisation and assessment reports.
- Those relating to the inventory of components used only in the backfill (i.e. those without an in-situ portion). In the case of Dragon, this applies only to the component of backfill sourced from the existing rubble stockpiles (INV-DRAGON-006). Further characterisation of this rubble is not expected until emplacement, well after the inventory freeze for assessments in support of the permit application. A sensitivity analysis covering this uncertainty is presented below.
- Those relating to the inventory of backfill components that also have an in-situ portion, which have been covered in sensitivity analyses in the relevant sections of this report:
    - Bioshield (Portland concrete, barytes concrete and rebar): inventories calculated using maximum rather than average activity concentrations.
    - Dragon Reactor (B70) and Fuel Store (B78) buildings: inventories calculated i) using an alternative fingerprint including Pu isotopes, ii) assuming that 100% rather than 5% of the surface contamination is present, and iii) calculating tritium ingress using maximum rather than average activity concentrations. (The fourth Reactor Building alternative inventory, considering an area of high tritium contamination in the Betalite store area, does not impact the backfill inventory as the Betalite area is below ground and will remain in-situ.)

466 The impact of these sensitivity analyses on the backfill inventory is discussed below.

Rubble Component

467 An alternative inventory for the volume of rubble expected to be needed to fill the remaining Dragon voids (once above-ground demolition material has been emplaced) can be calculated using the maximum rather than average activity concentrations. This is shown in the fifth column of Table **3.37**. Doing so results in a modest increase in total activity for this backfill component, from 1.53E+03 MBq to 1.68E+03 MBq and in the total backfill activity from 3.88E+03 MBq to 4.03E+03 MBq (at a date of 01/01/2027). The majority of this increase is associated with ^{3}H (5.66E+01 MBq to 1.72E+02 MBq), with smaller increases in ^{14}C and U isotopes. Assuming no increase in the inventory of any of the other Dragon features, the backfill would then contribute 54.7%, rather than 53.7%, to the overall Dragon inventory.

468 Since the approach to estimating the activity of the rubble mounds (Section 2.17) is believed to be conservative, this scenario is considered to sufficiently cover any

remaining uncertainty, and further uplift for this component would be unnecessarily pessimistic.

Impact of all Sensitivity Analyses

469 Table **3.37** presents an alternative inventory for the backfill, taking into account the higher inventories for the bioshield and rubble components as previously calculated, and also considering the impact of both assuming 100% of general area contamination is present and calculating ^{3}H ingress using maximum rather than average activity concentrations, and (separately) using the alternative Pu-containing fingerprint for the Dragon Reactor (B70) and Fuel Storage (B78) buildings. The latter scenario changes the distribution of radionuclides but has a negligible effect on the total activity.

470 The combined alternative bioshield and rubble inventories, but still assuming 5% general area contamination is present and ^{3}H ingress is calculated using average concentrations (as in the reference inventory estimate), result in an increase in total backfill activity from 3.88E+03 MBq to 7.72E+03 MBq.

471 The combined alternative bioshield and rubble inventories, assuming 100% general area contamination is present and ^{3}H ingress is calculated using maximum concentrations, result in an increase in total backfill activity from 3.88E+03 MBq to 1.16E+04 MBq. There would be a reduction in the contribution of backfill to the overall Dragon inventory from 53.7% to 51.7%.

472 If the higher bioshield inventory is not taken into account, the combined alternative rubble and Dragon Reactor Building inventories result in an increase in total backfill activity from 3.88+03 MBq to 7.87E+03 MBq.

Table 3.37: Alternative inventory for Dragon Reactor Building backfill, taking into account higher inventories for the above-cutline (ACL) bioshield and rubble components, and also considering the impact of both assuming 100% of general area contamination is present and calculating ^{3}H ingress using maximum rather than average activity concentrations (using both the fingerprint used in the reference inventory estimate and the alternative Pu-containing fingerprint for the Dragon Reactor Building). All activity data are presented at 01/01/2027.

Radio-nuclide	Bioshield Concrete - Portland [MBq]	Bioshield Concrete - Barytes [MBq]	Bioshield Rebar [MBq]	Stockpile rubble [MBq]	5% contam. present and average ^{3}H ingress, as per reference inventory			100% contam. present and max ^{3}H ingress					
					Using best-estimate FP as per reference inventory						Using alternative Pu fingerprint		
					B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]	B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]	B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]
^{3}H	3.29E+03	9.53E+02	2.32E+00	1.72E+02	9.95E+02	1.81E+02	5.59E+03	3.45E+03	8.68E+02	8.74E+03	3.42E+03	8.56E+02	8.69E+03
^{14}C	1.66E+01	4.80E+00	2.38E+00	1.30E+01	5.94E-01	1.92E-01	3.75E+01	1.19E+01	3.84E+00	5.24E+01	1.92E+01	6.22E+00	6.22E+01
^{36}Cl			1.17E+00				1.17E+00			1.17E+00	2.77E+01	8.94E+00	3.78E+01
^{41}Ca	3.75E+01	6.83E+00					4.43E+01			4.43E+01			4.43E+01
^{55}Fe			1.83E+00				1.83E+00			1.83E+00	1.55E-01	5.02E-02	2.03E+00
^{60}Co	4.42E+00	1.28E+00	4.06E-01	2.78E+00	1.42E-01	4.59E-02	9.07E+00	2.84E+00	9.19E-01	1.26E+01	1.87E+00	6.03E-01	1.14E+01
^{63}Ni	4.26E+01	1.23E+01	3.06E+01	6.66E+01	4.15E-01	1.34E-01	1.53E+02	8.30E+00	2.68E+00	1.63E+02	4.95E+01	1.60E+01	2.18E+02
^{90}Sr				3.30E+02	1.02E+01	3.29E+00	3.43E+02	2.04E+02	6.58E+01	5.99E+02	1.10E+02	3.56E+01	4.75E+02
^{125}Sb				2.06E-03			2.06E-03			2.06E-03			2.06E-03
^{137}Cs			1.03E-02	9.18E+02	7.42E+00	2.40E+00	9.28E+02	1.48E+02	4.79E+01	1.11E+03	2.95E+02	9.52E+01	1.31E+03
^{133}Ba	4.92E+00	1.26E+02					1.31E+02			1.31E+02			1.31E+02
^{148}Sm	1.92E-27	5.56E-28					2.48E-27			2.48E-27			2.48E-27
^{151}Sm	9.90E+00	2.87E+00					1.28E+01			1.28E+01			1.28E+01
^{152}Gd	4.13E-12	1.20E-12					5.32E-12			5.32E-12			5.32E-12
^{152}Eu	2.09E+02	6.06E+01					2.70E+02			2.70E+02	2.19E+01	7.06E+00	2.99E+02
^{154}Eu	5.19E+00	1.50E+00					6.70E+00			6.70E+00	2.06E+00	6.65E-01	9.42E+00

Radio-nuclide	Bioshield Concrete - Portland [MBq]	Bioshield Concrete - Barytes [MBq]	Bioshield Rebar [MBq]	Stockpile rubble [MBq]	5% contam. present and average ^{3}H ingress, as per reference inventory			100% contam. present and max ^{3}H ingress					
					Using best-estimate FP as per reference inventory						Using alternative Pu fingerprint		
					B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]	B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]	B70 ACL surface contam. and ^{3}H ingress [MBq]	B78 ACL surface contam. and ^{3}H ingress [MBq]	Total Disposal Inventory [MBq]
^{210}Pb					1.70E-01	5.49E-02	2.25E-01	3.40E+00	1.10E+00	4.50E+00	1.18E-08	3.81E-09	1.56E-08
^{226}Ra					3.25E-01	1.05E-01	4.30E-01	6.50E+00	2.10E+00	8.60E+00	1.38E-07	4.47E-08	1.83E-07
^{228}Ra					4.31E-03	1.39E-03	5.71E-03	8.63E-02	2.79E-02	1.14E-01	1.52E+00	4.92E-01	2.02E+00
^{227}Ac					7.39E-06	2.39E-06	9.77E-06	1.48E-04	4.77E-05	1.95E-04	3.02E-05	9.75E-06	3.99E-05
^{228}Th					3.31E-03	1.07E-03	4.38E-03	6.62E-02	2.14E-02	8.76E-02	1.17E+00	3.78E-01	1.55E+00
^{229}Th					8.28E-14	2.68E-14	1.10E-13	1.66E-12	5.35E-13	2.19E-12	5.93E-14	1.92E-14	7.84E-14
^{230}Th					8.27E-03	2.67E-03	1.09E-02	1.65E-01	5.34E-02	2.19E-01	7.26E-05	2.35E-05	9.60E-05
^{232}Th					6.60E-03	2.13E-03	8.73E-03	1.32E-01	4.26E-02	1.75E-01	2.33E+00	7.53E-01	3.08E+00
^{231}Pa					5.78E-05	1.87E-05	7.64E-05	1.16E-03	3.73E-04	1.53E-03	2.36E-04	7.63E-05	3.12E-04
^{233}U					3.02E-10	9.77E-11	4.00E-10	6.05E-09	1.95E-09	8.00E-09	2.19E-10	7.06E-11	2.89E-10
^{234}U				9.53E+00	9.79E-01	3.16E-01	1.08E+01	1.96E+01	6.32E+00	3.54E+01	8.97E-01	2.90E-01	1.07E+01
^{235}U				4.76E+00	3.10E-01	1.00E-01	5.17E+00	6.21E+00	2.01E+00	1.30E+01	1.27E+00	4.10E-01	6.44E+00
^{238}U				5.72E+01	1.21E+00	3.91E-01	5.88E+01	2.42E+01	7.82E+00	8.92E+01	1.57E+00	5.08E-01	5.92E+01
^{237}Np					1.61E-05	5.21E-06	2.13E-05	3.23E-04	1.04E-04	4.27E-04	2.00E+00	6.48E-01	2.65E+00
^{238}Pu				2.35E+00			2.35E+00			2.35E+00	8.52E-01	2.75E-01	3.47E+00
^{239}Pu				1.17E+01			1.17E+01			1.17E+01	6.28E-01	2.03E-01	1.25E+01
^{240}Pu				1.63E+01			1.63E+01			1.63E+01	2.72E+01	8.79E+00	5.23E+01
^{241}Pu				4.63E+01			4.63E+01			4.63E+01	1.19E-05	3.84E-06	4.63E+01
^{241}Am				2.46E+01	5.63E+00	1.82E+00	3.21E+01	1.13E+02	3.64E+01	1.74E+02	4.37E+00	1.41E+00	3.04E+01
^{244}Cm				8.13E+00			8.13E+00			8.13E+00			8.13E+00
Total	3.62E+03	1.17E+03	3.87E+01	1.68E+03	1.02E+03	1.90E+02	7.72E+03	4.00E+03	1.05E+03	1.16E+04	3.99E+03	1.04E+03	1.15E+04

3.9 Primary Mortuary Hole Structure in B78

3.9.1 Feature Description

473 The Dragon Mortuary Hole Structure was the storage area for Dragon fuel elements and, more recently, for various waste items from the PIE facility in A59. As stated in Section 3.1.2, there are 50 primary mortuary holes (tubes 41-90) for the storage of irradiated fuel and 40 additional holes (tubes 1-40) for the storage of fresh fuel. Only the primary hole structure is proposed for in-situ disposal, since the tubes are embedded in concrete and are difficult to remove. The additional tubes are expected to be relatively easy to remove owing to their cassette structure (Section 3.2) and are therefore not considered further in this inventory estimate.

474 The Primary Mortuary Hole Structure comprises 50 vertical mild steel storage tubes, with external diameter 10.75” specified on drawing AE185813. The fuel/waste items were cooled by a ventilation system, the inlet of which was a filtered vent house inside B78 and the outlet was routed out of B78 to the Dragon complex stack (see Figure **3.6**) [110, §2.1]. The storage tubes are interlinked with horizontal vent ducts and the outer row of storage tubes on each side, holes 81-90 and 41-50, are attached to the inlet duct and the outlet duct, respectively, as shown in Figure **3.18** [110, §2.1]. Figure **3.19** presents a plan view of the entire Dragon Fuel Mortuary Store, showing the layout of the primary and additional holes, the ventilation inlet and outlet for the primary holes, and the locations of the sump and storage pit.

475 The entire structure, made of mild steel, was constructed in a pit beneath ground level in B78 and the pit was then in-filled with concrete. The inventory assessment presented here assumes that all storage hole lids and any detachable parts of the system are removed and that the main ventilation ducts are removed to ground level, the point at which they are embedded in concrete. It is understood that the metal tube system for the fresh fuel holes (1-40) will be removed and then cleaned should any contamination remain (INV-DRAGON-009)[19]. It is also assumed that the metal lining of the storage pit will be removed and the area cleaned (INV-DRAGON-009). In addition, given that the mortuary holes, sump and storage pit are metal-lined, it is anticipated that there has been negligible radionuclide migration into the bulk system concrete (INV-DRAGON-009). Therefore, all that is included in the end state inventory assessment presented here is contamination associated with the steel structure of the spent fuel primary holes (tubes 41-90) and the ventilation and sump system, which are assumed to be infilled with clean grout[20].

476 Table **3.38** summarises dimensions associated with the Primary Mortuary Hole Structure.

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[19] Depending on the method of removal, it is possible that some concrete may be removed along with the additional tubes; such operational details are outside the scope of this report. It is conservative to assume that no concrete is removed and so this possibility is not considered further in the inventory estimate.

[20] The feasibility of full grout penetration into the horizontal vent ducts is outside the scope of this report, but will be discussed elsewhere in the permit application documentation. The starting assumption for this inventory report is that it is feasible.

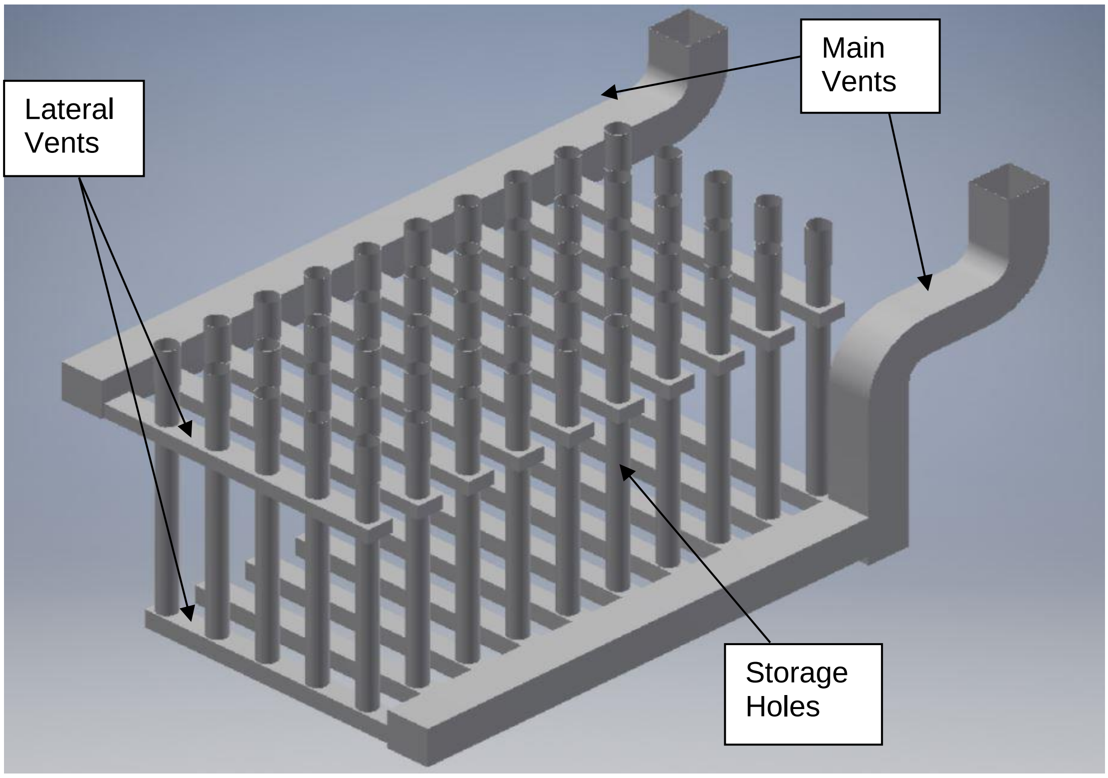


Figure 3.18: Dragon Mortuary Hole steel structure, showing the 50 primary storage tubes, main vents and interlinking ventilation ducts (sump and sump extension tube not shown) [110, Fig.1].

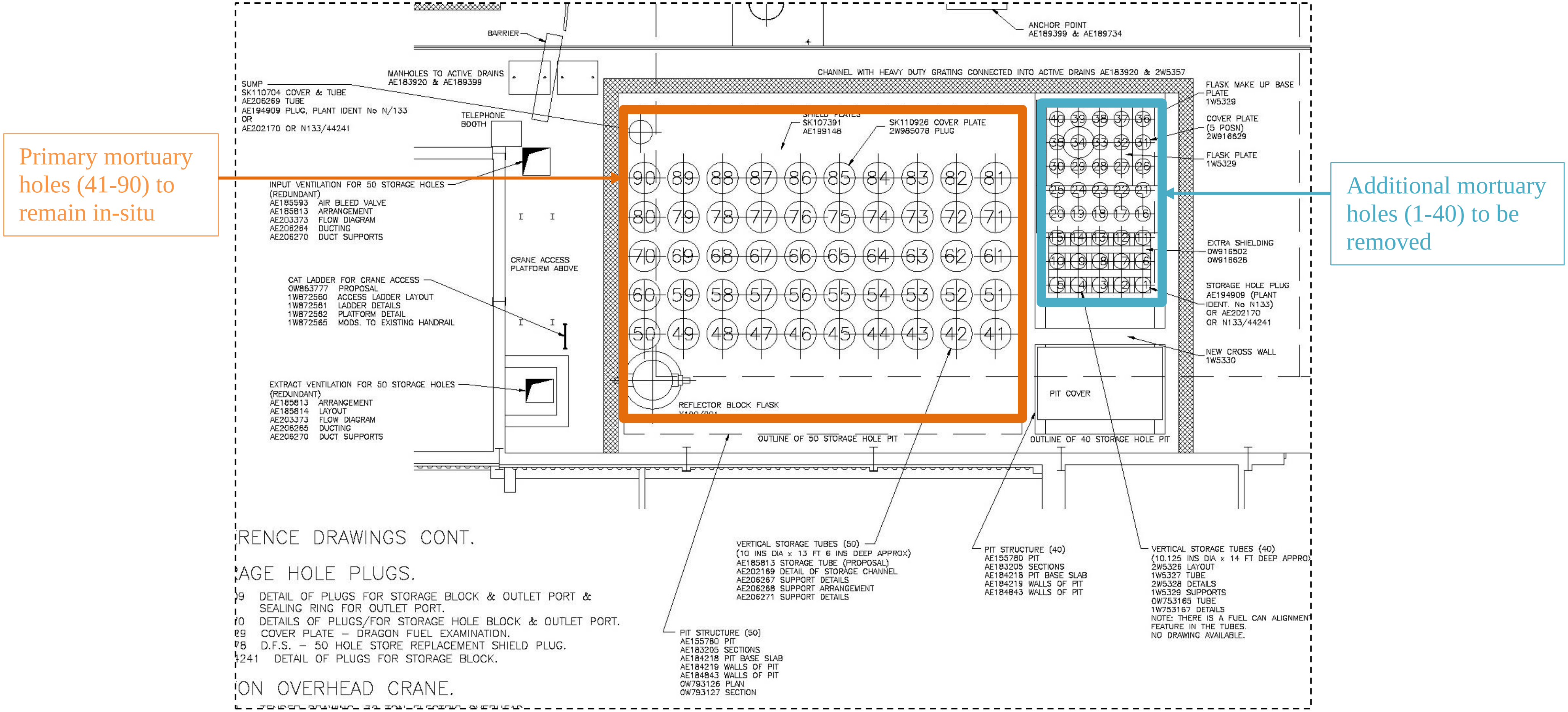


Figure 3.19: Plan view of the Dragon Fuel Store, showing the locations of the 90 storage tubes, sump, and ventilation inlet and outlet (extract from UKAEA drawing 1W985076/A, September 1998).

Table 3.38: Dimensions of the Dragon Fuel Mortuary Hole Structure.

Parameter	Value	Unit	Data Source/Assumption
Dimensions for the 50 primary / irradiated fuel mortuary holes (No. 41-90):			
Outer diameter of mortuary hole	0.273	m	AE185813 gives 10.75" for the outer diameter
Inner diameter of mortuary hole	0.260	m	AE202169A gives 10" for the inner diameter, but AE185813 implies a uniform wall thickness of 0.25", so an ID of 10.25". As a larger diameter gives a higher surface area, and to take some account of the fins in the tubes (as shown on AE202169A), the larger ID is conservatively assumed.
Depth of a mortuary hole tube	4.191	m	OW793126 states depth to base of tube as 13'9"
Pitch (spacing) of mortuary holes	0.762	m	OW793126 hole pitch is 2'6" square
Internal surface area of a single tube 41-90	3.481	m^2	Calculated. AE185813 indicates 6" openings in the tubes for the interlinking ventilation ducts, but this small surface area is not accounted for.
Volume of a single tube 41-90	0.223	m^3	Calculated
Sump dimensions:			
Inner diameter of sump extension tube	0.292	m	AE185813 gives 11.5" for the inner diameter for the top 2' section
Depth of sump extension tube	0.610	m	AE185813 defines the top section as 2' tall
Inner diameter of sump extension tube	0.256	m	AE185813 and AE206269B give 10.3/32" for the inner diameter of the main tube section
Length of sump extension tube	0.826	m	The B78 Store Ducting.ipt file defines the depth as 32.5". This is supported by AE206269B, which shows the top section is cut on a diagonal where it meets the top vent, with short side 2'3" and long side 3'2".
Depth of square sump at base of vent	0.152	m	AE185813 states 6" deep
Length/width of sump at vent base	0.610	m	AE185813 defines sump as 2' square
Internal surface area of sump	1.596	m^2	Calculated - neglects any lid at the top of the sump and doesn't include surface area of base as accounted for in main vent calculation below
Volume of sump	0.140	m^3	Calculated
Main ventilation ducts:			
Width of inlet/outlet ducts (in ground)	0.610	m	OW793126 gives width and depth as 2'; Also supported by drawing AE185813
Depth of inlet/outlet ducts (in ground)	0.610	m	OW793126 gives width and depth as 2'; Also supported by drawing AE185813
Length of inlet/outlet ducts (in ground)	9.525	m	AE185813 gives 31'3" for the duct length

Parameter	Value	Unit	Data Source/Assumption
Depth from surface to top of upper duct	0.838	m	AE185813 gives 2'9" for top of concrete block to top of top vent
Depth from surface to base of lower duct	4.572	m	AE185813 gives (4'0"+3'9"+4'0"+3'3") for top of concrete block to base of bottom vent
Vent steel wall thickness	n/a	m	Lacking any information on steel thickness, the external dimensions are conservatively assumed for the internal
Internal surface area of main ducts	56.357	m^2	Calculated. Curvature in vent ducts is ignored and simplified to right angles for surface calculations.
Volume of main ducts	8.863	m^3	Calculated
Lateral linking cross vents between Mortuary Holes 41-50:			
Top vent width (external)	0.381	m	AE185813 states 15" x 9" deep for the linking ducts
Top vent height (external)	0.229	m	AE185813 states 15" x 9" deep for the linking ducts
Top vent length (external)	4.013	m	AE185813 states (6"+5'8"+something [ass.5'9"]+6'11") for the length of the linking duct
Vent steel wall thickness	n/a	m	Lacking any information on steel thickness, the external dimensions are assumed to be the internal dimensions
Surface area of top linking vents	44.5	m^2	Calculated
Volume of top linking ducts	3.495	m^3	Calculated
Bottom vent width	0.305	m	AE185813 gives 12" square openings in the main ducts for the bottom linking ducts
Bottom vent height	0.305	m	AE185813 shows the height varies between 10" and 12" so 12" is assumed to maximise surface area
Bottom vent length	3.683	m	AE185813 states (9"+4'6"+6'10") for the length of the bottom linking duct
Surface area of bottom linking vents	43.2	m^2	Calculated
Volume of bottom linking ducts	3.422	m^3	Calculated
Total surface area:			
Total surface area	319.7	m^2	Calculated - includes the 50 irradiated mortuary holes, the sump, main ventilation ducts and lateral ventilation ducts
Total void volume to be filled with grout:			
Total void volume	27.1	m3	Calculated - includes the 50 irradiated mortuary holes, the sump, main ventilation ducts and lateral ventilation ducts
Mortuary hole monolith volume (assumes mortuary holes, vents and any other voids are filled with grout, and steel is treated as grout):			
Depth from surface to top of pit base level	4.72	m	OW793126 gives depth to the top of the base level as 15'6"; AE184218 gives 15'3" to the top of the floor layer and 16'9" to the bottom of the floor layer. The value from drawing OW793126 is used, as this dates from 1979 after

Parameter	Value	Unit	Data Source/Assumption
			construction, whilst drawing AE1842218 is a pre-construction drawing from 1961.
Depth from surface to base of base level	5.18	m	The additional 1'6" floor layer from AE184218 is included
Width of concrete base slab (exc walls)	6.40	m	AE184218 gives the base slab width as 21'0"
Length of concrete base slab (exc walls)	7.77	m	AE184218 gives the 41-90 hole area length as (25'6")
Concrete monolith volume	235.0	m^3	Calculated - uses the hole excavated in which the metal tube framework and vents were placed before being surrounded by concrete to ground level.
Contaminated steel volume	0.32	m^3	Calculated. Total volume of contaminated steel below ground - assumes 1 mm contamination of the mild steel and the total surface area calculated above.

3.9.2 Origin and Constraints on Radiological Inventory

477 The inventory estimate for the Primary Mortuary Hole Structure is principally based on the results of a systematic sampling campaign undertaken in 2023, described in detail in the sampling and analysis plan [159] and summarised below. This campaign was driven by a lack of existing characterisation, identified as a key uncertainty in previous versions of this inventory report.

478 The derivation of the inventory by NRS is described in a separate Note for the Record [160] and accompanying spreadsheet [161], from which the information below is summarised.

2023 Initial Dose Rate and Smear Sampling Survey of All Mortuary Holes

479 The following regime was completed in July 2023 for each mortuary hole (MH):

- Count rate surveys were completed using a teletector probe at the following positions (to give a dose rate (beta/gamma) radiation reading in μSv/hr):
  - over the hole (with the plug removed);
  - at 1 m, 2 m, 3 m and 4 m below the plug and at the base of the hole; and
  - at the top cross vent.
- Smear samples were taken at the following positions and analysed for alpha and beta/gamma radiation using a DP6 ‘DD’ probe to give a counts per second (cps) value:
  - the top portion of the hole;
  - the top cross vent (taken from the inside the cross vent using a long reach tool); and
  - the full height of the hole.

480 The smear from the full height of the MH was taken using a flexible spherical object which has the same diameter of the MHs. The object was covered in smear paper and was attached to a long pole and inserted into and pulled through the full height of each MH.

481 During the monitoring of MHs 67, 69, 73 and 85, elevated beta/gamma count rates were identified in specific locations within each MH. These were recorded on the monitoring and sampling record. The relevant smear samples did not always pick up the same levels of radioactivity, potentially suggesting localised hotspots of fixed contamination within these holes. To account for any potential discrepancy in pick-up factors (in calculating the inventory) caused by this fixed contamination, the count rates for the relevant smears were adopted as follows:

- Monitoring of MH67 recorded a localised reading of 5,500 cps close to the top of the MH. MH67 has the highest recorded top smear count rate; therefore, this is assumed to be representative of the fixed contamination measurement.
- Monitoring of MH69 recorded a localised dose rate measurement of 590 cps, 1 m from the top of MH plug. The count rate of the full height smear (background of 5 cps) cannot be justified as representative of the localised hotspot; the full height measurement for MH75 (the highest recorded full-height smear count rate at 114 cps) was therefore used instead.
- Monitoring of MH73 recorded a localised reading of 1,050 cps at the cross vent position; however, the cross vent smear recorded only a background reading of 5 cps beta/gamma. The highest recorded cross vent smear count rate (464 cps from MH83) was therefore used instead.
- Monitoring of MH85 recorded a localised measurement of 750 cps at the cross vent, with the corresponding smear measuring only 32 cps. The highest recorded cross vent smear count rate (464 cps from MH83) was therefore used instead.

482 The following measurements were not taken and count rates were adopted based on reasonable assumptions:

- The MH80 and MH90 top smear count rates were not recorded. These were assumed to be at background levels as all other measurements taken in these MHs were at background and no anomalies were noted by the surveyor.
- The cross vent smear from MH75 was not recorded. The highest recorded cross vent smear count rate (464 cps from MH83) was used, as the other measurements in MH75 indicate elevated levels of contamination.

483 The count rate results of the initial sampling exercise are presented in full in [161] and are summarised in Table **3.39**.

Table 3.39: Top smear, full height and cross vent count rate measurements in counts per second (cps) from all MHs. Background (bgd) measurements are given in [161]. Readings in brackets are the values assumed/substituted in the cases described in the text. Readings highlighted in blue are the highest measurements for each location (top smear, full height and cross vent).

Measurement	Beta (cps)	Beta (cps)	Beta (cps)	Beta (cps)	Beta (cps)
MH No.	41	51	61	71	81
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	18	bgd	40	bgd	11
Cross vent	27	bgd	140	bgd	20
MH No.	42	52	62	72	82
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	14	bgd	bgd	13	bgd
Cross vent	bgd	bgd	10	40	bgd
MH No.	43	53	63	73	83
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	15	bgd	30	bgd	48
Cross vent	28	24	25	bgd (464)	464
MH No.	44	54	64	74	84
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	11	11	bgd	bgd	12
Cross vent	bgd	bgd	bgd	bgd	19
MH No.	45	55	65	75	85
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	10	bgd	bgd	114	10
Cross vent	24	bgd	25	Not recorded (464)	32 (464)
MH No.	46	56	66	76	86
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	bgd	10	bgd	bgd	11
Cross vent	bgd	11	9	12	bgd
MH No.	47	57	67	77	87
Top smear	bgd	bgd	145	bgd	bgd
Full height	bgd	bgd	5	25	10
Cross vent	bgd	bgd	15	bgd	17
MH No.	48	58	68	78	88
Top smear	bgd	bgd	bgd	bgd	bgd
Full height	3	12	5	bgd	15
Cross vent	bgd	bgd	bgd	bgd	16
MH No.	49	59	69	79	89
Top smear	bgd	bgd	30	bgd	bgd
Full height	bgd	bgd	bgd (114)	15	bgd

Measurement	Beta (cps)	Beta (cps)	Beta (cps)	Beta (cps)	Beta (cps)
Cross vent	18	bgd	15	27	4
MH No.	50	60	70	80	90
Top smear	bgd	bgd	50	Not recorded (bgd)	Not recorded (bgd)
Full height	15	bgd	bgd	bgd	bgd
Cross vent	2	bgd	bgd	bgd	bgd

2023 Radioisotope Analysis of Selected Smear Samples

484 Following the initial exercise, the smear samples with the highest measurements for each location (highlighted in blue in Table **3.39**) were selected for detailed radioisotope analysis:

- MH67 had the highest reading for the top smear sample (145 cps beta/gamma);
- MH75 had the highest reading for the full-height smear sample (114 cps beta/gamma);
- MH83 had the highest reading for the smear samples taken from inside the cross vent (464 cps beta/gamma).

485 These samples were analysed by the Socotec laboratory in Didcot, Oxfordshire, with results provided in March 2024. All three samples were subjected to the following analysis: gamma spectrometry; gross alpha and beta; total tritium; ^{14}C, ^{55}Fe and ^{63}Ni; and ^{90}Sr. Additionally, the cross-vent sample from MH83 was analysed for alpha radionuclides (plutonium, americium, curium and uranium isotopes). This analysis was not completed for the samples from MH67 or MH75 as the probe count on the smears was low and it was not expected that this analysis would gain reliable data.

486 Results reported at the LOD were managed as follows:

- For ^{60}Co, ^{3}H, ^{55}Fe, ^{63}Ni and ^{241}Am, the LOD result was taken to be the actual value and used as part of the fingerprint.
- LOD results for alpha species (^{233}U, ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{243}Cm and ^{244}Cm) from the MH83 sample were used to estimate the activities of the equivalent isotopes for MH67 and MH75 by calculating the ratio of each species to ^{241}Am in MH83 and scaling to the ^{241}Am value in MH67 and MH75.

487 In both cases the LOD result was used at its full value and considered to represent a conservative estimate of the activity of each of the radionuclides named above.

Derivation of Fingerprints for MHs and Cross Vents Based on 2023 Sampling

488 In order to derive appropriate fingerprints to use in the inventory calculation for the MHs and cross vents, the radioisotope data from MHs 67, 75 and 83 described above were processed as follows:

- Isotopes that are either attributed to naturally-occurring radioactivity, have short half-lives or are part of the decay chain for another radionuclide were removed from the dataset (i.e. ^{7}Be, ^{40}K, ^{134}Cs, ^{208}Tl, ^{210}Pb, ^{212}Bi, ^{212}Pb, ^{214}Bi, ^{214}Pb, ^{226}Ra, ^{228}Ac, ^{234m}Pa and ^{234}Th).
- The contribution of individual alpha radionuclides to the fingerprints of MH67 and MH75 were assessed by applying the ratio of ^{241}Am measurements to the

ratio defined from the speciated alpha analysis completed for MH83, as described above.

- $^{233}U/^{234}U$, $^{239}Pu/^{240}Pu$ and $^{243}Cm/^{244}Cm$ were reported as a combined value from the alpha spectrometry analysis. Review of the other Dragon fingerprints identifies these ratios as close to 50/50 in other areas of the facility, so the reported value was split equally in each case. This is justified as the specific activities are low and the eco-toxicity is similar across each pair.
- The ^{235}U result from the alpha spectrometry analysis (for MH 83) was selected in preference to the gamma spectrometry measurement, as this is generally a more accurate measurement.

489 The resulting fingerprints for MHs 67, 75 and 83 are presented in Table **3.40**.

Inventory and Fingerprint for Main Ducts and Sump

490 An inventory assessment for the Primary Mortuary Hole Structure was completed in 2016 [110], which was based on a smear sample taken from the outlet ventilation stack. A sample was also taken from the inlet ventilation stack, but as expected, the outlet vent smear sample contained a higher level of activity than the inlet and was therefore used as conservative. This sample is considered to still be the most representative of the main ventilation ducts and sump, as the 2023 sampling campaign did not sample these components.

491 To derive an inventory for the main ducts and sump, the 2016 inventory was converted to activity concentrations (Bq/cm^2) by dividing the activity data by the total surface area of the ventilation system. The activity concentrations were then decayed to 17/07/2023 (the reference date for the 2023 dataset), before being multiplied by the surface area for the main ducts and sump to derive activity values for these components. As ^{233}U was not included in the 2016 assessment, it was assumed that ^{233}U had the same activity value as ^{234}U. The resulting fingerprint (activity values renormalised to 100%) is presented in Table **3.40**.

Table 3.40: Fingerprints used in the derivation of the inventory of the Primary Mortuary Hole Structure [160, Tab.4]. Reference date 17/07/2023.

Radio-nuclide	FPs based on 2023 dataset, used for MHs and cross vents			FP based on 2016 dataset, used for main ducts and sump (%)
	FP MH67 (%)	FP MH75 (%)	FP MH83 (%)	
^{241}Am	0.19	1.98	1.32	3.51
^{243}Cm	0.01	0.10	0.07	0.12
^{244}Cm	0.01	0.10	0.07	0.10
^{238}Pu	0.38	3.94	2.62	2.18
^{239}Pu	0.05	0.52	0.35	0.28
^{240}Pu	0.05	0.52	0.35	0.28
^{233}U	0.50	5.16	3.44	0.02
^{234}U	0.50	5.16	3.44	0.02
^{235}U	0.00	0.03	0.02	0.00
^{236}U	0.00	0.00	0.00	0.00

Radio-nuclide	FPs based on 2023 dataset, used for MHs and cross vents			FP based on 2016 dataset, used for main ducts and sump (%)
	FP MH67 (%)	**FP MH75 (%)**	**FP MH83 (%)**	
^{238}U	0.00	0.03	0.02	0.00
^{14}C	0.42	1.78	3.25	0.41
^{60}Co	0.08	0.19	0.03	0.07
^{137}Cs	65.06	49.77	57.85	54.43
^{55}Fe	0.16	0.11	0.04	0.01
^{3}H	0.04	0.16	0.52	0.77
^{63}Ni	0.41	0.18	0.08	0.03
^{241}Pu	0.56	5.82	3.88	12.26
^{90}Sr	31.58	24.43	22.65	25.51
Total	**100.00**	**100.00**	**100.00**	**100.00**

Probe Response Correction and Activity Conversion

492 As discussed in Section 3.5.2, the detection efficiency of the DP6 probe must be accounted for and the count rate converted to an activity. This was done using the following formula [160, Eqn.1]:

$$Activity\ Concentration = \frac{Direct\ probe\ count\ rate}{Instrument\ response\ for\ Bq/cm^2} \times probe\ surface\ area(cm^2) \times \frac{100}{pick\ up\ efficiency} \times \frac{1}{area\ smeared(cm^2)}$$

where the:

- Direct probe count rate was measured from the smear sample.
- The instrument response for the different fingerprints has been calculated [161] to be 17.51 for MH67, 18.87 for MH75 and 14.66 for MH83.
- The probe surface area is 100 cm^2.
- The pick-up efficiency of the smear is assumed to be 10%.
- The area smeared is assumed to be approximately 300 cm^2 (see Section 3.9.4 for further discussion).

493 The pick-up efficiency of 10% included within this equation is intended to account for both loose and fixed contamination not picked up by the smear, which will only remove a portion of the loose contamination present. A pick-up efficiency value of 10% is commonly used to estimate the activity not measured through smear sampling. However, due to the fact that the 2016 assessment showed that the majority of contamination is loose [110], the actual pick-up efficiency is likely to be higher; the 10% assumed in this assessment is therefore conservative.

494 The fingerprint associated with the fixed component of the contamination may be different to the loose contamination measured within this assessment, but as noted above, this is expected to be a minor component.

495 The uncertainty associated with the fingerprint and the pick-up efficiency is recorded in Appendix A (INV-DRAGON-008).

3.9.3 Inventory Estimate

Mortuary Holes

496 Two separate inventory estimates were derived for the mortuary holes component of the Primary Mortuary Hole Structure, one based on the top smear count rates for each MH and the fingerprint and probe response value for MH67 (the MH with the highest top smear reading), and the second based on the full-height smear count rates for each MH and the fingerprint and probe response value for MH75 (the MH with the highest full-height smear reading). For each estimate, the measured top or full-height smear count rate for each MH was converted to a Bq/cm^2 value using the probe response conversion formula, then multiplied by the surface area of a single mortuary hole ($3.418\ m^3$, Table **3.38**) to give a total inventory for each MH. The relevant fingerprint was then used to define the radionuclide breakdown.

497 An average inventory was defined by taking the average of the two activity values for each radionuclide, and a maximum inventory was defined by taking the highest value of the two activities for each radionuclide. For all but one radionuclide (^{63}Ni), the higher reading corresponds to the inventory derived using the MH75 fingerprint and full-height smear count rates.

Cross Vents

498 Two inventory estimates were derived for the cross vents component of the Primary Mortuary Hole Structure:

- An average inventory was calculated for each of ten (combined top and bottom) cross vents using the average cross vent count rate for the group of five MHs connected by each of the ten cross vents, and the fingerprint and probe response value for MH83 (the MH with the highest cross vent smear reading). The average cross vent count rate for each group of MHs was converted to a Bq/cm^2 value using the probe response conversion formula, then multiplied by the surface area of a single top vent plus corresponding bottom vent ($8.76\ m^3$, one tenth of the total surface area of all top and bottom cross vents as set out in Table **3.38**) to give a total inventory for each (top plus bottom) cross vent. The relevant fingerprint was then used to define the radionuclide breakdown.
- A maximum inventory was calculated in the same way but using the highest (rather than average) cross vent count rate for each group of five MHs connected by each of the ten cross vents.

Main Vents and Sump

499 The inventory for the main vents and sump was calculated directly from the 2016 inventory estimate as described in Section 3.9.2; no maximum inventory was derived.

Total Primary Mortuary Hole Structure Inventory

500 The average and maximum inventories for the individual components and the total inventory for the Mortuary Hole Structure, derived as described above for a reference date of 17/07/2023, are shown in Table **3.41**.

Table 3.41: Summary of Primary Mortuary Hole Structure component and total inventories in Bq for a reference date of 17/07/2023 [160, Tab.5].

	Mortuary Holes		Cross Vents		Main Ducts and Sump		Total Inventory	
Radionuclide	**A Average inventory**	**B Maximum inventory**	**C Average inventory**	**D Maximum inventory**	**E Main ducts**	**F Sump**	**Average inventory (A+C+E+F)**	**Maximum inventory (B+D+E+F)**
^{241}Am	5.37E+04	1.02E+05	1.32E+05	3.18E+05	7.65E+05	2.17E+04	9.73E+05	1.21E+06
^{243}Cm	2.78E+03	5.25E+03	6.84E+03	1.64E+04	2.52E+04	7.14E+02	3.56E+04	4.76E+04
^{244}Cm	2.78E+03	5.25E+03	6.84E+03	1.64E+04	2.22E+04	6.28E+02	3.24E+04	4.45E+04
^{238}Pu	1.07E+05	2.01E+05	2.62E+05	6.30E+05	4.76E+05	1.35E+04	8.59E+05	1.32E+06
^{239}Pu	1.42E+04	2.68E+04	3.50E+04	8.40E+04	6.19E+04	1.75E+03	1.13E+05	1.74E+05
^{240}Pu	1.42E+04	2.68E+04	3.50E+04	8.40E+04	6.19E+04	1.75E+03	1.13E+05	1.74E+05
^{233}U	1.40E+05	2.64E+05	3.44E+05	8.26E+05	3.87E+03	1.10E+02	4.88E+05	1.09E+06
^{234}U	1.40E+05	2.64E+05	3.44E+05	8.26E+05	3.87E+03	1.10E+02	4.88E+05	1.09E+06
^{235}U	7.74E+02	1.46E+03	1.91E+03	4.57E+03	2.59E+02	7.34E+00	2.95E+03	6.30E+03
^{236}U	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.59E+02	7.34E+00	2.67E+02	2.67E+02
^{238}U	7.56E+02	1.43E+03	1.86E+03	4.47E+03	2.59E+02	7.34E+00	2.88E+03	6.16E+03
^{14}C	5.21E+04	9.11E+04	3.25E+05	7.80E+05	8.90E+04	2.52E+03	4.69E+05	9.63E+05
^{60}Co	6.14E+03	9.71E+03	2.69E+03	6.46E+03	1.54E+04	4.35E+02	2.46E+04	3.20E+04
^{137}Cs	2.29E+06	2.55E+06	5.79E+06	1.39E+07	1.19E+07	3.36E+05	2.03E+07	2.86E+07
^{55}Fe	5.43E+03	5.82E+03	4.49E+03	1.08E+04	1.18E+03	3.34E+01	1.11E+04	1.78E+04
^{3}H	4.70E+03	8.21E+03	5.16E+04	1.24E+05	1.67E+05	4.72E+03	2.28E+05	3.04E+05
^{63}Ni	1.09E+04	1.28E+04	7.85E+03	1.88E+04	7.37E+03	2.09E+02	2.63E+04	3.92E+04
^{241}Pu	1.58E+05	2.98E+05	3.88E+05	9.31E+05	2.67E+06	7.57E+04	3.29E+06	3.98E+06
^{90}Sr	1.12E+06	1.25E+06	2.27E+06	5.44E+06	5.56E+06	1.57E+05	9.10E+06	1.24E+07
Total	**4.12E+06**	**5.12E+06**	**1.00E+07**	**2.40E+07**	**2.18E+07**	**6.17E+05**	**3.65E+07**	**5.15E+07**

501 The total Primary Mortuary Hole Structure inventories presented in Table **3.41** were then decayed to the inventory reference date of 01/01/2027 using the GoldSim-RT software package [132; 133; 134]. The average inventory is taken as the reference inventory.

502 To estimate the activity concentration required for PA calculations (especially for the inadvertent human intrusion assessment), three alternative contamination volumes have been assumed (INV-DRAGON-008), with the calculations reported in [15]:

- If it is assumed that all the contamination is located in the first 1 mm thickness of the entire mild steel structure, then the total volume of contaminated steel below ground is 0.32 m^3 (this is calculated using the total surface area of 319.7 m^2 from Table **3.38**).
- As the steel structure is assumed to be filled with grout and the contamination is thought to be generally loose surface contamination, the contamination could be averaged over the infill volume of 27.1 m^3 (Table **3.38**).
- The largest volume (235.0 m^3) that could be considered is that formed by the planned concrete monolith, namely the 4.7 m deep x 6.4 m wide by 7.8 m long (Table **3.38**) pit in which the steel structure and eventual grout infill sit (the small volume occupied by the steel is treated as concrete for this calculation).

503 Table **3.42** presents the estimated Primary Mortuary Hole Structure total activity and range of activity concentration options at 01/01/2027. The entire inventory is assumed to form a below-ground level in-situ disposal.

Table 3.42: Estimated Dragon Primary Mortuary Hole Structure in-situ disposal reference inventory, including a range of possible average activity concentrations, presented for an inventory reference date of 01/01/2027. The activity concentrations have been calculated assuming a density for plain carbon steel of 7,860 kg/m^3 [39, p.12-204] and a density for concrete of 2,400 kg/m^3 (Table **3.23**).

Radionuclide	Activity averaged over 1 mm of steel [Bq/g]	Activity averaged over grout infill [Bq/g]	Activity averaged over pit volume [Bq/g]	Total primary mortuary hole activity [MBq]
^{3}H	7.46E-02	2.89E-03	3.32E-04	1.88E-01
^{14}C	1.87E-01	7.21E-03	8.31E-04	4.69E-01
^{55}Fe	1.84E-03	7.13E-05	8.22E-06	4.63E-03
^{63}Ni	1.02E-02	3.96E-04	4.56E-05	2.57E-02
^{60}Co	6.22E-03	2.41E-04	2.77E-05	1.56E-02
^{90}Sr	3.33E+00	1.29E-01	1.48E-02	8.37E+00
^{137}Cs	7.45E+00	2.88E-01	3.32E-02	1.87E+01
^{210}Pb	1.60E-10	6.19E-12	7.13E-13	4.02E-10
^{226}Ra	4.63E-09	1.79E-10	2.06E-11	1.16E-08
^{228}Ra	3.30E-15	1.28E-16	1.47E-17	8.30E-15
^{227}Ac	4.55E-09	1.76E-10	2.03E-11	1.14E-08
^{228}Th	1.07E-15	4.12E-17	4.75E-18	2.68E-15

Radionuclide	Activity averaged over 1 mm of steel [Bq/g]	Activity averaged over grout infill [Bq/g]	Activity averaged over pit volume [Bq/g]	Total primary mortuary hole activity [MBq]
^{229}Th	6.35E-05	2.45E-06	2.83E-07	1.59E-04
^{230}Th	6.18E-06	2.39E-07	2.75E-08	1.55E-05
^{232}Th	1.81E-14	7.00E-16	8.07E-17	4.55E-14
^{231}Pa	8.58E-08	3.32E-09	3.82E-10	2.15E-07
^{233}U	1.94E-01	7.51E-03	8.65E-04	4.88E-01
^{234}U	1.94E-01	7.51E-03	8.65E-04	4.88E-01
^{235}U	1.17E-03	4.54E-05	5.23E-06	2.95E-03
^{236}U	1.06E-04	4.10E-06	4.73E-07	2.67E-04
^{238}U	1.15E-03	4.44E-05	5.11E-06	2.88E-03
^{237}Np	4.36E-07	1.69E-08	1.94E-09	1.10E-06
^{238}Pu	3.32E-01	1.29E-02	1.48E-03	8.35E-01
^{239}Pu	4.49E-02	1.74E-03	2.00E-04	1.13E-01
^{240}Pu	4.49E-02	1.74E-03	2.00E-04	1.13E-01
^{241}Pu	1.11E+00	4.29E-02	4.94E-03	2.79E+00
^{241}Am	3.92E-01	1.52E-02	1.75E-03	9.85E-01
^{243}Am	1.06E-08	4.10E-10	4.72E-11	2.66E-08
^{243}Cm	1.30E-02	5.04E-04	5.80E-05	3.27E-02
^{244}Cm	1.13E-02	4.37E-04	5.04E-05	2.84E-02
Total	**1.34E+01**	**5.18E-01**	**5.97E-02**	**3.37E+01**

3.9.4 Sensitivity Analysis and Further Characterisation

504 The reference inventory estimate for the Primary Mortuary Hole Structure includes several conservatisms and uncertainties that have been treated conservatively:

- Where LOD values for Co-60, H-3, Fe-55, Ni-63, Am-241 are reported, they have been used at their full value.
- The activity concentrations found in the ventilation system outlet stack have been assumed across the whole of the main ducts and sump. This is conservative as the inlet of the system would be expected to have significantly less contamination present than the outlet. This statement is supported by analysis of a series of smear, coupon and paint samples taken from the inlet and outlet stacks of the Dragon MH ventilation system in 2016 [110].
- The 2016 inventory estimate [110] also identifies that the majority of the contamination present in the ventilation system is loose contamination (determined from analysis of the metal coupons taken from both the inlet and outlet stacks). The estimate presented above conservatively assumes a low pick-up efficiency of 10% for the smears taken.
- The full-height smear is assumed to have a sample size of 300 cm^2. This surface area is used to scale the recorded count rate to the full size of the MH. In reality, the area smeared was approximately the same as the surface area of the MH (34,810 cm^2 (Table **3.38**)); if this were to be assumed the inventory estimates

for the mortuary holes component would be approximately 100 times lower. However, it is likely that the full-height smear would not have picked up all of the loose contamination that it would be expected to pick up over a smaller sample area. To account for this an area equivalent to a typical smear size has been used. This is a pessimistic approach as it is likely to significantly overestimate the inventory within each MH, noting that coupon samples from the ventilation system showed that the majority of the contamination was loose contamination. Although the fingerprint associated with the fixed component of the contamination may be different to the loose contamination measured within this assessment, this is expected to be a minor component.

- Alpha-emitting radionuclides were calculated as being present in the MH67 and MH75 fingerprints in proportions that are equivalent to those found in MH83. As the measurements of gross alpha are very low for both MH67 and MH75 (6.9 and 23.2 Bq/smear respectively), this uncertainty is of low significance and will have a small impact on the overall inventory estimate.

However, there is still some residual uncertainty with the potential to increase the inventory (INV-DRAGON-008); for example, relating to the fingerprint of fixed contamination, and the fact that no account has been taken of the potential for accumulation or increased contamination in the bottom corners of the system or in the bottom horizontal ventilation linking ducts (cross vents). This is assumed to be sufficiently covered by the adoption of an alternative inventory for the mortuary hole structure that is equivalent to the maximum total inventory presented in Table **3.41**, decayed in GoldSim and with activity concentrations calculated in the same way as for the reference inventory. The alternative inventory is presented in Table **3.43**. Assuming no changes to any other components, the contribution of the Mortuary Hole Structure increases from 0.5% to 0.7% for the alternative inventory.

Table 3.43: Alternative inventory for the Dragon Primary Mortuary Hole Structure, including a range of possible average activity concentrations, presented for an inventory reference date of 01/01/2027. The activity concentrations have been calculated assuming a density for plain carbon steel of 7,860 kg/m^3 [39, p.12-204] and a density for concrete of 2,400 kg/m^3 (Table **3.23**).

Radionuclide	Activity averaged over 1 mm of steel [Bq/g]	Activity averaged over grout infill [Bq/g]	Activity averaged over pit volume [Bq/g]	Total primary mortuary hole activity [MBq]
^{3}H	9.94E-02	3.84E-03	4.43E-04	2.50E-01
^{14}C	3.83E-01	1.48E-02	1.71E-03	9.62E-01
^{55}Fe	2.95E-03	1.14E-04	1.31E-05	7.41E-03
^{63}Ni	1.52E-02	5.90E-04	6.79E-05	3.83E-02
^{60}Co	8.08E-03	3.12E-04	3.60E-05	2.03E-02
^{90}Sr	4.54E+00	1.76E-01	2.02E-02	1.14E+01
^{137}Cs	1.05E+01	4.07E-01	4.69E-02	2.64E+01
^{210}Pb	3.59E-10	1.39E-11	1.60E-12	9.02E-10
^{226}Ra	1.04E-08	4.01E-10	4.62E-11	2.61E-08

Radionuclide	Activity averaged over 1 mm of steel [Bq/g]	Activity averaged over grout infill [Bq/g]	Activity averaged over pit volume [Bq/g]	Total primary mortuary hole activity [MBq]
^{228}Ra	3.30E-15	1.28E-16	1.47E-17	8.30E-15
^{227}Ac	9.73E-09	3.76E-10	4.33E-11	2.44E-08
^{228}Th	1.07E-15	4.12E-17	4.75E-18	2.68E-15
^{229}Th	1.42E-04	5.50E-06	6.34E-07	3.57E-04
^{230}Th	1.39E-05	5.36E-07	6.17E-08	3.48E-05
^{232}Th	1.81E-14	7.00E-16	8.07E-17	4.55E-14
^{231}Pa	1.83E-07	7.09E-09	8.17E-10	4.61E-07
^{233}U	4.36E-01	1.68E-02	1.94E-03	1.09E+00
^{234}U	4.36E-01	1.68E-02	1.94E-03	1.09E+00
^{235}U	2.51E-03	9.70E-05	1.12E-05	6.30E-03
^{236}U	1.06E-04	4.10E-06	4.73E-07	2.67E-04
^{238}U	2.45E-03	9.48E-05	1.09E-05	6.16E-03
^{237}Np	5.40E-07	2.09E-08	2.41E-09	1.36E-06
^{238}Pu	5.11E-01	1.98E-02	2.28E-03	1.28E+00
^{239}Pu	6.94E-02	2.68E-03	3.09E-04	1.74E-01
^{240}Pu	6.94E-02	2.68E-03	3.09E-04	1.74E-01
^{241}Pu	1.34E+00	5.18E-02	5.96E-03	3.36E+00
^{241}Am	4.85E-01	1.88E-02	2.16E-03	1.22E+00
^{243}Am	1.42E-08	5.49E-10	6.32E-11	3.57E-08
^{243}Cm	1.74E-02	6.75E-04	7.77E-05	4.38E-02
^{244}Cm	1.55E-02	6.00E-04	6.91E-05	3.90E-02
Total	**1.90E+01**	**7.33E-01**	**8.44E-02**	**4.76E+01**

Note that the inventory estimates presented in Section 3.9.3 and 3.9.4 assume that there has been negligible radionuclide migration into the bulk steel concrete, given that the mortuary holes, sump and storage pit are steel-lined. It is possible that migration could have occurred at the steel joints, although there is no evidence for this, and it is listed as an uncertainty in Appendix A (INV-DRAGON-009).

3.10 Dragon Reactor Complex Inventory Summary

3.10.1 Summary Tables

505 Estimates for the maximum and average activity concentrations and radiological inventory for the different Dragon features have been compiled from the previous sections and are presented in Table **3.44**, Table **3.45** and Table **3.46**, respectively.

Table 3.44: Dragon maximum activity concentrations (Bq/g) summary presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted in red.

Radio-nuclide	Maximum Activity Concentration [Bq/g]					
	Below cutline Bioshield	Below cutline B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure (averaged over 1 mm steel)	Below cutline B78 Building Contamination
^{3}H	2.31E+01	7.04E+01	9.45E+01	7.04E+01	7.46E-02	7.04E+01
^{14}C	1.16E-01	5.09E-01	6.73E+00	5.09E-01	1.87E-01	5.09E-01
^{36}Cl						
^{41}Ca	1.66E-01					
^{55}Fe					1.84E-03	
^{60}Co	3.11E-02	3.47E-01	7.73E-01	3.47E-01	1.02E-02	3.47E-01
^{63}Ni	2.99E-01	6.58E-01	1.98E+00	6.58E-01	6.22E-03	6.58E-01
^{90}Sr		1.34E+01	7.02E+01	1.34E+01	3.33E+00	1.34E+01
^{137}Cs		6.81E+01	1.18E+04	6.81E+01	7.45E+00	6.81E+01
^{133}Ba	3.06E+00					
^{148}Sm	1.35E-29					
^{151}Sm	6.96E-02					
^{152}Gd	2.90E-14					
^{152}Eu	1.47E+00					
^{154}Eu	3.65E-02					
^{210}Pb		1.54E-01		1.54E-01	1.60E-10	1.54E-01
^{226}Ra		1.49E-01		1.49E-01	4.63E-09	1.49E-01
^{228}Ra		5.51E-03		5.51E-03	3.30E-15	5.51E-03
^{227}Ac		5.52E-05		5.52E-05	4.55E-09	5.52E-05
^{228}Th		5.26E-03		5.26E-03	1.07E-15	5.26E-03
^{229}Th		4.96E-13		4.96E-13	6.35E-05	4.96E-13
^{230}Th		4.82E-03		4.82E-03	6.18E-06	4.82E-03
^{232}Th		6.00E-03		6.00E-03	1.81E-14	6.00E-03
^{231}Pa		2.06E-04		2.06E-04	8.58E-08	2.06E-04
^{233}U		7.63E-10		7.63E-10	1.94E-01	7.63E-10
^{234}U		2.21E+00		2.21E+00	1.94E-01	2.21E+00
^{235}U		4.70E-01		4.70E-01	1.17E-03	4.70E-01
^{236}U					1.06E-04	
^{238}U		2.45E+00		2.45E+00	1.15E-03	2.45E+00
^{237}Np		1.70E-05		1.70E-05	4.36E-07	1.70E-05
^{238}Pu					3.32E-01	
^{239}Pu					4.49E-02	
^{240}Pu					4.49E-02	
^{241}Pu					1.11E+00	
^{241}Am		2.49E+00		2.49E+00	3.92E-01	2.49E+00
^{243}Am					1.06E-08	

Radio-nuclide	Maximum Activity Concentration [Bq/g]					
	Below cutline Bioshield	Below cutline B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure (averaged over 1 mm steel)	Below cutline B78 Building Contamination
^{243}Cm					1.30E-02	
^{244}Cm					1.13E-02	
Total	**2.84E+01**	**1.61E+02**	**1.20E+04**	**1.61E+02**	**1.34E+01**	**1.61E+02**
GBq/tonne	**2.84E-02**	**1.61E-01**	**1.20E+01**	**1.61E-01**	**1.34E-02**	**1.61E-01**

Table 3.45: Dragon average activity concentrations (Bq/g) summary presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted in red.

Radio-nuclide	Average Activity Concentration [Bq/g]					
	Below cutline Bioshield	Below cutline B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure (averaged over grout infill)	Below cutline B78 Building Contamination
^{3}H	4.86E+00	4.74E+00	9.45E+01	1.69E-01	2.89E-03	4.74E+00
^{14}C	4.70E-02	2.27E-01	6.73E+00	1.62E-03	7.21E-03	2.27E-01
^{36}Cl	2.82E-03			4.25E-05		
^{41}Ca	7.41E-02			1.12E-03		
^{55}Fe	4.01E-03			6.04E-05	7.13E-05	
^{60}Co	1.05E-02	5.42E-02	7.73E-01	3.89E-04	3.96E-04	5.42E-02
^{63}Ni	1.66E-01	1.58E-01	1.98E+00	7.71E-03	2.41E-04	1.58E-01
^{90}Sr		3.88E+00	7.02E+01	2.66E-02	1.29E-01	3.88E+00
^{137}Cs	5.29E-05	2.83E+00	1.18E+04	7.21E-02	2.88E-01	2.83E+00
^{133}Ba	2.16E-01			3.26E-03		
^{148}Sm	4.14E-30			6.24E-32		
^{151}Sm	2.10E-02			3.17E-04		
^{152}Gd	8.91E-15			1.34E-16		
^{152}Eu	4.51E-01			6.80E-03		
^{154}Eu	1.24E-02			1.86E-04		
^{210}Pb		6.48E-02		1.75E-05	6.19E-12	6.48E-02
^{226}Ra		1.24E-01		3.34E-05	1.79E-10	1.24E-01
^{228}Ra		1.65E-03		4.43E-07	1.28E-16	1.65E-03
^{227}Ac		2.82E-05		7.59E-10	1.76E-10	2.82E-05
^{228}Th		1.26E-03		3.40E-07	4.12E-17	1.26E-03
^{229}Th		3.16E-14		8.51E-18	2.45E-06	3.16E-14
^{230}Th		3.15E-03		8.50E-07	2.39E-07	3.15E-03
^{232}Th		2.52E-03		6.78E-07	7.00E-16	2.52E-03

Radio-nuclide	Average Activity Concentration [Bq/g]					
	Below cutline Bioshield	Below cutline B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure (averaged over grout infill)	Below cutline B78 Building Contamination
^{231}Pa		2.20E-05		5.94E-09	3.32E-09	2.20E-05
^{233}U		1.15E-10		3.11E-14	7.51E-03	1.15E-10
^{234}U		3.73E-01		5.19E-04	7.51E-03	3.73E-01
^{235}U		1.18E-01		2.41E-04	4.54E-05	1.18E-01
^{236}U					4.10E-06	
^{238}U		4.61E-01		2.63E-03	4.44E-05	4.61E-01
^{237}Np		6.15E-06		1.66E-09	1.69E-08	6.15E-06
^{238}Pu				1.82E-04	1.29E-02	
^{239}Pu				9.10E-04	1.74E-03	
^{240}Pu				1.27E-03	1.74E-03	
^{241}Pu				3.60E-03	4.29E-02	
^{241}Am		2.15E+00		2.49E-03	1.52E-02	2.15E+00
^{243}Am					4.10E-10	
^{243}Cm					5.04E-04	
^{244}Cm				6.31E-04	4.37E-04	
Total	**5.86E+00**	**1.52E+01**	**1.20E+04**	**3.02E-01**	**5.18E-01**	**1.52E+01**
GBq/tonne	**5.86E-03**	**1.52E-02**	**1.20E+01**	**3.02E-04**	**5.18E-04**	**1.52E-02**

Table 3.46: Dragon disposal inventory (MBq) summary presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted in red.

Radio-nuclide	Total Activity [MBq]								
	Below ground Bioshield	Below ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{3}H	1.25E+03	7.93E+02	7.48E+00	2.17E+03	1.88E-01	3.82E+01	4.23E+03	3.84E+01	4.26E+03
^{14}C	1.21E+01	4.26E-01	5.33E-01	2.09E+01	4.69E-01	4.04E-02	3.42E+01	5.09E-01	3.47E+01
^{36}Cl	7.27E-01			5.47E-01			1.27E+00	0.00E+00	1.27E+00
^{41}Ca	1.91E+01			1.44E+01			3.34E+01	0.00E+00	3.34E+01
^{55}Fe	1.03E+00			7.77E-01	4.63E-03		1.81E+00	4.63E-03	1.81E+00
^{60}Co	2.71E+00	1.02E-01	6.12E-02	5.01E+00	2.57E-02	9.68E-03	7.70E+00	3.54E-02	7.73E+00
^{63}Ni	4.27E+01	2.98E-01	1.57E-01	9.93E+01	1.56E-02	2.82E-02	1.38E+02	4.39E-02	1.38E+02
^{90}Sr		7.30E+00	5.56E+00	3.43E+02	8.37E+00	6.93E-01	3.33E+02	9.06E+00	3.42E+02
^{137}Cs	1.36E-02	5.32E+00	9.37E+02	9.28E+02	1.87E+01	5.05E-01	1.81E+03	1.92E+01	1.83E+03
^{133}Ba	5.57E+01			4.19E+01			9.76E+01	0.00E+00	9.76E+01
^{148}Sm	1.07E-27			8.03E-28			1.87E-27	0.00E+00	1.87E-27
^{151}Sm	5.42E+00			4.08E+00			9.50E+00	0.00E+00	9.50E+00
^{152}Gd	2.29E-12			1.73E-12			4.02E-12	0.00E+00	4.02E-12
^{152}Eu	1.16E+02			8.75E+01			2.04E+02	0.00E+00	2.04E+02
^{154}Eu	3.19E+00			2.40E+00			5.59E+00	0.00E+00	5.59E+00
^{210}Pb		1.22E-01		2.25E-01	4.02E-10	1.16E-02	3.47E-01	1.16E-02	3.58E-01
^{226}Ra		2.33E-01		4.30E-01	1.16E-08	2.21E-02	6.63E-01	2.21E-02	6.85E-01
^{228}Ra		3.09E-03		5.71E-03	8.30E-15	2.94E-04	8.80E-03	2.94E-04	9.10E-03
^{227}Ac		5.30E-06		9.77E-06	1.14E-08	5.03E-07	1.51E-05	5.14E-07	1.56E-05

Radio-nuclide	Total Activity [MBq]								
	Below ground Bioshield	Below ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{228}Th		2.38E-03		4.38E-03	2.68E-15	2.25E-04	6.76E-03	2.25E-04	6.98E-03
^{229}Th		5.94E-14		1.10E-13	1.59E-04	5.64E-15	1.69E-13	1.59E-04	1.59E-04
^{230}Th		5.93E-03		1.09E-02	1.55E-05	5.63E-04	1.69E-02	5.78E-04	1.75E-02
^{232}Th		4.73E-03		8.73E-03	4.55E-14	4.49E-04	1.35E-02	4.49E-04	1.39E-02
^{231}Pa		4.14E-05		7.64E-05	2.15E-07	3.93E-06	1.18E-04	4.15E-06	1.22E-04
^{233}U		2.17E-10		4.00E-10	4.88E-01	2.06E-11	6.17E-10	4.88E-01	4.88E-01
^{234}U		7.02E-01		6.68E+00	4.88E-01	6.66E-02	7.38E+00	5.55E-01	7.93E+00
^{235}U		2.23E-01		3.10E+00	2.95E-03	2.11E-02	3.32E+00	2.41E-02	3.35E+00
^{236}U					2.67E-04		0.00E+00	2.67E-04	2.67E-04
^{238}U		8.68E-01		3.39E+01	2.88E-03	8.23E-02	3.48E+01	8.52E-02	3.49E+01
^{237}Np		1.16E-05		2.13E-05	1.10E-06	1.10E-06	3.29E-05	2.19E-06	3.51E-05
^{238}Pu				2.35E+00	8.35E-01		2.18E+00	8.35E-01	3.02E+00
^{239}Pu				1.17E+01	1.13E-01		1.09E+01	1.13E-01	1.10E+01
^{240}Pu				1.63E+01	1.13E-01		1.52E+01	1.13E-01	1.53E+01
^{241}Pu				4.63E+01	2.79E+00		4.31E+01	2.79E+00	4.59E+01
^{241}Am		4.04E+00		3.21E+01	9.85E-01	3.83E-01	3.51E+01	1.37E+00	3.65E+01
^{243}Am					2.66E-08		0.00E+00	2.66E-08	2.66E-08
^{243}Cm					3.27E-02		0.00E+00	3.27E-02	3.27E-02
^{244}Cm				8.13E+00	2.84E-02		7.56E+00	2.84E-02	7.59E+00
Total	**1.51E+03**	**8.12E+02**	**9.50E+02**	**3.88E+03**	**3.37E+01**	**4.01E+01**	**7.06E+03**	**7.37E+01**	**7.13E+03**

506 For each component, the highest maximum activity concentration is 11,825 Bq/g for ^{137}Cs in the residual contamination from the PGPC contaminated water spill, followed by 70.4 Bq/g for ^{3}H in the building contamination and backfill and 23.1 Bq/g for ^{3}H in the bioshield (Table **3.44**). Many of the maximum activity concentrations are associated with the building contamination activity, reflecting the encompassing nature of the fingerprint derived.

507 Residual contamination from the PGPC contaminated water spill and building contamination also dominate the highest average activity concentrations (Table **3.45**), although some radionuclides are highest in the bioshield. This relates partly to differences in radionuclide composition, but also partly due to the depth (~0.75 m) over which elevated activities are maintained by neutron activation. In the other features the inventory is entirely related to contamination and activities fall off rapidly in the first few centimetres. The mortuary holes have the highest maximum and average activity concentration only where the radionuclides are only present in this feature (other than for ^{244}Cm and the plutonium isotopes).

508 The total estimated radionuclide inventory for Dragon is 7.23E+03 MBq, with the majority of this associated with the B70 below-ground disposal (Table **3.46**). The backfill contributes the highest proportion (54%) of the total radionuclide inventory (Figure **3.20**), followed by the below-ground bioshield (21%). The backfill dominates the inventory due to its high average activity concentrations and the large volume over which it is applied. The low average activity concentrations and low volume of the mortuary hole disposal results in a small contribution to the total inventory (0.5%).

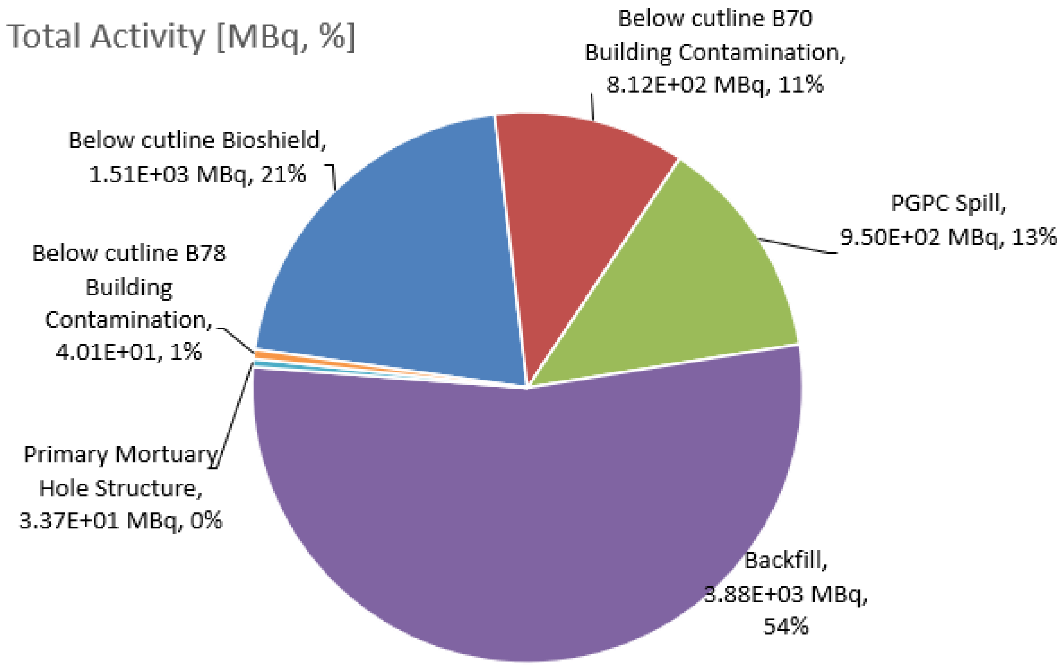


Figure 3.20: Dragon radionuclide inventory in the different features by total activity (MBq) and as a percentage for a reference date of 01/01/2027.

3.10.2 Inventory Location

509 In Figure **3.21** a plan view and cross-section of the Dragon in-situ disposal structure is illustrated along with the inventory of the in-situ features (excluding backfill). The figure shows that the majority of the in-situ inventory is located in a relatively small volume of the structure.

3.10.3 Inventory Fingerprint

510 Figure **3.22** presents pie charts illustrating the main radionuclides in each Dragon disposal feature. The figure shows that ^{3}H dominates in all the features with the exception of residual contamination from the PGPC contaminated water spill and the mortuary hole structure, which are both dominated by ^{137}Cs. The mortuary hole structure inventory also stands out due to the abundance of ^{137}Cs, ^{90}Sr and actinides including U and Pu isotopes. In the bioshield a number of radionuclides predicted by neutron activation modelling are visible, including ^{151}Sm, as well as measured activation products such as ^{152}Eu and ^{63}Ni. Note that, as tritium has a relatively short half-life of 12 years, it is of less importance than other radionuclides for long-term safety assessments.

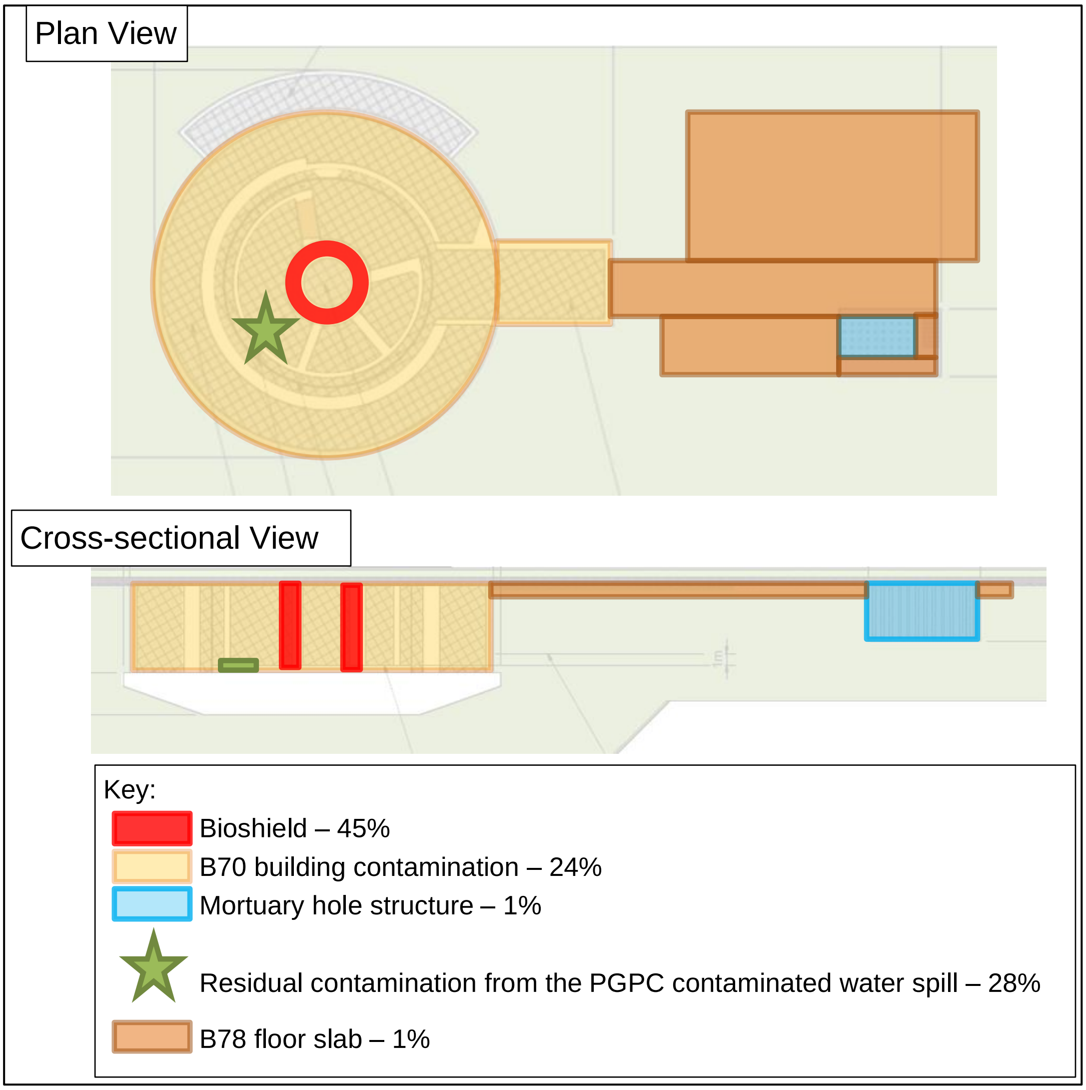


Figure 3.21: Plan and cross-sectional views of the Dragon in-situ disposal inventory by feature. The unshaded hatched area outside of the main Dragon Reactor building outline is the service duct, which is assumed to be uncontaminated. Position of the PGPC contaminated water spill is indicative. Percentage activity figures exclude the backfill. Based on [162; 104, Fig.3; 115, Fig.1; 158].

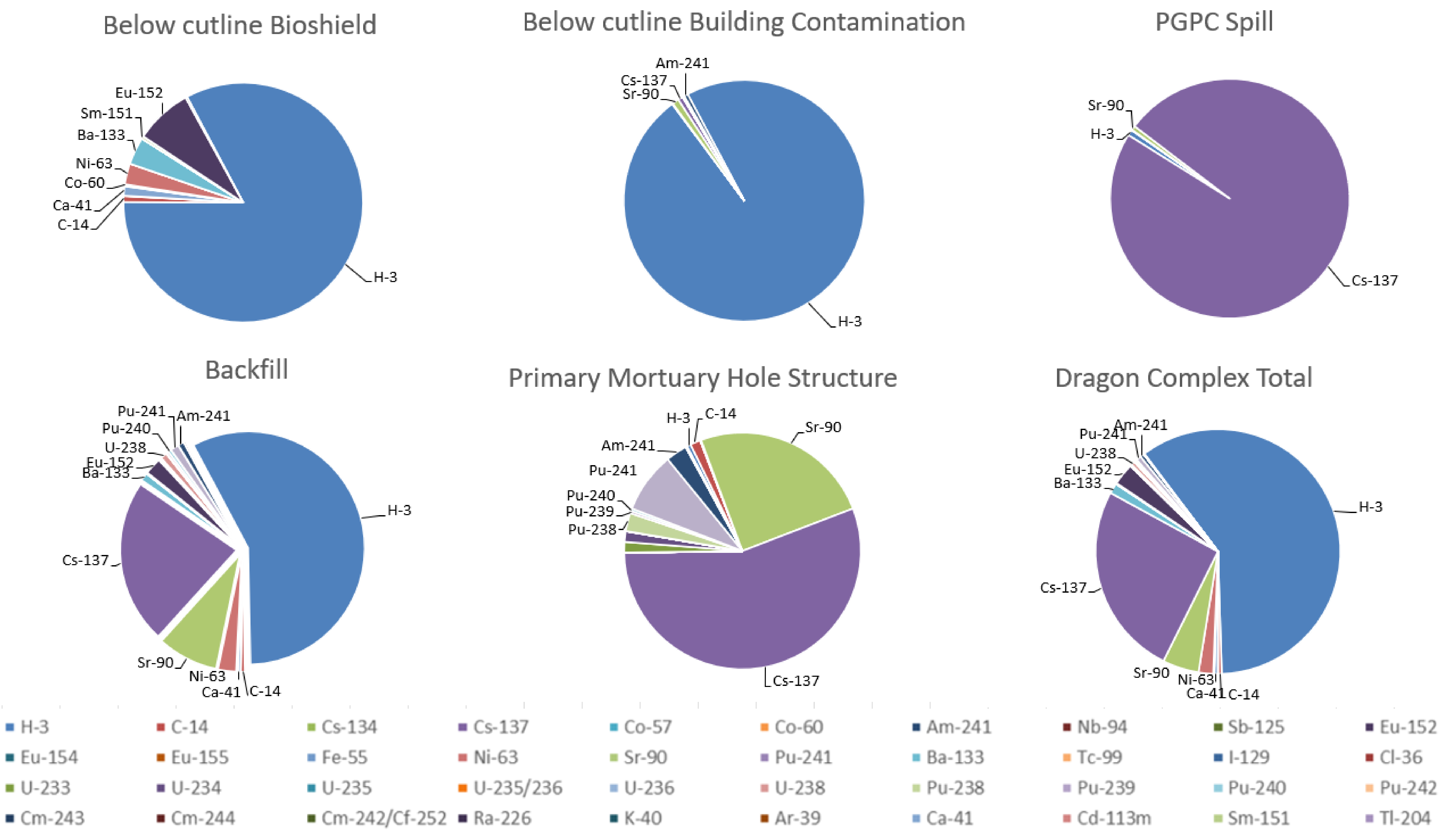


Figure 3.22: Radionuclide inventory fingerprint for Dragon disposal features at 01/01/2027. Radionuclides contributing less than 0.3% are unlabelled. *(This page is set to print on A3.)*

3.10.4 Sensitivity Analysis Summary

511 Throughout Section 3 sensitivity analyses have been used to explore and account for various uncertainties in the inventory data. Although the alternative inventories explore the impact of uncertainties, they are not considered to be realistic estimates. Table **3.47** shows the difference between the reference inventory estimate and the alternative inventories calculated in these analyses for the different Dragon components (where multiple alternatives apply, they have been combined to create a "maximum" inventory).

512 Table **3.48** presents the "maximum" alternative inventories for the main features. As previously discussed, the individual alternative inventories comprising the overall alternative inventory are considered to be pessimistic. In the overall alternative inventory, the total Dragon inventory increases by a factor of 3.5 from 7.23E+03 MBq to 2.55E+04 MBq (less than an order of magnitude).

513 The largest relative increase is seen in the Betalite store area surface contamination, which increases by a factor of 279, with a corresponding increase in contribution to the overall Dragon inventory from 0.1% to 10%. This is due to the effect of including the anomalously high tritium result from a single paint sample in the fingerprint for this area. However, as tritium has a relatively short half-life, this scenario is not expected to have a significant impact on the long-term radiological PA.

514 The next largest relative increases are seen in the Dragon Reactor and Fuel Store Building general area surface contamination components (both in-situ and backfill portions), which are twenty times higher in the "maximum" inventory; however, these still only contribute 1.8% and 0.2% (in-situ), and 3.1% and 1.0% (backfill) respectively to the overall Dragon inventory. The only other component more than five times higher in the alternative inventory is the Betalite store area ^{3}H ingress inventory, which increases by a factor of seven, but still contributes only 5.5% to the overall Dragon inventory.

515 Table **3.49** presents a "maximum" alternative inventory using the Pu-containing fingerprint for Dragon (B70 and B78) general building contamination. In this table, the total inventories are virtually the same for every component, but the presence and distribution of radionuclides is different for the below-ground B70 and B78 contamination and the backfill. Different radionuclides behave differently in the environment and make different contributions to dose; this is an important consideration in the PA.

Table 3.47: Comparison between the main Dragon inventory estimate and the maximum inventory as explored in sensitivity analyses. % is the proportion each component makes to the overall Dragon inventory; in the maximum inventory column this considers the maximum alternative inventories for all components. Blue rows denote B70 components below ground level that will remain in-situ; orange rows denote components that will contribute to B70 backfill; green rows denote B78 components below ground level that will remain in-situ. Activity data are presented at 01/01/2027.

Component	**Reference inventory estimate**		**"Maximum" alternative inventory estimate**			
	MBq	**%**	**Changes made**	**MBq**	**Increased by factor**	**%**
Below cutline Bioshield – Portland concrete	1.11E+03	15.4	Max rather than average activity concentrations used	4.81E+03	4.3	18.9
Below cutline Bioshield – barytes concrete	3731E+02	5.2		1.55E+03	4.2	6.1
Below cutline Bioshield – rebar	2.65E+01	0.4		5.14E+01	1.9	0.2
Below cutline B70 building general area surface contamination	2.35E+01	0.3	100% contamination assumed to be present	4.70E+02	20.0	1.8
Below cutline B70 building Betalite store surface contamination	9.06E+00	0.1	100% contamination assumed to be present; high ^{3}H FP	2.53E+03	279.0	10.0
Below cutline B70 building ^{3}H ingress	5.79E+02	8.0	Max rather than average activity concentrations used	1.90E+03	3.3	7.5
Below cutline B70 building Betalite store ^{3}H ingress	2.00E+02	2.8	Max rather than average activity concentrations used; high ^{3}H FP	1.40E+03	7.0	5.5
PGPC Spill	9.50E+02	13.1	None	9.50E+02	1.0	3.7
Total in-situ B70	***3.27E+03***	***45.3***		***1.237E+04***	***4.2***	***53.8***
Above cutline Bioshield – Portland concrete	8.36E+02	11.6	Max rather than average activity concentrations used	3.62E+03	4.3	14.3
Above cutline Bioshield – barytes concrete	2.81E+02	3.9		1.17E+03	4.2	4.6
Above cutline Bioshield – rebar	1.99E+01	0.3		3.87E+01	1.9	0.2
Above cutline B70 building general area surface contamination	3.95E+01	0.5	100% contamination assumed to be present	7.90E+02	20.0	3.1
Above cutline B70 building ^{3}H ingress	9.83E+02	13.6	Max rather than average activity concentrations used	3.21E+03	3.3	12.7
Above cutline B78 building general area surface contamination	1.28E+01	0.2%	100% contamination assumed to be present	2.55E+02	20.0	1.0

Component	Reference inventory estimate		"Maximum" alternative inventory estimate			
	MBq	%	Changes made	MBq	Increased by factor	%
Above cutline B78 building ^{3}H ingress	1.77E+02	2.5%	Max rather than average activity concentrations used	7.90E+02	4.5	3.1
Rubble from stockpiles	1.53E+03	21.2%		1.68E+03	1.1	6.6
Total backfill B70	***3.88E+03***	***53.7%***		***1.16E+04***	***3.0***	***45.4***
Total B70	**7.16E+03**	**99.0%**		**2.52E+04**	**3.5**	**98.9**
Primary Mortuary Hole Structure (B78)	3.37E+01	0.5%	Max rather than average inventories of MH and cross vent components used	4.76E+01	1.4	0.2
Below cutline B78 building general area surface contamination	2.69E+00	0.0%	100% contamination assumed to be present	5.37E+02	20.0	0.2
Below cutline B78 building ^{3}H ingress	3.74E+01	0.5%	Max rather than average activity concentrations used	1.66E+02	4.5	0.7
Total in-situ B78	***7.37E+01***	***1.0%***		***2.68E+02***	***3.6***	***1.1***
Total Dragon inventory	**7.23E+03**	**100%**		**2.54E+04**	**3.5**	**100**

Table 3.48: Dragon disposal inventory (MBq) summary assuming maximum alternative inventories for each feature (as explored in sensitivity analysis, excluding the alternative Pu-containing fingerprint inventory for the building general contamination), presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted in red.

Radio-nuclide	Total Activity [MBq]								
	Below-ground Bioshield	Below-ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below-ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{3}H	5.64E+03	5.91E+03	7.48E+00	8.74E+03	2.50E-01	1.83E+02	2.03E+04	1.83E+02	2.05E+04
^{14}C	3.15E+01	8.38E+00	5.33E-01	5.24E+01	9.62E-01	8.09E-01	9.29E+01	1.77E+00	9.46E+01
^{36}Cl	1.56E+00			1.17E+00			2.73E+00	0.00E+00	2.73E+00
^{41}Ca	5.88E+01			4.43E+01			1.03E+02	0.00E+00	1.03E+02
^{55}Fe	2.43E+00			1.83E+00	7.41E-03		4.25E+00	7.41E-03	4.26E+00
^{60}Co	8.10E+00	2.01E+00	6.12E-02	1.26E+01	3.83E-02	1.94E-01	2.26E+01	2.32E-01	2.29E+01
^{63}Ni	1.14E+02	5.85E+00	1.57E-01	1.63E+02	2.03E-02	5.65E-01	2.78E+02	5.85E-01	2.79E+02
^{90}Sr		1.44E+02	5.56E+00	5.99E+02	1.14E+01	1.39E+01	7.25E+02	2.53E+01	7.51E+02
^{137}Cs	1.36E-02	1.05E+02	9.37E+02	1.11E+03	2.64E+01	1.01E+01	2.09E+03	3.65E+01	2.13E+03
^{133}Ba	1.74E+02			1.31E+02			3.05E+02	0.00E+00	3.05E+02
^{148}Sm	3.29E-27			2.48E-27			5.76E-27	0.00E+00	5.76E-27
^{151}Sm	1.70E+01			1.28E+01			2.97E+01	0.00E+00	2.97E+01
^{152}Gd	7.07E-12			5.32E-12			1.24E-11	0.00E+00	1.24E-11
^{152}Eu	3.58E+02			2.70E+02			6.28E+02	0.00E+00	6.28E+02
^{154}Eu	8.89E+00			6.70E+00			1.56E+01	0.00E+00	1.56E+01
^{210}Pb		2.40E+00		4.50E+00	9.02E-10	2.31E-01	6.89E+00	2.31E-01	7.12E+00
^{226}Ra		4.58E+00		8.60E+00	2.61E-08	4.42E-01	1.32E+01	4.42E-01	1.36E+01
^{228}Ra		6.09E-02		1.14E-01	8.30E-15	5.87E-03	1.75E-01	5.87E-03	1.81E-01
^{227}Ac		1.04E-04		1.95E-04	2.44E-08	1.01E-05	3.00E-04	1.01E-05	3.10E-04

Radio-nuclide	Total Activity [MBq]								
	Below-ground Bioshield	Below-ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below-ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{228}Th		4.67E-02		8.76E-02	2.68E-15	4.51E-03	1.34E-01	4.51E-03	1.39E-01
^{229}Th		1.17E-12		2.19E-12	3.57E-04	1.13E-13	3.36E-12	3.57E-04	3.57E-04
^{230}Th		1.17E-01		2.19E-01	3.48E-05	1.13E-02	3.36E-01	1.13E-02	3.47E-01
^{232}Th		9.31E-02		1.75E-01	4.55E-14	8.98E-03	2.68E-01	8.98E-03	2.77E-01
^{231}Pa		8.15E-04		1.53E-03	4.61E-07	7.86E-05	2.34E-03	7.91E-05	2.42E-03
^{233}U		4.26E-09		8.00E-09	1.09E+00	4.11E-10	1.23E-08	1.09E+00	1.09E+00
^{234}U		1.38E+01		3.54E+01	1.09E+00	1.33E+00	4.92E+01	2.43E+00	5.17E+01
^{235}U		4.38E+00		1.30E+01	6.30E-03	4.23E-01	1.74E+01	4.29E-01	1.78E+01
^{236}U					2.67E-04		0.00E+00	2.67E-04	2.67E-04
^{238}U		1.71E+01		8.92E+01	6.16E-03	1.65E+00	1.06E+02	1.65E+00	1.08E+02
^{237}Np		2.27E-04		4.27E-04	1.36E-06	2.19E-05	6.54E-04	2.33E-05	6.77E-04
^{238}Pu				2.35E+00	1.28E+00		2.18E+00	1.28E+00	3.47E+00
^{239}Pu				1.17E+01	1.74E-01		1.09E+01	1.74E-01	1.11E+01
^{240}Pu				1.63E+01	1.74E-01		1.52E+01	1.74E-01	1.53E+01
^{241}Pu				4.63E+01	3.36E+00		4.31E+01	3.36E+00	4.65E+01
^{241}Am		7.94E+01		1.74E+02	1.22E+00	7.66E+00	2.52E+02	8.88E+00	2.61E+02
^{243}Am					3.57E-08		0.00E+00	3.57E-08	3.57E-08
^{243}Cm					4.38E-02		0.00E+00	4.38E-02	4.38E-02
^{244}Cm				8.13E+00	3.90E-02		7.56E+00	3.90E-02	7.60E+00
Total	**6.41E+03**	**6.30E+03**	**9.50E+02**	**1.16E+04**	**4.76E+01**	**2.20E+02**	**2.51E+04**	**2.68E+02**	**2.54E+04**

Table 3.49: Dragon disposal inventory (MBq) summary assuming the alternative Pu-containing fingerprint inventory for the building general contamination and maximum alternative inventories for each feature, presented for a reference date of 01/01/2027. The feature with the highest activity for each radionuclide is highlighted in red.

Radio-nuclide	Total Activity [MBq]								
	Below-ground Bioshield	Below-ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below-ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{3}H	5.64E+03	5.89E+03	7.48E+00	8.69E+03	2.50E-01	1.80E+02	2.02E+04	1.81E+02	2.04E+04
^{14}C	3.15E+01	1.28E+01	5.33E-01	6.22E+01	9.62E-01	1.31E+00	1.07E+02	2.27E+00	1.09E+02
^{36}Cl	1.56E+00	1.65E+01		3.78E+01	0.00E+00	1.88E+00	5.58E+01	1.88E+00	5.77E+01
^{41}Ca	5.88E+01			4.43E+01	0.00E+00		1.03E+02	0.00E+00	1.03E+02
^{55}Fe	2.43E+00	9.23E-02		2.03E+00	7.41E-03	1.06E-02	4.55E+00	1.80E-02	4.57E+00
^{60}Co	8.10E+00	1.42E+00	6.12E-02	1.14E+01	3.83E-02	1.27E-01	2.07E+01	1.65E-01	2.09E+01
^{63}Ni	1.14E+02	3.04E+01	1.57E-01	2.18E+02	2.03E-02	3.37E+00	3.57E+02	3.39E+00	3.60E+02
^{90}Sr		8.81E+01	5.56E+00	4.75E+02	1.14E+01	7.51E+00	5.46E+02	1.89E+01	5.65E+02
^{137}Cs	1.36E-02	1.92E+02	9.37E+02	1.31E+03	2.64E+01	2.01E+01	2.37E+03	4.65E+01	2.42E+03
^{133}Ba	1.74E+02			1.31E+02	0.00E+00		3.05E+02	0.00E+00	3.05E+02
^{148}Sm	3.29E-27			2.48E-27	0.00E+00		5.76E-27	0.00E+00	5.76E-27
^{151}Sm	1.70E+01			1.28E+01	0.00E+00		2.97E+01	0.00E+00	2.97E+01
^{152}Gd	7.07E-12			5.32E-12	0.00E+00		1.24E-11	0.00E+00	1.24E-11
^{152}Eu	3.58E+02	1.30E+01		2.99E+02	0.00E+00	1.49E+00	6.70E+02	1.49E+00	6.71E+02
^{154}Eu	8.89E+00	1.23E+00		9.42E+00	0.00E+00	1.40E-01	1.95E+01	1.40E-01	1.97E+01
^{210}Pb		3.76E-01		1.56E-08	9.02E-10	8.02E-10	3.76E-01	1.70E-09	3.76E-01
^{226}Ra		7.18E-01		1.83E-07	2.61E-08	9.41E-09	7.18E-01	3.55E-08	7.18E-01
^{228}Ra		9.16E-01		2.02E+00	8.30E-15	1.04E-01	2.93E+00	1.04E-01	3.04E+00
^{227}Ac		3.43E-05		3.99E-05	2.44E-08	2.05E-06	7.42E-05	2.08E-06	7.63E-05
^{228}Th		7.03E-01		1.55E+00	2.68E-15	7.96E-02	2.25E+00	7.96E-02	2.33E+00

Radio-nuclide	Total Activity [MBq]								
	Below-ground Bioshield	Below-ground B70 Building Contamination	PGPC Spill	Backfill	Primary Mortuary Hole Structure	Below-ground B78 Building Contamination	B70 Building Sub-total	B78 Building Sub-total	Dragon Complex Total
^{229}Th		2.18E-13		7.84E-14	3.57E-04	4.03E-15	2.97E-13	3.57E-04	3.57E-04
^{230}Th		1.83E-02		9.60E-05	3.48E-05	4.94E-06	1.84E-02	3.97E-05	1.85E-02
^{232}Th		1.40E+00		3.08E+00	4.55E-14	1.59E-01	4.48E+00	1.59E-01	4.64E+00
^{231}Pa		2.68E-04		3.12E-04	4.61E-07	1.61E-05	5.80E-04	1.65E-05	5.97E-04
^{233}U		7.98E-10		2.89E-10	1.09E+00	1.49E-11	1.09E-09	1.09E+00	1.09E+00
^{234}U		2.70E+00		1.07E+01	1.09E+00	6.11E-02	1.34E+01	1.16E+00	1.46E+01
^{235}U		1.44E+00		6.44E+00	6.30E-03	8.63E-02	7.88E+00	9.26E-02	7.97E+00
^{236}U					2.67E-04		0.00E+00	2.67E-04	2.67E-04
^{238}U		3.61E+00		5.92E+01	6.16E-03	1.07E-01	6.28E+01	1.13E-01	6.30E+01
^{237}Np		4.27E-05		2.65E+00	1.36E-06	8.09E-07	2.65E+00	2.17E-06	2.65E+00
^{238}Pu		1.19E+00		3.47E+00	1.28E+00	1.36E-01	4.50E+00	1.42E+00	5.92E+00
^{239}Pu		5.07E-01		1.25E+01	1.74E-01	5.80E-02	1.22E+01	2.32E-01	1.25E+01
^{240}Pu		3.74E-01		5.23E+01	1.74E-01	4.28E-02	5.15E+01	2.17E-01	5.17E+01
^{241}Pu		1.62E+01		4.63E+01	3.36E+00	1.85E+00	5.93E+01	5.22E+00	6.45E+01
^{241}Am		1.50E+01		3.04E+01	1.22E+00	2.97E-01	4.44E+01	1.52E+00	4.60E+01
^{243}Am					3.57E-08		0.00E+00	3.57E-08	3.57E-08
^{243}Cm					4.38E-02		0.00E+00	4.38E-02	4.38E-02
^{244}Cm				8.13E+00	3.90E-02		7.56E+00	3.90E-02	7.60E+00
Total	**6.41E+03**	**6.29E+03**	**9.50E+02**	**1.15E+04**	**4.76E+01**	**2.19E+02**	**2.51E+04**	**2.67E+02**	**2.54E+04**

4 Confidence in the Inventory

516 As discussed in the preceding sections, the inventory estimate for the two reactor complexes is based on datasets that may not be fully comprehensive, statistically significant or accompanied by a DQO process. There is limited information for some components and in many cases access limitations prevent further sampling and characterisation at this time. Assumptions have been made to allow an inventory estimate to be made, as documented in Appendix A.

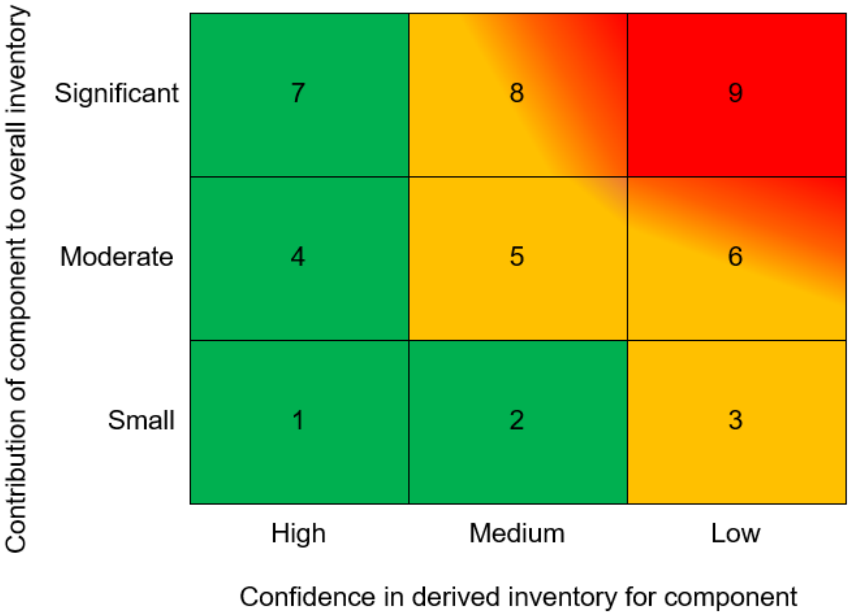
517 The purpose of this section is to summarise the approaches used to derive an inventory estimate for each reactor complex component and to present an assessment of relative confidence in the estimates.

4.1 RAG Approach

518 Table **4.1** and Table **4.2** summarise, for SGHWR and Dragon respectively, the characterisation data, inventory derivation approaches, uncertainties and overall confidence in the inventory (and significance of this) for components of each reactor complex considered in this report.

519 Red-Amber-Green (RAG) colour coding is used to indicate at a glance:

1) Whether an inventory is derived for each component (Green: inventory derived, Red: no inventory derived).
2) The comprehensiveness of characterisation data supporting the inventory, based on a qualitative judgement (Green: comprehensive, Amber: limited, Red: none).
3) The confidence in inventory derivation approach, based on a qualitative judgement (Green: high, Amber: moderate, Red: low).
4) The overall confidence in the inventory derived for each component and how significant this is as a function of the total SGHWR or Dragon inventory, based on Figure **4.1** and the following explanation. The overall RAG score and supporting discussion takes account of any uncertainty relating to radionuclides that could drive significant impacts in the PA.



Confidence in derived inventory: a qualitative judgement based on the RAG scores and reasoning set out in the characterisation data and inventory derivation approach columns.

Contribution of component to the overall inventory:

- Significant: >25%
- Moderate: 5-25%
- Small: <5%

Figure 4.1: Diagram showing how an overall RAG score is assigned in Table **4.1** and Table **4.2**.

An overall RAG score is assigned according to Figure **4.1** as follows:

- For components where there is high confidence in the inventory estimate, an overall RAG score of Green is assigned regardless of the contribution that component makes to the overall inventory of the reactor complex in question ***(boxes 1, 4 and 7)***.
- For components making a small contribution to the overall inventory where there is medium confidence in the inventory estimate ***(box 2)***, an overall score of Green is assigned.
- For components making a small contribution to the overall inventory where there is low confidence in the inventory estimate ***(box 3)***, and those making a moderate contribution to the overall inventory where there is medium confidence in the inventory estimate ***(box 5)***, an overall score of Amber is assigned.
- For components making a significant contribution to the overall inventory where there is medium confidence in the inventory estimate ***(box 8)***, and those making a moderate contribution to the overall inventory where there is low confidence in the inventory estimate ***(box 6)***, there is potential for an overall

score of either Amber or Red to be assigned. Which score is assigned for any given component is a judgement based on how close to the upper bound of each category the component is considered to lie.

- For components making a significant contribution to the overall inventory where there is low confidence in the inventory estimate ***(box 9)***, an overall score of Red is assigned.

521 The overall RAG score is intended to allow high-level identification of where the main uncertainties are in the overall inventory estimate presented in this report, and to act, alongside PA results, as an approximate guide to where future characterisation effort should be directed to reduce these uncertainties.

4.2 Confidence in the SGHWR Inventory

Table 4.1: Summary of the inventory derivation approaches and relative confidence for SGHWR components considered in this inventory (see Section 4.1 for full explanation). *(These pages are set to print on A3.)*

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Bioshield: concrete and paint	No statistically representative sampling campaign undertaken. Two cores available, one originating from the LSD plant room (334) containing concrete and paint and the other from the Active Tools Store (room 245) containing concrete, rebar and samples of the flexcell joint [43; 45].	Activation modelling [44], FP-028[1]	The derivation of the bioshield inventory is discussed in Section 2.10. The concrete activation estimate for this component was derived from the activity values of the two bioshield cores. For radionuclides not in the sampling analytical suite, activities have been derived by scaling the fingerprint values derived from activation modelling [44] to the ^{60}Co activity measured on the inner face of the bioshield. Inventory contribution from activation is assumed up to the flexcell joint, consistent with the core data. The inventory arising from paint and concrete contamination is added to the activation inventory for the bioshield. Activities for paint are derived from a single sample from the inner edge of the bioshield, while the contamination inventory is derived by scaling FP-028 (SGHWR Primary External Contamination) to the measured ^{137}Cs activity in contaminated sections of the cores (on the inner face of the bioshield and flexcell joint). A contamination depth of 20 mm is assumed for concrete.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-004 (extent of bioshield activation) INV-SGHWR-005 (poor fit to SGHWR activation modelling) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Medium confidence** in inventory as measurements of the component are expected to be bounding (the cores were taken from areas of expected high neutron flux), but might not be representative of the entire structure. There is fairly low confidence in the accuracy of the activation modelling used to derive the fingerprint, but this is only used where radionuclides have not been measured directly. Very conservative assumptions and simplifications are therefore adopted in the derivation of the inventory. **Significant contribution** to inventory as the bioshield contributes over 50% of the total inventory (of which all is in-situ). **Therefore, overall RAG score is Amber (box 8)**.	Mass: 7.37E+05 kg Volume: 3.07E+02 m^3 100% is to be disposed of in-situ. (Note activation and contamination inventories were estimated separately so there is some conservative double counting of material volume and mass).	Alternative inventory derived by assuming activation activities in line with the activation modelling.
Yes	Bioshield: rebar	Three samples from the Active Tools Store (room 245) core, but with low/LOD reported activities and located relatively deep into the bioshield / away from the more activated region.	Activation modelling [44]	The derivation of the bioshield activation inventory is discussed in Section 2.10. The inventory is derived from activation modelling only due to limited rebar measurement data. The modelled activity was scaled based on the ratio between measured and modelled activation of the concrete. The volume of rebar in the bioshield is not known; it is assumed that rebar comprises 3% of the bioshield by volume (the concrete activity calculation assumes the bioshield is 100% concrete by volume, conservatively resulting in double counting of volume).	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-004 (extent of bioshield activation) INV-SGHWR-005 (poor fit to SGHWR activation modelling) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Low confidence** as there are limited supporting measurements. The inventory is entirely based on modelling data. The volume of rebar in the bioshield is also unknown. **Small contribution** to inventory, rebar forms <1% of the bioshield inventory. FP is generally composed of short-lived radionuclides so has lower impact on long-term safety assessments. **Therefore, overall RAG score is Amber (box 3)**.	Mass: 5.22E+04 kg Volume: 6.64E+00 m^3 100% is to be disposed of in-situ.	Alternative inventory derived by assuming activation activities predicted by the activation modelling.
Yes	Mortuary Tubes	None	Activation modelling [44], FP-034, FP-030, Primary circuit pipework [25]	The derivation of the mortuary tubes inventory is discussed in Section 2.11. In the absence of sample data for the mortuary tubes a high-level approach has been taken to account for the potential sources of activity. The activity of the tube liners is considered equal to the sum of: • The primary circuit pipework fingerprint at the activity of the contaminated run. • The moderator circuit fingerprint at the activity of the moderator circuit contaminated run. • The ponds fingerprint at the activity of the fuel pond liner. • The rebar activation fingerprint at the activation modelling activity of 100 g of activated fuel channel tube. • The rebar activation fingerprint at the activation modelling rebar activity in the 18' to 38' radial interval.	INV-SGHWR-011 (lack of SGHWR mortuary tubes characterisation data)	**Low confidence** as there are no supporting measurements; derived inventory is very speculative. Conservative approach to inventory estimate taken (more conservative fingerprint considered in alternative inventory). **Small contribution** to the overall inventory (~1% of SGHWR inventory), although this is highly dependent on assumptions and could increase to **moderate**. **Therefore, overall RAG score is Red (box 6)**.	Mass: 2.75E+03 kg Volume: 3.50E-01 m^3 100% is to be disposed of in-situ.	Adopt Zircaloy activation fingerprint from activation modelling.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Primary containment: main space internal surfaces	Samples from paint and concrete in walls and floor of basement structure (Room 111) - 8 cores in 2005 targeting both areas expected to be clean and those expected to be contaminated [43, Tab.9; 54] and 17 cores in 2019 [55; 56] (the latter specifically to support the end state, targeting areas of expected contamination). However, no samples from elsewhere in the primary containment structure.	FP-028[1]	Section 2.12 discusses the derivation of the primary containment inventory. Inventory derived from an average of the decay-corrected 2005 and 2019 samples for Room 111 [54]. Missing radionuclides scaled according to ratios to ^{137}Cs, ^{60}Co, ^{241}Am or ^{235}U as appropriate, with ratios from fingerprint FP-028. Data from Room 111 conservatively assumed to apply to the top 150 mm of the walls and floors of the entire Primary Containment main space.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-009 (ongoing and future activities could lead to further contamination)	**Medium confidence** as although the sample data are not representative of the entire structure, the adoption of activities from the basement structure is expected to be pessimistic. FP-028 is derived for ancillary pipework rather than structural concrete. **Moderate contribution** to the overall inventory (~8%). **Therefore, overall RAG score is Amber (box 5)**.	Mass: 7.24E+05 kg Volume: 3.03E+02 m^3 67% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Primary containment: Bulk concrete ^{3}H	Deeper intervals of primary containment concrete are sampled by the two cores targeting the bioshield (which first pass through the primary containment). Respectively taken from the LSD plant room (334) and the Active Tools Store (245) [43; 45].	None.	Derived to account for tritium activity not captured by the surface samples. The primary containment walls and floor are assumed to be uniformly contaminated with tritium to a depth of 5' (the nominal primary containment wall thickness; 1,524 mm). Inventory derived assuming that tritium activity is equal to the average activity in the first 5' of each bioshield core.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-009 (ongoing and future activities could lead to further contamination)	**Medium confidence** as sample data are not representative of the entire structure. **Moderate contribution** to the overall inventory (~5%). **Therefore, overall RAG score is Amber (box 5)**.	Mass: 6.61E+06 kg Volume: 2.75E+03 m^3 67% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Primary containment: Sump in Room 111	One core from the floor of the octagonal sump with limited analytical suite [55].	FP-028	The sump in the basement of the primary containment structure (Room 111) collected active effluent to feed to active drainage. The component consists of an octagonal sump, a duct and a square sump. A single core was taken from the octagonal sump and applied to the entire sub-structure. Certain radionuclides missing from the analytical suite were inferred from other cores elsewhere in the primary containment.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Medium confidence** due to the limited analytical suite for the core from the sump. **Small contribution** to the overall inventory (< 1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 2.35E+04 kg Volume: 9.81E+00 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Primary containment: Cluster loop room and two element loop rooms	None	FP-028	For the derivation of the inventory for the Cluster loop room (612), it was assumed that the contamination is equal to that of the LSD plant room, which shares a contamination pathway (primary circuit) and should be pessimistic as the LSD plant room is one of the more contaminated process plant rooms. The two element loop was an experimental circuit that was never used. No inventory is derived for the two element loop room (611) as it did not house any active circuitry or operations.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-009 (ongoing and future activities could lead to further contamination) INV-SGHWR-010 (uncharacterised rooms)	**Low confidence** as the inventory is either not derived or extrapolated from similar areas. **Small contribution** to the overall inventory (< 1%) from cluster loop room. Two element loop room is expected to be less contaminated than cluster loop. **Therefore, overall RAG score is Amber (box 3).**	Mass: 2.59E+05 kg Volume: 1.08E+02 m^3 0% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Secondary containment: Effluent (124), Delay and Sludge tank rooms (125 & 126)	9 cores and 3 chipping samples from rooms 124 and 126 [73; 74]. No statistically representative sampling campaign undertaken.	FP-003	The inventory derivation approach for components in the secondary containment is discussed in Section 2.13. Inventory derived based on an average of cores and samples from Rooms 124 and 126; Room 125 was assumed to have same level of activity as adjacent Room 124. Radionuclides missing from analytical suite derived by scaling FP-003 (D60 General Area waste) to activities of relevant markers.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as the activity for Room 125 is inferred from measurements from the neighbouring room rather than measurements of the room itself. **Small contribution** to the overall inventory (< 1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.79E+05 kg Volume: 7.46E+01 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Secondary containment: Cofferdams	Total of 6 sediment samples and 9 cores drilled from walls of alternate cofferdams [63; 71].	FP-034	Cofferdams data grouped along the north (140, 141, 142, 143 & 144), east (134 - 139) and south (129 - 133 & 145) sides. Additional analytes from Coffer Dam 132 applied across all dams and remaining radionuclides scaled from FP-034 (ponds). A contamination depth of 200 mm assumed.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**High confidence** in this inventory as the sample data are expected to be representative of the whole component and the fingerprint is appropriate for the expected contamination pathway. **Small contribution** to the overall inventory (< 1%). **Therefore, overall RAG score is Green (box 1)**.	Mass: 8.00E+05 kg Volume: 3.41E+02 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Secondary containment: Process plant rooms	One core each from 2005 for the ion chamber room (247), LSD plant room (334), and neutron shield plant room (516). Two cores targeting the former area of the FCD plant room (431) [22; 58-60]. No statistically representative sampling campaign undertaken.	FP-003 FP-028	For rooms 247, 334, 516 and 431, an inventory has been derived supported by characterisation data for the individual room. Missing radionuclides were scaled from an appropriate fingerprint selected based on the process history of each area. The north transducer loop room (332) and cluster loop transducer room (628) do not have any sampling data. The inventory for both rooms adopt the activity densities for the LSD plant room as the source of contamination is the same (primary circuit); this is expected to be pessimistic as the LSD plant room is one of the more contaminated primary circuit process plant rooms. No inventory has been derived for the loop make-up room (723) as it was a clean feedwater system and the potential for contamination is low.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-009 (ongoing and future activities could lead to further contamination) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for most rooms. Areas lacking sample data are of low importance. **Moderate contribution** to the overall inventory (~8%). **Therefore, overall RAG score is Amber (box 5)**.	Mass: 4.15E+05 kg Volume: 1.73E+02 m^3 50% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Secondary containment: Moderator process areas	Single wall core from D_2O ion exchange room (336), and two cores targeting screed layer of floor of deuterising plant room floor (427) [58; 59]. No statistically representative sampling campaign undertaken.	FP-030	The inventory for Room 336 is derived from the average activity of the wall core. Room 427 no longer exists; the activity of the floor area of the former room is derived from the average of the floor core data. The final moderator process room, the D_2O hold up tank room (formerly Room 246), no longer exists and has been incorporated into the area of the ion chamber room (247). The inventory for Room 246 is assumed to be included in the inventory derived for Room 247, which captures substantial tritium contamination.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Medium confidence** as activities are derived from sampling data for all rooms. **Small contribution** to the overall inventory (<2%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.35E+05 kg Volume: 3.96E+01 m^3 95% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Secondary containment: Primary circuit process areas	Between one and six cores for each room, except for the ECW tank room (446) which has none [58; 63].	FP-003 FP-026	For majority of rooms (240, 326/2, 328, 329, 330) the inventory is derived based on the average activity of the sampling data for the paint and concrete for each room. The activity of the ECW tank room (446) is assumed to be equal to the adjacent Feed Heater Cell (Room 330). Inventory for deaerator tank room (922) assumes activity of most active core applies to entire room.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-009 (ongoing and future activities could lead to further contamination)	**Medium confidence** as activities are derived from sampling data for all rooms. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.53E+06 kg Volume: 6.40E+02 m^3 50% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Secondary containment: Pond Clean Up Areas	Cores taken prior to decontamination for each room in the group; 3 cores from each of Rooms 222, 224 and 225, 20 cores from Room 223, and 10 cores from Room 228 [66; 72].	FP-018	Only characterisation data prior to decontamination are available. Activities are derived from the average paint and concrete activities for each room (222, 223, 224, 225, 228) and the duct in Rooms 224/228. In decontaminated areas, no paint was assumed to be present. For a crack in the floor of the duct between Rooms 224 and 228, the contamination was extrapolated from the contamination profile of a nearby paint sample. Radionuclides missing from the analytical suite have been derived by scaling FP-018 (Pond Clean-up areas) to activities of relevant markers.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Medium confidence** in this inventory as activities are derived from modified pre-decontamination concrete data, not contemporary data. **Small contribution** to the overall inventory (<1%) **Therefore, overall RAG score is Green (box 2)**.	Mass: 4.87E+05 kg Volume: 2.03E+02 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Secondary containment: Ventilation system and support areas	Cores and/or chipping samples are available for each of rooms 236-238, 522-526 and 726-727 [58; 59; 68]. No statistically representative sampling campaign undertaken.	FP-003 FP-026	Inventory was derived for rooms 236-238, 522-526 and 726-727 based on the sampling data for each room. In places, ^{241}Pu, Ni, Fe and Sr results for one room are applied across rooms in the area lacking analytical data for those radionuclides. Rooms 842 and 728 lack sample data and were assumed to have activity equal to adjacent ventilation system rooms. There is no sample data for the Level 4 ventilation plant rooms and clean-up and filter beds (435-438); no inventory has been derived for these rooms.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for most rooms. Areas lacking sample data are of low importance. **Small contribution** to the overall inventory (<1%), remaining uncharacterised rooms are expected to make a small contribution. **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.13E+06 kg Volume: 4.85E+02 m^3 37% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Secondary containment: General areas	One core each from rooms 423, 433, 722, 827 and 923 [58; 59]. No statistically representative sampling campaign undertaken.	FP-003 FP-026	Inventory was derived for rooms 423, 433, 722, 827 and 923 based on the average concrete and paint activities for each room. The activities in room 924 were assumed to be equal to those in room 923. Radionuclides missing from the analytical suite have been derived by scaling FP-003 or FP-026 activities of relevant markers based on the closest match to measurement data. A number of the remaining areas in the general areas are not characterised and do not have an inventory derived. These rooms do not have a history of active processes or operations and are considered to be of low significance. The majority of these areas are airlocks (11 rooms), corridors, platforms or walkways (14 rooms) or ducts (3 rooms), with the remainder housing electrical distribution or turbine-related plant or acting as storage.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for significant rooms. Areas lacking sample data are of low importance. **Small contribution** to the overall inventory (<1%), remaining uncharacterised rooms are expected to make a small contribution. **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.12E+06 kg Volume: 4.66E+02 m^3 15% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Secondary containment: Other areas	One core each from rooms 227, 230, 231 and 322. Two cores from room 242, five cores from room 321 and eight cores for room 520 [58; 59]. Chipping samples from 836, 837, 840, 842, 844 and 835 and one core from 836.	FP-003 FP-026 FP-038	The inventory for rooms 227, 230, 231, 242, 321 and 322 was derived based on the average paint and concrete sample data for each room. The inventory for the maintenance and decontamination pit (520) also includes an inventory for its fibreglass liner, assuming a thickness of 3 mm. For the Instrument active workshops on Level 8 and related areas (836, 837, 838, 840, 841, 844 845 and 846), the inventory was derived based on a core from room 836 and chipping samples from various rooms. The rooms lacking ample data are expected to be the least active so this should be conservative.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for all rooms. **Small contribution** to the overall inventory (~1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.27+06 kg Volume: 5.36E+02 m^3 81% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Ponds	A total of 126 wall cores and 17 floor cores from ponds; a sampling plan was followed based on non-parametric random sampling for the floors and systematic sampling for the walls supported by targeted sampling of areas of interest [79; 83].	FP-034	The inventory derivation approach for the SGHWR ponds is discussed in Section 2.14. Fibreglass liner and concrete activities were derived from the average sample measurement for each pond, assuming a liner thickness of 3 mm and a contamination depth of 200 mm in the concrete. Additional inventory was included from contaminated cracks and construction joints that were targeted by a subset of the wall cores.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**High confidence** in this inventory as the activities are derived from representative sampling data, the dimensions of the ponds are well known and the adopted fingerprint is derived specifically for this area. **Small contribution** to the overall inventory (~2%). **Therefore, overall RAG score is Green (box 1)**.	Mass: 1.17E+06 kg Volume: 4.87E+02 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated by adopting more pessimistic dimensional assumptions.
Yes	Ancillary areas: Active workshops (251-252)	A large-scale monitoring campaign was undertaken to develop a contamination map that was used to identify preferred sample locations for coring. A total of 19 floor and wall cores were taken, including paint samples [84].	FP-016*	The inventory derivation approach for the ancillary areas is discussed in Section 2.15. The active workshops inventory includes all external walls of the workshops and the low dividing brick walls within the spaces. Missing radionuclide activities were scaled from FP-016, which is derived for the workshops based on a sample from a hotspot.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**High confidence** in this inventory as the activities are derived from extensive sampling data. The adopted fingerprint is derived specifically for this area. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 1)**.	Mass: 4.82E+05 kg Volume: 2.02E+02 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Ancillary areas: ACW system	A single historical core from Room 256 [58]. No statistically representative sampling campaign undertaken.	FP-003	The ACW system rooms consist of the ACW pump house basement (256), ACW pump house (484) and ACW switch room (483/1). The inventories for these rooms have been derived from a single historical core from Room 256 with activities of missing radionuclides scaled from FP-003 (D60 General Area waste). The ACW system was a clean feedwater system so there is low potential for contamination.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** in the inventory for this area as results are based on a single core and a generic fingerprint was adopted. Rooms should have low contamination. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 6.10E+05 kg Volume: 2.55E+02 m^3 44% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Ancillary areas: Boiler House Basement, Fuel Oil Tank room and Cooling Water Washout Pit	At least one core and chipping sample taken from each room [68]. No statistically representative sampling campaign undertaken.	FP-026	For Rooms 253, 254 and 258 the inventory is derived separately from core and chipping samples from each room. Missing radionuclides were derived from FP-026 (SGHWR Off-Gas Beds). For Ni, Fe and Sr, the results obtained for the Fuel Oil Tank room (Room 253) were applied to the other rooms in this area as the rooms share a common source term. Room 323 forms a sub-structure within Room 324; data were amalgamated from across both areas and treated as a single area.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness)	**Medium confidence** as activities are derived from sampling data for all rooms. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 5.76E+05 kg Volume: 2.40E+02 m^3 100% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Mixed	Ancillary areas: Ventilation system and support areas	One core each from rooms 559/560, 859 and 852. Chipping samples from room 559/560 and various CCR plant rooms [85]. No statistically representative sampling campaign undertaken.	FP-026	The inventories for each of the CCR vent plant and equipment rooms on Level 8 (852, 853, 854, 855 and 859) were derived from the average activities of samples from the area. Inventories for the heater room/main airlet (559/560) were derived from the average activity for the room. Missing radionuclides for all rooms were derived from FP-026 (SGHWR Off-Gas Beds). There are no sample data and no inventory derived for Room 663 (but room remediated to background contamination levels).	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for grouped rooms. Areas lacking sample data are of low importance. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 1.99E+05 kg Volume: 8.77E+01 m^3 0% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Ancillary areas: North Annexe miscellaneous areas	One floor core and at least two chipping samples from each of rooms 352,353, 357, 358, 360, 361, 362. One core each from 551, 552, 951 and 952 [86; 69]. Some rooms not characterised.	FP-026	Building materials in this area are a mixture of concrete and brick, although relative fractions are not known. Inventory has been derived for each room from the sample data, it is pessimistically assumed for scaling activities that all building material has the density of concrete. Missing activities scaled from FP-026 (SGHWR Off-Gas Beds). The remaining rooms outside the secondary containment in the North Annexe are uncharacterised and consist of offices, corridors, toilets, stores and electrical facilities. These uncharacterised rooms are assumed to be inactive.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** as activities are derived from sampling data for the most significant rooms. Areas lacking sample data are of low importance. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Green (box 2)**.	Mass: 8.61E+05 kg Volume: 3.58E+02 m^3 67% by volume is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Mixed	Ancillary areas: South Annexe miscellaneous areas	One historical core from each of rooms 470, 476 and 480 and chipping samples from 485 and 439 [58; 59; 91]. Some rooms not characterised.	FP-003	For rooms 470, 476 and 480 the inventory is derived separately from the average activity of core and chipping samples from each room. The inventory for room 485 and 439 is derived from the combined data from both rooms. Missing radionuclides are derived from FP-003. The inventory for the uncharacterised Room 458 adopts the activity of the adjacent room 470. Remaining uncharacterised rooms outside the secondary containment in the South Annexe consist of the laundry and various switch rooms, toilets, labs, stores and offices. The majority of these rooms are in the office complex on Level 6. No inventory is derived for these rooms.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Low confidence** in this inventory as activities are derived for single samples for large rooms. Ongoing operations in several areas may contribute additional sources of activity that are not captured by the historical cores. **Small contribution** to the overall inventory (<1%). **Therefore, overall RAG score is Amber (box 3)**.	Mass: 1.01E+06 kg Volume: 4.22E+02 m^3 0% is to be disposed of in-situ.	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	SGHWR bulk structure	Extant radiological characterisation data for the SGHWR structure *(see rows above)*.	None	The inventory derivation approach to account for the tritium contamination in the remainder of the SGHWR bulk structure is discussed in Section 2.16. The mass is calculated from the estimated total volume of SGHWR structural materials and the volume of the remainder of the SGHWR inventory, assuming all material to be concrete. Tritium activity only is assumed to be equal to the median tritium activity of rooms in the SGHWR for which an inventory is derived.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-010 (uncharacterised rooms)	**Low confidence** in the inventory as there is no sample data for much of the contributing volume; there is also uncertainty in the total volume. Fingerprint uncertainty addressed in alternative inventory. **Moderate contribution** to the overall inventory (~5%). **Therefore, overall RAG score is Amber (box 6)**.	Mass: 4.78E+07 kg Volume: 1.99E+04 m^3 60% by volume is to be disposed of in-situ.	Alternative inventory calculated by adopting the average activity of radionuclides from the ancillary areas.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints/ other relevant information	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to SGHWR inventory (box no. refers to Figure 4.1)	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Rubble mounds	Groundhog survey and non-representative surface and near-surface sampling of mounds [93]. No statistically representative sampling campaign undertaken.	FP-004	The inventory derivation approach for the rubble mounds is discussed in Section 2.17. Activities are either from the average sample data from the mounds or based on the ratios of the A59 general fingerprint (FP-004) scaled to EPR16 OoS levels.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-007 (impact of uncertain demolition strategy)	**Medium confidence** in the inventory for this component as there are insufficient samples to consider the mounds fully characterised. There is also uncertainty about the volume that will be used as infill. **Small contribution** to the overall inventory (~4%); there is a low activity concentration but applied to a large volume. **Therefore, overall RAG score is Green (box 2)**.	Mass: 3.46E+07 kg Volume: 2.46E+04 m^3 0% is to be disposed of in-situ.	Alternative inventory calculated using maximum concentration where based on sample data.
Yes	SGHWR void backfill	N/A	N/A	The inventory derivation approach for the backfill is discussed in Section 2.17. The backfill inventory is derived from the sum of inventories derived for all contaminated volume above the demolition datum with material from the rubble mounds filling in the remaining void volume [21]. For the volume calculation it is assumed 6300 m^3 of the demolished structure is emplaced as wire-cut blocks and the remainder is in the form of compacted rubble.	INV-SGHWR-002 (uncertainties in applicability of fingerprints) INV-SGHWR-003 (uncertainties in material densities) INV-SGHWR-006 (adequateness of characterisation data and lack of statistical robustness) INV-SGHWR-007 (impact of uncertain demolition strategy) INV-SGHWR-009 (ongoing and future activities could lead to further contamination) INV-SGHWR-010 (uncharacterised rooms)	**Medium confidence** due to the moderate overall confidence in the contributing inventories and uncertainties in the backfill volume. **Moderate contribution** to the overall inventory (~13%). **Therefore, overall RAG score is Amber (box 5).**	Mass: 6.12E+07 kg Volume: 2.97E+04 m^3 0% is to be disposed of in-situ.	Alternative inventory calculated assuming alternative inventory activities for contributing features and components above, and assuming an additional 10% volume contribution from all sources.

[1] FP-028 is derived from the contamination of ancillary pipework rather than bulk concrete; contributing materials are smear samples (50%), paint (20%) and metal (30%), and may therefore not be representative of concrete.

4.3 Confidence in the Dragon Inventory

Table 4.2: Summary of the inventory derivation approaches and relative confidence for Dragon components considered in this inventory (see Section 4.1 for full explanation).

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints / other relevant info	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to Dragon inventory	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Bioshield – Portland concrete	Six cores through the bioshield, giving a total of 57 samples: - 2017: single core at -5' level, 85° radial (high activation expected due to PGPC neutron pathway): 29 concrete samples along length. Gamma spectroscopy and activation products. - 2013: three cores all at 315° radial, approx. heights +6'3", +1'0", -5'0". Total of 5 concrete samples: one from either end for two cores and at outer end for third. - 2005: one core at ground floor 0° radial (intended to be max neutron flux); one core at +18' (furthest point from max flux). Both cores subsampled giving 23 concrete samples.	2013 FP for concrete bricks from Upper Support Ring. SGHWR neutron activation modelling also used to inform inventory by analogy (for a small number of radionuclides).	FP for Dragon bioshield concrete developed from characterisation data, using (generally bounding) ratios (generally to ^{152}Eu) in previous fingerprints and indicative activation analysis to infer results where analysis for individual radionuclides in specific cores were either not requested or were below the LOD. Tailored approach for individual radionuclides; some excluded due to short half-lives. Based on the profile observed in cores, it is conservatively assumed that the derived FP applies only to the first 750 mm of the bioshield and that it is not activated beyond this. The fingerprint is therefore calculated from the average of the above-LOD values less than 750 mm from the bioshield inner surface. The FP was calculated both as average and maximum activity in Bq/g, and as average %. Total activation inventory calculated using the derived FP (Bq/g), assuming a uniform activity profile over the first 750 mm, volume according to AutoCAD model dimensions (minus volume associated with barytes concrete), and density of 2,437 kg/m^3 as determined from bioshield samples.	INV-DRAGON-004 (derived FP assumed to be representative of the bioshield; no statistical analysis of characterisation data robustness). INV-DRAGON-005 (SGHWR activation modelling may not be closely analogous to Dragon activation – but reliance on this in the inventory derivation is limited). INV-DRAGON-006 (generic material specifications used).	**Medium to high confidence** in the inventory: it is supported by significant characterisation data and the derivation approach is sound, but there is some uncertainty over the representativeness of the FP. **Moderate to significant contribution** (15% in-situ, plus 12% as backfill) to the overall Dragon inventory. **Therefore, overall RAG score is Amber (box 5)**.	For in-situ component only: 7.75E+01 m^3 1.89E+05 kg	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Bioshield – barytes concrete	No samples specific to bioshield barytes concrete.	None specific to barytes concrete.	Indicative generic composition information on barytes concrete from two studies used to calculate average proportion reduction in Ca content and increase in Ba content (in barytes concrete compared to Portland). This is assumed to scale directly to give corresponding proportion changes in the Dragon bioshield concrete FP derived above. Approximate volume of barytes concrete estimated by scaling from drawing, conservatively assuming the barytes regions extend the full height of the bioshield and limiting to the first 750 mm. Total activation inventory calculated using this volume, the derived FP (Bq/g) and an assumed barytes concrete density of 3,650 kg/m^3.	As above; particularly INV-DRAGON-006 (uncertain specification of barytes concrete significant given lack of characterisation data and limited material knowledge. Extent also unknown, but modelled volume believed to be bounding).	**Low to medium confidence** in the inventory: it is not underpinned by any direct samples and is based on generic composition information and an unknown extent. However, it is believed to be bounding. **Moderate contribution** (5% in-situ, plus 4% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Amber (box 5).**	For in-situ component only: 1.50E+01 m^3 5.47E+04 kg	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Bioshield – rebar	Four rebar samples from the bioshield concrete cores detailed above: - One from the 2017 core, based on the section of rebar with greatest activity (subsampled as swarf) - Three from the 2005 cores, one from the ground floor and two from the +18' level. However, only two of these provide a few above LOD results.	2014 FP for mild steel baseplate. Thermal shield sample analysis (from 2017 coring). SGHWR activation modelling (not used in final inventory calculation)	Significantly different radionuclide ratios seen in the steel baseplate FP / thermal shield results compared to the SGHWR activation modelling. Therefore, two alternative FPs (based on each of these) were derived and the most conservative one taken as the Dragon bioshield rebar FP. Therefore, FP derived using average rebar sample results from 2017 and 2005 cores, scaled according to the mild steel baseplate FP. The FP was calculated both as average and maximum activity in Bq/g, and as average %. Conservatively estimated to be 150 kg rebar per m^3 of bioshield (volume calculated according to AutoCAD model dimensions as above). Total activation inventory calculated by applying the derived FP (Bq/g) to the calculated mass.	INV-DRAGON-004 (derived FP assumed to be representative of rebar; no statistical analysis of characterisation data robustness). INV-DRAGON-006 (generic material specifications used; also amount of rebar per m^3.)	**Medium confidence** in the inventory: it is supported by a small number of samples whose representativeness is unknown, but conservatisms are built into the derivation approach. **Small contribution** (0.4% in-situ, plus 0.3% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2)**.	For in-situ component only: 1.39E+04 kg in the 9.25E+01 m^3 of activated bioshield thickness	Alternative inventory calculated using maximum instead of average activity concentrations.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints / other relevant info	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to Dragon inventory	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Reactor building (B70) – general area – surface contamination	- Radiological characterisation data from ten sampling datasets from various locations throughout the Dragon complex (1999-2016) (total of 264 samples, covering a range of material types and both surface and core locations – 99 surface samples used for surface contamination FP). - ViridiScope in-situ remote laser sampling (total 147 samples) to identify hotspots (2018), used to scale the FP. However, no single statistically representative sampling campaign undertaken.	Several general area contamination FPs, some using results for items subsequently removed (not used in this study). Magnox standard probe response calibration and activity conversion.	Total surface area that may have been exposed to general atmospheric contaminants assumed to include all below-ground concrete and brick surfaces for the entire building expected to remain at the end state, including walls, floors, ceilings, vent plant room (excluding B78 and the Betalite store area). Also, below-ground portion of steel shell. Calculated using dimensions taken from CAD model. New general area FP derived for this study incorporating recent sampling data and verified historic data. Samples grouped into areas to check for characterisation sub-groups; considered appropriate to average samples from the following areas: B70 inner and outer walls, vehicle airlock, and B78. Radionuclides included/excluded on basis of factors such as measurements, half-life and likelihood of presence. FP used with the highest activity identified in ViridiScope sampling (100 cps) to calculate a surface activity (Bq/cm^2), via a NRS standard probe response calibration and activity conversion. This activity is assumed to apply to the total surface area, but as the resulting inventory is very pessimistic, an assumption is made that only 5% of the surface activity is present.	INV-DRAGON-004 (derived FP assumed to be representative; no statistical analysis of characterisation data robustness). INV-DRAGON-007 (highest measured hotspot contamination applied to whole building is extremely pessimistic, so assumed that only 5% of the surface contamination is present – based on assumption of no significant contamination, but this is not underpinned. Pu isotopes excluded but lack of evidence for them is not conclusive).	**Low to medium confidence** in the inventory: it is supported by significant characterisation data, but there is considerable uncertainty about its representativeness, and the assumption that only 5% of the surface activity is present is not underpinned. **Small contribution** (0.3% in-situ, plus 0.5% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	For in-situ component only: 1.30E+01 m^3 3.12E+04 kg	Alternative inventories calculated: i) assuming 100% rather than 5% of surface contamination is present; and ii) applying an alternative fingerprint containing Pu isotopes.
Yes	Reactor building (B70) – Betalite store area (room 121; -25' level) – surface contamination	As above (total 264 samples from various locations including 29 from the Betalite store). 5 surface samples from the Betalite store used to calculate ^{3}H in the FP.	None specific to the Betalite store.	Floor surface area approximated as a rectangle. New general area FP derived for this study: as above except using ^{3}H results from the Betalite store area only. Average ^{3}H for the Betalite store area calculated from four sample results (excluding one anomalously high paint sample that is assumed to be either not be real or highly localised; in either case not suitable for inclusion in a general area FP). Inventory calculated as above, using the Betalite store area-specific FP.	As above; also INV-DRAGON-007 (single anomalously high ^{3}H result assumed to be in error and not included in fingerprint, but possibility it could be real).	**Low to medium confidence** in the inventory: it is supported by significant characterisation data, but there is considerable uncertainty about its representativeness, and the assumption that only 5% of the surface activity is present is not underpinned. **Small contribution** (0.1% in-situ) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	For in-situ component only: 2.68E+00 m^3 6.43E+03 kg	Alternative inventories calculated: i) assuming 100% rather than 5% of surface contamination is present; and ii) including the anomalously high ^{3}H result in the FP.
Yes	Reactor building (B70) – general area – tritium ingress	Tritium ingress based on two sampling datasets from the ten noted in the row for B70 general area surface contamination, which consist of concrete cores sub-sampled along their length (total of 141 subsurface concrete samples from areas excluding the Betalite store).	None	Mass of paint and concrete layers calculated using the surface areas used for surface contamination of the equivalent areas, generic values for paint and concrete density, and depth ranges as follows: paint layer (1 mm), 0-5 cm, 5-10 cm, 10-15 cm and 15-30 cm). For each of the depth ranges, the ^{3}H activity was calculated by multiplying the mass by the average activity of samples in the depth range from the general B70 area (excluding Betalite store).	INV-DRAGON-004 (derived FP assumed to be representative; no statistical analysis of characterisation data robustness). INV-DRAGON-006 (generic material specifications used).	**Medium to high confidence** in the inventory: supported by significant characterisation data; some remaining uncertainties. **Moderate to significant contribution** (8% in-situ, plus 14% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Amber (box 5).**	For in-situ component only: 1.83E+03 m^3 4.34E+06 kg	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Reactor building (B70) - Betalite store area (room 121; -25' level) – tritium ingress	As above (24 of the subsurface concrete samples in these datasets are from the Betalite store).	None	As above, except the average activity of samples taken only from the Betalite store was used.	As above.	**Medium to high confidence** in the inventory: supported by significant characterisation data; some remaining uncertainties. **Small contribution** (3% in-situ) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	7.77E+01 m^3 1.86E+05 kg	Alternative inventories calculated: i) using maximum instead of average activity concentrations, and ii) including the anomalously high ^{3}H result in the FP.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints / other relevant info	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to Dragon inventory	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
Yes	Residual contamination from the PGPC contaminated water spill	Dose rate measurements plus MicroShield modelling (further characterisation data expected in future).	Dragon primary coolant FP considered to be applicable	Upper value of current estimated range for total activity used, together with primary coolant fingerprint. Activity concentrations calculated assuming contamination has penetrated 10 mm into concrete floor of density 2400 kg/m^3 over an area of 3.3 m^3. For total Winfrith end state in-situ disposal inventory, it is assumed that 95.5% of the contamination will be removed during clean-up, the level needed to reduce the activity concentration to the upper limit for LLW.	INV-DRAGON-010 (all PGPC-related uncertainties including total activity, radionuclides present, penetration depth and extent of clean-up)	**Low confidence** in the inventory; activity assumptions not supported by any sampling data yet; significant remaining uncertainties. **Moderate contribution** (13% in-situ) to overall Dragon inventory. This is highly dependent on assumptions, especially level of clean-up, and could increase. **Therefore, overall RAG score is Red (box 6).**	3.30E-02 m^3 7.92E+01 kg	None; discussion of contribution if only 90% clean-up achieved, but assumed to be decontaminated to at least LLW.
Yes	Fuel Store building (B78) – surface contamination	As for B70 general area surface contamination (29 of the total 264 samples were from B78; no ViridiScope locations in B78).	None	As for B70 general area surface contamination: FP derived in this study for Dragon general area surface contamination adopted; highest count rate (100 cps) of ViridiScope survey and probe efficiency calibration used to calculate a hotspot equivalent surface activity of 9.91 Bq/cm^2; 5% of total surface activity assumed to be present.	As for B70 general area surface contamination: INV-DRAGON-004 (FP assumed to be representative) INV-DRAGON-007 (assumption of 5% surface contamination not well underpinned).	**Low to medium confidence** in the inventory: supported by some characterisation data, but uncertainty about its representativeness, and the assumption that only 5% of the surface activity is present is not underpinned. **Small contribution** (0.04% in-situ, plus 0.2% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2)**.	For in-situ component only: 1.49E+00 m^3 3.57E+03 kg	Alternative inventories calculated: i) assuming 100% rather than 5% of surface contamination is present; and ii) applying an alternative fingerprint containing Pu isotopes.
Yes	Fuel Store building (B78) – tritium ingress	As for B70 tritium ingress (17 of the subsurface concrete samples in these datasets are from B78).	None	As for B70 tritium ingress. Only ingress up to 15cm considered, to avoid double counting (as the walls in B78 are generally only 30cm thick).	As for B70 general area tritium ingress: INV-DRAGON-004 (derived FP assumed to be representative). INV-DRAGON-006 (generic material specifications used).	**Medium confidence** in the inventory: supported by some characterisation data; some remaining uncertainties. **Small contribution** (0.5% in-situ, plus 2% as backfill) to overall Dragon inventory. **Therefore, overall RAG score is Amber (box 2)**.	For in-situ component only: 1.07E+02 m^3 2.56E+05 kg	Alternative inventory calculated using maximum instead of average activity concentrations.
Yes	Reactor building (B70) backfill	See rows for bioshield (Portland cement, barytes concrete and rebar), B70 general area surface contamination and B70 general area tritium ingress. Rubble stockpiles: Groundhog survey and non-representative surface and near-surface sampling of mounds [93].	See relevant rows above. A59 general fingerprint (FP-004) for rubble stockpiles.	It is assumed that demolition material (concrete blocks and/or rubble) from the above-ground concrete structure of the Dragon Reactor buildings will be used to fill the Dragon below-ground voids, topped up with approximately 1,100 m^3 from the existing rubble stockpiles. The inventory comprises the above-ground portions of the bioshield (including barytes concrete and rebar), general B70 surface contamination and tritium ingress, as derived as in the corresponding below-ground rows above, and the inventory associated with the stockpile rubble. Demolition material from B78 is assumed not to be included in the backfill. The inventory derivation approach for the rubble mounds is discussed in Section 2.17. Activities are either from the average sample data from the mounds or based on the ratios of FP-004 scaled to EPR16 OoS levels. For the purposes of calculating an activity concentration for the backfill, a compacted rubble density (intermediate between stacked blocks and loose rubble) of 1,960 kg/m^3 has been assumed.	INV-DRAGON-002 (demolition strategy – above/below ground proportions, location/ concentration of inventory may change). INV-DRAGON-003 (backfill strategy – materials balance (demolition volume), contribution of B78, backfill form/density are all uncertain). INV-SGHWR-002 (uncertainties associated with fingerprint use) INV-SGHWR-006 (adequateness of characterisation data in relation to the rubble mounds) Fundamental uncertainties are those associated with the corresponding component rows.	**Medium confidence** in the inventory: reflects the overall status of underlying inputs (see rows above), uncertainty in the characterisation of the rubble mounds, and uncertainty in volume and density of the backfill material. **Significant contribution** (54%) to overall Dragon inventory, comprising components noted in rows above plus 20% rubble from existing stockpiles. **Therefore, overall RAG score is Amber (box 8)**.	6.54E+03 m^3 1.29E+07 kg	Alternative inventories calculated: i) using maximum instead of average activity concentrations for rubble component, and ii) carrying through alternative inventories calculated for components with an in-situ portion as described in rows above.

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints / other relevant info	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to Dragon inventory	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
No	Region beneath the fuel carousel (B70)	Potential for low-level actinide contamination but can only be characterised once accessible	None – expect different FP to rest of B70	No inventory derived as it is assumed that this area will be decontaminated as appropriate once it is accessible and the residual inventory will therefore be negligible.	INV-DRAGON-001 (assumption as stated)	Although no inventory has been derived, this area is expected to be decontaminated and so there is **medium confidence** that the residual inventory will be insignificant. Expected **negligible contribution** to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	Not determined	None
No	Steel-lined sump beneath the reactor (B70)	Potential for low-level fission product contamination but can only be characterised once accessible	None – expect different FP to rest of B70	No inventory derived as it is assumed that this area will be decontaminated as appropriate once it is accessible and the residual inventory will therefore be negligible.	INV-DRAGON-001 (assumption as stated)	Although no inventory has been derived, this area is expected to be decontaminated and so there is **medium confidence** that the residual inventory will be insignificant. Expected **negligible contribution** to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	Not determined	None
Yes	Primary Mortuary Hole Structure (B78), including sump, ventilation ducts and lateral vents	- Systematic sampling campaign (following DQO process) undertaken in 2023. - All MHs surveyed and smear samples taken at top, cross vent and full-height. Samples with highest cps readings at the three locations underwent radioisotope analysis. - Results used to derive inventories for MH and cross vents components. - Main ventilation ducts and sump inventory based on 2016 inventory estimate which used a smear from the ventilation stack outlet (assumed to still be most representative sample for these components).	Magnox standard probe response calibration and activity conversion.	For the mortuary holes component, two separate inventories were calculated, one using the fingerprint and probe response value from the MH with the highest cps reading for the top smear sample in the initial survey, and one using the fingerprint and probe response value from the MH with the highest beta/gammas cps reading for the full-length smear sample in the initial survey, applied to the beta/gamma cps reading for the top smear and full-length smear respectively from each MH. The "average" inventory is the average value of the two inventories for each radionuclide and the "maximum" inventory is the highest value of the two inventories for each radionuclide. For the cross vent component, the "average" inventory was calculated for each of ten (top and bottom) cross vents, using the fingerprint and probe response value from the MH with the highest cps reading for the cross vent smear sample in the initial survey, applied to the average beta/gamma cps reading of the cross vent smear sample from the group of five MHs connected by each of the ten cross vents. The "maximum" inventory was calculated using the highest (rather than average) beta/gamma cps reading of the cross vent smear sample from the group of five MHs connected by each of the ten cross vents. Total inventories calculated by summing the MH, cross vents, main ducts and sump components. To calculate activity concentration, three alternative contamination volumes have been considered: i) the first 1 mm of the entire steel structure; ii) loose contamination averaged over the infill volume; iii) the planned concrete monolith comprising the entire pit in which the steel structure and grout infill will sit.	INV-DRAGON-008 (FP of fixed contamination and ratio of loose to fixed contamination uncertain; smear pick-up efficiency uncertain; no direct characterisation of some parts of system including bottom cross vents; unclear how to characterise contamination volume in order to calculate activity concentration)	**Medium to high confidence** in the inventory, as based on data from systematic sampling campaign (but uncertainty remains regarding fixed contamination and potential for higher contamination in bottom cross vents). **Small contribution** (0.5%) to overall Dragon inventory. **Therefore, overall RAG score is Green (box 2).**	Three alternatives as described in inventory derivation approach column: i) 3.20E-01 m^3; 2.51E+03 kg contaminated steel ii) 2.71E+01 m^3; 6.50E+04 kg contaminated grout infill iii) 2.35E+02 m^3; 5.64E+08 kg contaminated concrete monolith	Alternative inventory adopts the "maximum" inventories for MH and cross vent components as described in inventory derivation approach column.
No	Fresh fuel Mortuary Hole Structure and metal lining of storage pit (B78)	None	None	No inventory derived as these are expected to be removed and disposed of off-site.	INV-DRAGON-009 (assumption as stated)	Although no inventory has been derived, this structure is expected to be removed and so there is **high confidence** that the residual inventory will be insignificant. Expected **negligible contribution** to overall Dragon inventory. **Therefore, overall RAG score is Green (box 1).**	Not determined	None

Rad. Inv. Derived?	Inventory disposal area components	Characterisation data supporting the inventory	Fingerprints / other relevant info	Inventory derivation approach	Uncertainty cross-reference	Overall confidence & significance to Dragon inventory	Assumed contaminated volume & mass	Sensitivity analysis / basis for alternative inventory
No	B78 Mortuary Hole Structure bulk concrete	None, but some may be available after the current (2023) mortuary hole sampling programme	None	No inventory derived as no radionuclides are expected to have migrated into the concrete due to the steel structure of the mortuary holes and lining of the pit. There is a small possibility of migration having occurred at steel joints, but there is currently no evidence for this.	INV-DRAGON-009 (assumption as stated)	**Medium confidence** that the inventory will be negligible: logical arguments suggest this, but there are no data to confirm it. Expected **negligible contribution** to overall Dragon inventory. **Therefore, overall RAG score is Green (box 1).**	Not determined	None

5 Inventory Associated with A59

5.1 Background

522 Building A59 was the PIE facility and was used to examine material from various reactors, as well as undertake operations involving re-concentration of D_2O and decontamination activities using acids and other decontamination liquids. The A591 facility formed part of the connection between the operations within the A59 building and the discharge of active liquid wastes to ALES. A number of historical incidents were recorded in A59 and A591 that had the potential for ground contamination; remediation activities have been undertaken but some radiological contamination is known to be present. The current A59 area includes two Areas of Potential Concern (APCs): the Pit 3 / Pressurised Suit Area (PSA) APC and the A591 / Heavy Vehicle Airlock (HVA) APC. Spotty contamination is understood to be present in the remaining A59 area.

523 A full description of the A59 area, its building history, incidents with potential for ground contamination, building demolition and the remediation programme can be found in Section 2 of the separate A59 inventory report [10].

5.2 Proposed End State

524 The two current APC features of the A59 area could potentially be contaminated to above OoS levels. Following a review of inventory and options for the A59 area [11], it is anticipated that remediation will be undertaken as necessary to satisfy OoS requirements[21]. Thus, the A59 area does not form part of the permit application for on-site disposal. However, inventory information for the A59 area is needed to support site decommissioning activities and the radiological performance assessment, and to inform site monitoring expectations. Therefore, an estimate for the remediated OoS A59 area inventory is included here.

5.3 Origin and Constraints on the Radiological Inventory

525 The dataset on which the A59 inventory is based consists of the following elements:

- Excavation surface sampling and monitoring (2007-2008). This relates to a programme of verification gamma monitoring and radiochemical sampling and analysis on the surface of the A59 and A591 excavations (which were undertaken as separate projects) prior to backfilling and capping, as well as post-backfill monitoring and sampling once the backfilling and capping was complete. This is the primary information source for the radiological inventory of the in-situ A59 contamination (i.e. post remediation).
- Restored ground surface sampling and monitoring (2009-2010). Following completion of the remediation works the A59 excavation area was backfilled using a combination of soil determined to meet a remediation criterion and new

[21] Once remediated (if necessary) and demonstrated to be OoS, these areas would be classed as “Former APCs”.

soil imported onto the Winfrith site. The new ground surface was then gamma surveyed and sampled for radiological analysis.

- Recent site investigations and monitoring, including 2016-2018 site investigations involving systematic soil profile sampling; a 2019 review of activities in groundwater; further analysis of archive soil samples in 2019; and a background radioactivity study undertaken in 2019.

526 These datasets are presented and discussed in detail in Section 3 of the separate A59 inventory report [10].

5.4 Inventory Estimate

527 Reference inventory estimates have been derived for three A59 features: each of the two current APCs, and all other areas of the former A59 building footprint taken together. Section 4 of the separate A59 inventory report [10] explains the approach taken to estimating these inventories, including:

- Determination of appropriate fingerprints and dimensions.
- Derivation of inventories for the above three features for both the remediation works excavation surface and the infill inventory.
- Development and application of a method accounting for the "spottiness" of contamination to reduce the impact of over-representing zones of elevated contamination.

528 The resulting inventories are presented in terms of total activity, average activity concentrations and maximum activity concentrations, at three dates: 2008 (reflecting the historic remediation works), 2023 and 2027 (for direct use in the end state assessments). These inventories can be found in Section 4.6 and Appendix B of [10] and are reproduced in the tables below for 2027.

529 A sum-of-fractions comparison of the total activities (and average activity concentrations) against radionuclide-specific OoS levels in EPR16 shows that the reference inventory estimates for all A59 features already meet OoS criteria; if future characterisation confirms this to be the case then no additional remediation would be required.

Table 5.1: Average and maximum activity concentrations for the three A59 features (and soils used as infill), presented for a reference date of 01/01/2027 [10, Tab.B.1 and B.2].

Radio-nuclide	PSA/Pit 3		A591/HVA		A59 Other Areas		Infill	
	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]
^{60}Co	4.97E-04	3.68E-03	2.14E-02	2.39E-01	6.17E-04	9.20E-03	6.34E-04	4.97E-03
^{63}Ni	0.00E+00	0.00E+00	1.41E+00	1.80E+01	7.84E-02	1.19E+00	8.19E-02	6.42E-01
^{90}Sr	1.27E-02	2.41E-01	1.58E-01	1.87E+00	4.00E-02	4.72E-01	3.12E-02	9.03E-02
^{125}Sb	8.10E-06	4.04E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{137}Cs	1.72E-02	8.57E-01	4.74E-02	6.33E-01	1.29E-02	1.52E-01	1.41E-02	2.90E-02
^{226}Ra	7.72E-09	2.77E-08	4.79E-08	1.96E-07	1.49E-08	2.63E-08	9.74E-09	1.02E-08

Radio-nuclide	PSA/Pit 3		A591/HVA		A59 Other Areas		Infill	
	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]	Average [Bq/g]	Maximum [Bq/g]
^{227}Ac	1.73E-07	6.18E-07	8.51E-07	2.86E-06	1.65E-07	2.86E-07	1.06E-07	1.11E-07
^{230}Th	1.97E-06	7.08E-06	1.22E-05	4.99E-05	3.80E-06	6.70E-06	2.48E-06	2.59E-06
^{231}Pa	7.22E-07	2.57E-06	3.54E-06	1.19E-05	6.86E-07	1.19E-06	4.41E-07	4.61E-07
^{234}U	1.18E-02	4.24E-02	7.32E-02	2.99E-01	2.28E-02	4.01E-02	1.49E-02	1.55E-02
^{235}U	1.88E-03	6.70E-03	9.22E-03	3.10E-02	1.79E-03	3.10E-03	1.15E-03	1.20E-03
^{238}U	1.33E-02	4.82E-02	7.18E-02	2.96E-01	2.34E-02	4.12E-02	1.53E-02	1.59E-02
^{237}Np	5.50E-08	1.25E-06	1.03E-07	1.96E-06	1.76E-08	4.56E-07	1.26E-08	1.97E-08
^{238}Pu	8.19E-04	2.06E-02	9.79E-04	1.04E-02	2.92E-04	7.35E-03	2.42E-04	7.80E-04
^{239}Pu	1.03E-02	2.75E-01	9.13E-04	1.13E-02	1.42E-03	3.67E-02	1.76E-03	1.63E-02
^{240}Pu	7.46E-03	2.00E-01	1.43E-03	1.77E-02	1.97E-03	5.11E-02	2.45E-03	2.27E-02
^{241}Pu	2.41E-02	5.04E-01	1.92E-02	1.29E-01	6.13E-03	1.45E-01	4.83E-03	1.50E-02
^{241}Am	9.72E-03	2.20E-01	1.77E-02	3.32E-01	3.08E-03	7.95E-02	2.21E-03	3.60E-03
^{244}Cm	0.00E+00	0.00E+00	9.03E-04	1.26E-02	9.80E-04	2.55E-02	6.97E-04	1.02E-03
Sum	**1.10E-01**	**2.42E+00**	**1.83E+00**	**2.19E+01**	**1.94E-01**	**2.25E+00**	**1.71E-01**	**8.58E-01**

Table 5.2: Estimated reference radionuclide inventory for the three A59 features, presented for a reference date of 01/01/2027 [10, Tab.4.26]. The inventory is derived assuming the average activity concentrations. The reference inventory for each feature incorporates both the excavation surface and backfill inventory contributions.

Radio-nuclide	PSA/Pit 3 [MBq]	A591/HVA [MBq]	A59 Other Areas [MBq]	Total [MBq]
^{60}Co	1.46E+00	1.50E+01	1.27E+01	2.91E+01
^{63}Ni	4.81E+01	9.88E+02	1.62E+03	2.66E+03
^{90}Sr	4.62E+01	1.13E+02	7.80E+02	9.39E+02
^{125}Sb	1.78E-02	0.00E+00	0.00E+00	1.78E-02
^{137}Cs	4.61E+01	3.45E+01	2.70E+02	3.50E+02
^{226}Ra	2.27E-05	3.43E-05	2.83E-04	3.40E-04
^{227}Ac	3.81E-04	5.91E-04	2.66E-03	3.63E-03
^{230}Th	5.79E-03	8.76E-03	7.21E-02	8.66E-02
^{231}Pa	1.59E-03	2.46E-03	1.11E-02	1.51E-02
^{234}U	3.47E+01	5.25E+01	4.32E+02	5.19E+02
^{235}U	4.13E+00	6.40E+00	2.89E+01	3.94E+01
^{238}U	3.83E+01	5.16E+01	4.43E+02	5.33E+02
^{237}Np	1.28E-04	7.27E-05	3.39E-04	5.40E-04
^{238}Pu	1.94E+00	7.05E-01	5.75E+00	8.40E+00
^{239}Pu	2.37E+01	8.26E-01	3.04E+01	5.49E+01
^{240}Pu	1.79E+01	1.26E+00	4.24E+01	6.15E+01

Radio-nuclide	PSA/Pit 3 [MBq]	A591/HVA [MBq]	A59 Other Areas [MBq]	Total [MBq]
^{241}Pu	5.58E+01	1.39E+01	1.20E+02	1.89E+02
^{241}Am	2.27E+01	1.25E+01	5.93E+01	9.45E+01
^{244}Cm	4.09E-01	7.04E-01	1.88E+01	1.99E+01
Sum	**3.41E+02**	**1.29E+03**	**3.86E+03**	**5.49E+03**

5.5 Sensitivity Analysis

530 There are various uncertainties in the A59 inventory, as discussed in detail throughout, and summarised in Appendix A of, the separate A59 inventory report [10]. To account for these uncertainties, alternative inventories were derived using the maximum instead of average activity concentrations for each feature [10, Tab.4.28]. These alternative inventories are considered to be conservative, as they are based on the maximum concentrations measured or calculated for every radionuclide across the dataset, which do not occur in any one sample.

531 A sum-of-fractions comparison against radionuclide-specific OoS levels shows that the total activities for the Pit 3/PSA and A591/HVA alternative inventories calculated in this way are above EPR16 OoS criteria. Since it is intended to remediate both areas to satisfy OoS requirements, revised alternative inventories have been calculated by applying scaling factors to reduce them to OoS levels, but still retaining the same fingerprint proportions. The alternative total inventory for the “Other” area is already OoS, so no scaling is applied. Table **5.3** presents the revised alternative total inventories, which represent the highest activities possible once OoS criteria are met via remediation, while maintaining individual feature fingerprints.

532 Overall, the total A59 inventory in the revised alternative estimate is approximately 2.4 times higher than in the reference inventory estimate. Split out by feature, this results from a 2.7-fold increase in the inventory for the Pit 3/PSA and “Other” areas, and a 1.2-fold increase in the inventory for A591/HVAC.

Table 5.3: Alternative estimated radionuclide inventory for the three A59 features, presented for a reference date of 01/01/2027. The inventory is derived assuming the maximum activity concentrations [10, Tab.4.28], then scaled so that they just meet OoS criteria. The inventory for each feature incorporates both the excavation surface and backfill inventory contributions.

Radio-nuclide	PSA/Pit 3 [MBq]	A591/HVA [MBq]	A59 Other Areas [MBq]	Total [MBq]
^{60}Co	3.05E+00	1.74E+01	4.68E+01	6.73E+01
^{63}Ni	2.33E+02	1.31E+03	6.05E+03	7.59E+03
^{90}Sr	9.00E+01	1.37E+02	1.73E+03	1.95E+03
^{125}Sb	9.20E-02	0.00E+00	0.00E+00	9.20E-02
^{137}Cs	2.05E+02	4.64E+01	5.58E+02	8.10E+02
^{210}Pb	2.93E-06	2.37E-06	5.07E-05	5.60E-05

Radio-nuclide	PSA/Pit 3 [MBq]	A591/HVA [MBq]	A59 Other Areas [MBq]	Total [MBq]
^{210}Po	2.69E-06	2.17E-06	4.64E-05	5.12E-05
^{226}Ra	1.78E-05	1.44E-05	3.08E-04	3.40E-04
^{227}Ac	3.55E-04	2.10E-04	3.35E-03	3.92E-03
^{230}Th	4.55E-03	3.67E-03	7.85E-02	8.67E-02
^{231}Pa	1.48E-03	8.72E-04	1.40E-02	1.63E-02
^{234}U	2.72E+01	2.20E+01	4.70E+02	5.20E+02
^{235}U	3.85E+00	2.27E+00	3.63E+01	4.25E+01
^{235}U	2.86E-05	8.32E-07	1.20E-04	1.49E-04
^{238}U	3.04E+01	2.17E+01	4.83E+02	5.35E+02
^{237}Np	3.02E-04	1.43E-04	1.21E-03	1.65E-03
^{238}Pu	5.07E+00	7.65E-01	2.14E+01	2.73E+01
^{239}Pu	6.81E+01	1.01E+00	1.60E+02	2.29E+02
^{240}Pu	5.31E+01	1.55E+00	2.23E+02	2.78E+02
^{241}Pu	1.32E+02	9.54E+00	4.22E+02	5.64E+02
^{241}Am	5.33E+01	2.42E+01	2.11E+02	2.88E+02
^{244}Cm	3.71E-01	9.30E-01	6.70E+01	6.83E+01
Sum	**9.05E+02**	**1.60E+03**	**1.05E+04**	**1.30E+04**

533 The activity concentrations associated with the original alternative inventories are the maximum values in Table **5.1**. A sum-of-fractions comparison against radionuclide-specific OoS levels shows that these are above OoS criteria for all three A59 features. Since it is intended to remediate both Pit 3/PSA and A591/HVA to satisfy OoS requirements, revised alternative activity concentrations have been calculated, again by applying scaling factors to reduce them to OoS levels. No remediation is planned in the "Other" area because both the reference inventory and alternative total activity are OoS. The alternative activity concentration is only above OoS because it is conservatively based on a few samples from a small subsection of the area, with the remaining, much larger, area having sample measurements less than 10% of OoS, if not at the LOD. It is not realistic to assume that the entire volume is at this concentration and therefore the alternative activity concentration has also been scaled to just satisfy OoS (with the expectation that it would be lower than this in reality). Table **5.4** presents the revised activity concentrations associated with the alternative inventories for the three A59 features.

534 Separate factors are applied to the alternative total activity and activity concentration inventories because, as noted above (Section 5.4), the alternative activity concentration reflects the highest measured samples in the above 10% of OoS areas (without accounting for the lower activity in the "outside" areas). As it is planned to remediate the PSA/Pit 3 and HVA/A591 areas to satisfy OoS, it is appropriate to model the alternative inventory as no more than OoS in both the natural evolution (total activity) and inadvertent human intrusion/site occupancy (activity concentration) assessments. Therefore, scaling factors are applied to ensure that both inventories are OoS.

Table 5.4: Activity concentrations associated with the alternative estimated inventories for the three A59 features (and soil used as infill), presented for a reference date of 01/01/2027. These equate to the maximum activity concentrations from Table **5.1**, scaled (where necessary) to reduce them to OoS levels.

Radio-nuclide	PSA/Pit 3 [Bq/g]	A591/HVA [Bq/g]	A59 Other Areas [Bq/g]	Infill [Bq/g]
^{60}Co	4.36E-04	2.53E-02	3.54E-03	4.97E-03
^{63}Ni	0.00E+00	1.91E+00	4.57E-01	6.42E-01
^{90}Sr	2.86E-02	1.98E-01	1.82E-01	9.03E-02
^{125}Sb	4.79E-05	0.00E+00	0.00E+00	0.00E+00
^{137}Cs	1.02E-01	6.70E-02	5.83E-02	2.90E-02
^{210}Pb	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{210}Po	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{226}Ra	3.29E-09	2.07E-08	1.01E-08	1.02E-08
^{227}Ac	7.32E-08	3.03E-07	1.10E-07	1.11E-07
^{230}Th	8.38E-07	5.28E-06	2.58E-06	2.59E-06
^{231}Pa	3.05E-07	1.26E-06	4.58E-07	4.61E-07
^{234}U	5.02E-03	3.17E-02	1.54E-02	1.55E-02
^{235}U	7.94E-04	3.28E-03	1.19E-03	1.20E-03
^{235}U	0.00E+00	0.00E+00	0.00E+00	0.00E+00
^{238}U	5.71E-03	3.14E-02	1.58E-02	1.59E-02
^{237}Np	1.48E-07	2.08E-07	1.76E-07	1.97E-08
^{238}Pu	2.44E-03	1.10E-03	2.83E-03	7.80E-04
^{239}Pu	3.26E-02	1.20E-03	1.41E-02	1.63E-02
^{240}Pu	2.37E-02	1.87E-03	1.97E-02	2.27E-02
^{241}Pu	5.97E-02	1.37E-02	5.59E-02	1.50E-02
^{241}Am	2.61E-02	3.52E-02	3.06E-02	3.60E-03
^{244}Cm	0.00E+00	1.34E-03	9.79E-03	1.02E-03
Sum	**2.86E-01**	**2.32E+00**	**8.66E-01**	**8.58E-01**

6 Conclusions

535 A summary of the radiological inventory that represents a cautious but credible estimate of the inventory that could be left on the Winfrith site at the end state is given in Table **6.1**. The inventory is based on the current understanding of the features proposed for on-site disposal, namely those comprising the SGHWR and Dragon Reactor complexes and the A59 area, drawing on the characterisation, decontamination and decommissioning carried out so far. The inventory is presented in terms of total activity, and in terms of average and maximum activity concentrations. Note that the maximum concentration has been derived from the maximum activity concentration measured for each radionuclide across all samples obtained for a given feature, not from the sample with the maximum total concentration; therefore, it is very unlikely that such a conservative maximum would occur in any one future sample.

536 The summary data presented in Table **6.1** are plotted as two pie charts in Figure **6.1**. These clearly indicate the dominance of the SGHWR inventory (98% of the total radioactivity). The most significant features of the SGHWR inventory are the bioshield (59% of the SGHWR total), and then the secondary (11%) and primary (10%) containments.

537 The inventory estimates have been developed using a number of assumptions as there is limited information for some components and access limitations prevent sampling and characterisation at this time. A list of the assumptions and main uncertainties associated with the reactor complex inventory estimates[22] is given in the Uncertainty Management Plan (UMP) table of Appendix A. The uncertainties and assumptions discussed in this report for each of the candidate features are summarised in the UMP in the following groups:

- Potential for additional plant, structures and any contaminated land associated with SGHWR and Dragon to be included in the inventory scope.
- Uncertainties associated with comprehensiveness, scope, and applicability of waste fingerprints.
- Use of generic material compositions and densities due to lack of site-specific data.
- Adequateness and statistical robustness of the available characterisation data.
- Impact of changes to current outline demolition and backfill plans.

538 The key uncertainties and recommendations for future actions for SGHWR and for the Dragon Reactor complex inventories are summarised below.

539 The estimated SGHWR bioshield inventory comprises around 57% of the total Winfrith end state inventory; however, characterisation data available for this feature comprises radiological data from just two concrete cores. The cores were analysed for a relatively limited radiological suite when compared to the range of radionuclides which may be expected to be formed during concrete activation. Only three rebar samples were taken from the bioshield cores. One of these was taken at the periphery of the activated

[22] Those associated with A59 are presented in Appendix A of the separate A59 inventory report and are not discussed further here.

region, however the remaining two were further away from the reactor and as such provide no information regards the level of activation. As with the concrete samples, a relatively limited radiological suite was applied and so knowledge of the degree and nature of rebar activation is low. Estimates for gaps in the bioshield concrete and rebar fingerprints and derived inventory have been filled using an SGHWR activation modelling study; however, this study also has limitations. For instance, a number of generic parameters are used in the modelling study such that it is not specific to the SGHWR (e.g. trace element data for generic concretes and steels were used which will influence both the fingerprint and quantity of the radioactivity predicted to be present). Gaining further characterisation data on the bioshield concrete and rebar would reduce uncertainty here and lessen, or remove reliance on, the activation modelling.

540 The SGHWR secondary containment comprises the second most significant feature with 11.2% of the total Winfrith end state inventory, and 11 of 40 of the maximum radionuclide concentrations in the SGHWR inventory. The prominence of the secondary containment for maximal activities relates to a number of localised areas of more elevated contamination in this feature, with much of the remaining structure containing only very low-levels of contamination. Some of the more elevated areas of contamination have limited characterisation datasets; however, characterisation is still in progress in a number of these areas, which will reduce uncertainty.

541 A significant number of the SGHWR maximum radionuclide concentration values (11 of 40) are associated with the mortuary tubes, and this feature also accounts for most of the highest average concentrations across the features (23 of 40). This is largely due to the fact that the mortuary tubes inventory comprises one active component of a small volume, for which speculative and conservative contamination assumptions have been made.

542 The dominant feature of the estimated Dragon Reactor complex inventory is the backfill, which comprises 54% of the total Dragon complex inventory although less than 1% of the total Winfrith end state inventory. The Dragon backfill inventory comprises contributions from the bioshield and general building contamination estimates, as well as material from existing rubble stockpiles:

- The ordinary concrete bioshield fingerprint has been derived using the average activity across all core samples and applied over the whole inner bioshield concrete volume. Further characterisation to confirm that there are no areas of higher activation in the bioshield than already identified would support this approach. An indicative estimate for the barytes concrete fingerprint has been produced by assuming a proportionate scaling of the ordinary concrete bioshield activation fingerprint, but knowledge of the location and composition of the barytes concrete, and radiological sample analysis, would support improved understanding of the barytes concrete inventory. In addition, there are limited bioshield rebar sample data and differences in the steel trace contaminants could lead to significantly different activation fingerprints than assumed.
- The samples used to derive the Dragon general building contamination fingerprint were taken for different reasons from ten campaigns over the period 1999-2016, and were not taken to systematically sample all areas of the facility. For many radionuclides in the characterisation dataset the measured activities are very low and could be considered to be at the level of noise in the results.

Other than for ^{3}H, ^{137}Cs and ^{60}Co, there are insufficient results for the identified radionuclides to draw statistically meaningful conclusions (regarding activity or distribution across the facility). Further characterisation to reduce this uncertainty may be undertaken as decommissioning proceeds. In addition, the Dragon building surface contamination inventory has been calculated assuming that the highest measured surface contamination hotspot activity is present on 5% of the building surface area – this assumption is made on the basis that the building does not have any significant contamination, but further underpinning with additional sampling or uncertainty analysis in the PA is advised.

543 A recent (2023) systematic sampling campaign has significantly reduced uncertainty in the estimated inventory for the Dragon Primary Mortuary Hole Structure. Limited characterisation data exist for the PGPC spill inventory estimate; however, further characterisation to reduce the uncertainty in this estimate will be undertaken during clean-up and NRS is confident that it can be decontaminated to at least LLW.

544 The identified gaps, uncertainties and assumptions have been used in this report to support a qualitative assessment of the confidence in the inventory estimates for each component and in derivation of alternative, more conservative, inventory estimates. The alternative inventories explore the impact of uncertainties, but are not considered to be realistic estimates. The reference inventory for the SGHWR is estimated to have a total activity at 01/01/2027 of 6.12E+05 MBq, which is increased by a factor of 9.7 to 5.91E+06 MBq when accounting for uncertainties in the alternative inventory components. The increase in the Dragon Reactor complex inventory is not so significant, increasing by a factor of 3.5 from 7.23E+03 MBq to 2.55E+04 MBq. The OoS A59 inventory increases by a factor of 2.4 from 5.49E+03 MBq to 1.30E+04 MBq. The alternative inventory estimates also account for variations in possible fingerprints and radionuclide content for components where this is considered appropriate. Both the reference and alternative inventories will be considered in the radiological PA.

545 The identified gaps and uncertainties are also used by NRS to inform the need for additional characterisation. NRS will undertake further characterisation as demolition activities proceed and additional parts of the facilities are safely accessible, in an approach set out in the Strategic Inventory Management Plan. Additional characterisation will be undertaken as needed during the EA’s determination period and during implementation of the end state. EAC will also be applied to control material that is emplaced in the reactor voids. NRS recognises that it carries a risk in undertaking characterisation after the permit application has been submitted. If material is discovered beyond that indicated in the permit application then, following an options assessment, it is acknowledged that the material will need to be removed for off-site disposal or a delay in the permit application/final release incurred as revised assessment documentation is submitted. However, the inventory estimates presented here are considered to be a credible but cautious estimate of the end state activity, that are, in the main, characterised proportionately to the hazard presented. Sensitivity to alternative inventory assumptions has also been considered. In practice, the inventory estimates are expected to reduce as decommissioning proceeds and further characterisation information becomes available. The end state radiological inventory report will be revised as necessary as additional characterisation and sampling data become available.

Table 6.1: Features of the reference and alternative inventories of radioactive waste potentially left on the Winfrith site at the end state giving the estimated activity concentration and total inventory associated with each feature of SGHWR, Dragon and A59. Activities presented for a reference date of 01/01/2027. *(This page is set to print on A3.)*

Feature		Rationale for Inventory Estimate	Contaminated Mass [kg]	Contaminated Volume [m^3]	REFERENCE INVENTORY						ALTERNATIVE INVENTORY					
					Feature Total Activity		Activity Concentration [Bq/g]	Maximum Activity Concentration [Bq/g]	Reactor Complex / Area Activity		Feature Total Activity			Activity Concentration [Bq/g]	Reactor Complex / Area Activity	
					[MBq]	Feature %			[MBq]	Feature %	[MBq]	Feature %	Increased by factor		[MBq]	Feature %
SGHWR	Bioshield	Based on characterisation data from 2 concrete cores and neutron activation modelling of the concrete and rebar in the bioshield.	7.65E+05	3.14E+02	3.58E+05	58.6%	4.69E+02	8.92E+03	6.12E+05	98.0%	5.22E+06	88.3%	14.6	6.83E+03	5.91E+06	99.4%
	Mortuary Tubes	Preliminary, high level approach in the absence of characterisation data which adopts the sum of activities of primary circuit pipework, moderator circuit pipework, ponds liners, activated rebar, and activated reactor components for mortuary tubes liners. Further characterisation expected.	2.75E+03	3.50E-01	8.11E+03	1.3%	2.95E+03	9.37E+03			2.56E+04	0.4%	3.2	9.30E+03		
	Primary	Based on available characterisation data from Room 111, and deeper concrete intervals in the primary containment, assumed depths of penetration of contamination into the building fabric, SGHWR primary external contamination fingerprint (FP-028) and proportion more than 1 m below ground level.	4.96E+06	2.07E+03	6.05E+04	9.9%	1.22E+01	1.55E+03			2.55E+05	4.3%	4.2	5.15E+01		
	Secondary	Based on available characterisation data from the structure, 7 secondary containment fingerprints, assumed depths of penetration, and the proportion more than 1 m below ground level (assume comprises Levels 1-3). Some areas assumed to be inactive.	4.21E+06	1.75E+03	6.97E+04	11.4%	1.65E+01	5.62E+03			1.35E+05	2.3%	1.9	3.20E+01		
	Ponds	Based on 2016 ponds characterisation programme comprising 17 cores from pond floor areas and 126 wall cores.	1.17E+06	4.87E+02	1.09E+04	1.8%	9.32E+00	7.01E+03			2.01E+04	0.3%	1.9	1.73E+01		
	Ancillary Areas	Based on characterisation data where available and applicable fingerprints (including FP-003, FP-016 and FP-026). Some areas assumed to be inactive.	1.89E+06	7.89E+02	3.33E+03	0.5%	1.76E+00	7.81E+01			1.66E+04	0.3%	5.0	8.77E+00		
	Bulk structure	To account for tritium contamination of the bulk concrete. Based on the mass of accounted for structure and the median tritium activity for components with an inventory.	2.86E+07	1.19E+04	1.86E+04	3.0%	6.50E-01	6.50E-01			3.54E+04	0.6%	1.9	1.24E+00		
	Backfill	Rubble mounds assumed to be at out-of-scope (OoS) of RSR (to be confirmed in future characterisation). Incorporates inventory from demolished Levels 4-10.	6.12E+07	2.97E+04	8.23E+04	13.5%	1.35E+00	2.74E+03			2.04E+05	3.4%	2.5	3.33E+00		
Dragon	Below cutline Bioshield	Based on characterisation data from 6 cores, fingerprints for Dragon Upper Support Ring concrete blocks and the mild steel baseplate, and by analogy with SGHWR neutron activation modelling.	2.57E+05	9.25E+01	1.51E+03	20.9%	5.86E+00	2.84E+01	7.23E+03	1.2%	6.41E+03	25.2%	4.2	2.49E+01	2.55E+04	0.4%
	Below	Fingerprint derived from characterisation data for 10 datasets at various locations in Dragon and	4.58E+06	1.91E+03	8.12E+02	11.2%	1.52E+01	1.61E+02			6.30E+03	24.7%	7.8	1.55E+01		

Feature		Rationale for Inventory Estimate	Contaminated Mass [kg]	Contaminated Volume [m^3]	REFERENCE INVENTORY						ALTERNATIVE INVENTORY					
					Feature Total Activity		Activity Concentration [Bq/g]	Maximum Activity Concentration [Bq/g]	Reactor Complex / Area Activity		Feature Total Activity			Activity Concentration [Bq/g]	Reactor Complex / Area Activity	
					[MBq]	Feature %			[MBq]	Feature %	[MBq]	Feature %	Increased by factor		[MBq]	Feature %
	cutline B70 Building Contamination	scaled using most active hotspot measured in the 2018 in-situ sampling campaign; assumes 5% of the building structure is surface contaminated. ^{3}H ingress 30 cm into building structure. Betalite store area inventory (included in this row) calculated separately using a dedicated fingerprint.														
	PGPC Spill	Preliminary estimate based on Dragon primary coolant fingerprint (shown to closely correlate with Purge Gas Pre-Cooler (PGPC) contamination), together with an estimate for the total activity remaining in the contaminated floor region derived from MicroShield dose modelling. 95.5% of contamination is assumed to be removed. Further characterisation expected.	7.92E+01	3.30E-02	9.50E+02	13.1%	1.20E+04	1.20E+04			9.50E+02	3.7%	1.0	1.20E+04		
	Backfill	Assumed to comprise the above-ground portions of the bioshield, B70 building and B78 building, emplaced as concrete blocks and/or rubble, together with some material from the existing rubble stockpiles. Based on demolition to ground level and known dimensions, the inventory is expected to contain 51% of total bioshield activity and 65% of the building surface contamination activity and ^{3}H ingress into structure.	1.29E+07	6.54E+03	3.88E+03	53.7%	3.02E-01	1.61E+02			1.16E+04	45.4%	3.0	8.98E-01		
	Primary Mortuary Hole Structure	Estimate for mortuary holes and cross vents based on systematic 2023 survey and sampling campaign; estimate for main ventilation ducts and sump based on smear from ventilation outlet stack (2016 inventory).	2.51E+03	3.20E-01	3.37E+01	0.5%	5.18E-01	1.34E+01			4.76E+01	0.2%	1.4	7.33E-01		
	B78 Floor Slab	Fingerprint, contamination level and % of contamination present are assumed to be the same as for B70 general building contamination. ^{3}H ingress 30 cm into building structure.	2.56E+05	1.07E+02	4.01E+01	0.6%	1.52E+01	4.29E+01			2.20E+02	0.9%	5.5	1.57E+01		
A59	PSA / Pit 3 APC	Remediated OoS end state inventory estimate derived for (i) the historical remediation works excavation surface at the cessation of soil removal; and (ii) infill material used to create the post-remediation ground surface. Primarily based on remediation dataset including verification gamma monitoring, radiochemical sampling and analysis prior to backfilling; and post-backfill monitoring and sampling, supplemented by subsequent analysis.	2.20E+06	1.10E+03	3.41E+02	6.2%	1.10E-01	2.42E+00	5.49E+03	0.9%	9.05E+02	7.0%	2.7	2.86E-01	1.30E+04	0.2%
	A591 / HVA APC		6.95E+05	3.47E+02	1.29E+03	23.5%	1.83E+00	2.19E+01			1.60E+03	12.3%	1.2	2.32E+00		
	Other A59 Areas		1.61E+07	8.07E+03	3.86E+03	70.3%	1.94E-01	2.25E+00			1.05E+04	80.7%	2.7	8.66E-01		

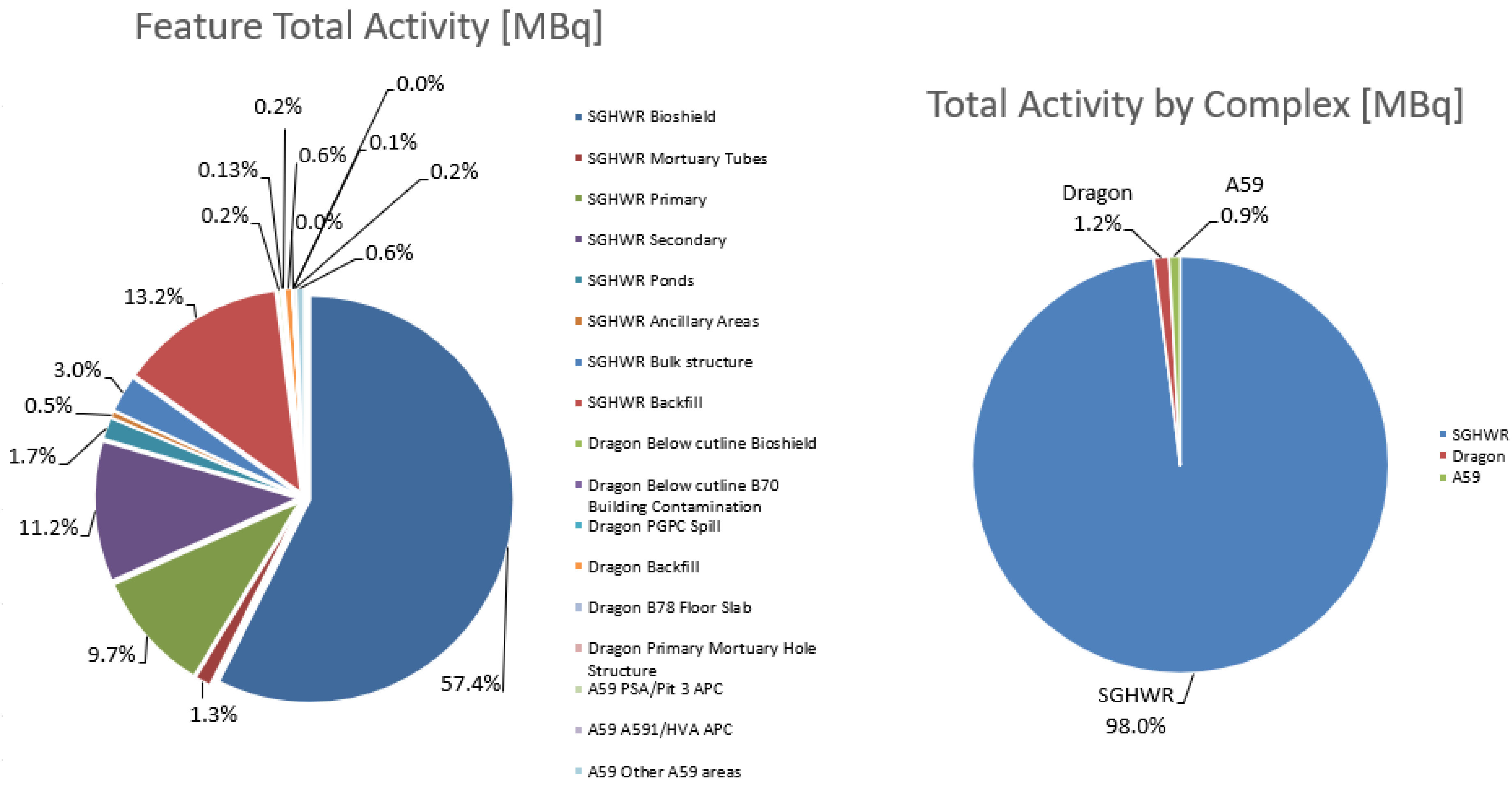


Figure 6.1: Distribution of radioactivity in the SGHWR, Dragon and A59 end state features of the Winfrith site at 01/01/2027.

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Appendix A Uncertainties, Assumptions and Gaps

Table A.1: List of the information gaps and uncertainties identified in the development of this radiological inventory report for inclusion in the central Winfrith Disposal Permit Uncertainty Management Plan Register [16]. The uncertainties relating to the inventory derivation for features and components that have both above and below cutline portions apply equally to the below cutline portion that will remain in-situ and the above cutline portion that will be demolished and used as backfill (i.e. they have not been duplicated in this table). The term "feature" is used in a more general sense in this table than in the rest of this report (and as defined in the Glossary). Uncertainties already listed in the separate A59 Inventory Report [10] are not reproduced here.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
INV-SGHWR-001	SGHWR external areas and contaminated land	Other outbuildings and structures exist in close proximity to the SGHWR, some of which have a history of contamination. These structures and their surrounding land are not included in the inventory calculations in this report. The subsurface beneath the SGHWR is inaccessible. Contaminating events with a ground impact have taken place historically in and around the SGHWR and a residual inventory is possible. Less mobile radionuclides may not be detected	It is assumed that any radiological inventory associated with external SGHWR outbuildings and structures is trivial, or that these features will be removed prior to the IEP. It is assumed that any existing contaminated land inventory beneath the SGHWR is trivial.	L	Continue routine monitoring through decommissioning to support trivial risk assumption. Ensure coordination with Site Remediation Zone Close Out.

[23] A "High" (H) rating is given if i) the uncertainty is not reduced, additional practical mitigation measure(s) is/are certain or very likely to be necessary in the near term; and/or ii) the magnitude of the uncertainty is currently such that robust demonstration of environmental safety (including optimisation) over the site life-cycle is likely to be impossible or very difficult.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
		in the groundwater monitoring programme.			
INV-SGHWR-002	Waste fingerprints	The inventory uses a number of waste fingerprints to fill gaps which may be present in the analytical dataset. A number of uncertainties are associated with use of the fingerprints including: - These are applied at a room scale; however significant heterogeneity may be present within a room/material. - Do the fingerprints capture all the radionuclides present in the waste? - Were the fingerprints derived from analysis of material that is remaining or material that has been removed? - What common determinands should be used to apply the fingerprints to the waste? - Some fingerprints (i.e. for the bioshield) are derived from modelling results which use generic input parameters.	The fingerprints have been reviewed and adjusted if possible (e.g. emphasising the concrete component of a combined paint and concrete FP) to improve applicability to the inventory features/components; they are assumed in the calculations to be appropriate for general application to the SGHWR wastes. Where there is a risk that key radionuclides (particularly those likely to drive significant impacts in the PA, such as actinides) may not be captured, alternative fingerprint assumptions are considered in the derivation of alternative inventories.	M	Capture in general uncertainty analysis of the SGHWR inventory during the assessment process. Refine fingerprints on receipt of any new information / data.
INV-SGHWR-003	Material densities	A number of material densities are included in the calculations from underlying Winfrith data sources and external sources (e.g. the bioshield	A single density value is adopted for each material based on Winfrith documentation and other literature. Where there is a range of material densities, the higher value is adopted.	L	Capture in general uncertainty analysis of the SGHWR inventory during the

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
		activation study). These are not based on site measurements.	This is conservative as the inventory derivation calculations involve the multiplication of material masses and activity concentrations.		assessment process. Where opportunities arise to gain site specific data these should be taken.
INV-SGHWR-004	Bioshield activation extent	Currently only two cores define the activity profile in the bioshield: one from the Active Tool Store (Room 245) and one from the Liquid Shut Down (LSD) Room. The Active Tool Store core is limited in extent (gaps). Both cores have only a limited analytical dataset in comparison to predicted radionuclides. Additionally, there has been no analysis or characterisation of the activation associated with radial penetrations in the bioshield, such as the ion chamber basket through tubes. Just three rebar samples were taken from the cores and only one of these is in the activation zone (at the edge).	A number of extremely conservative assumptions and simplifications were made in the bioshield inventory estimate. In particular, that the core samples represent the average activation of the bioshield concrete; this is extremely conservative as the core samples are disproportionately located toward the inner face of the bioshield. Rebar activation has been based on the SGHWR activation modelling. Bq/g values were reduced by a factor equivalent to the observed discrepancy between measured and predicted concrete ^{60}Co activity concentrations.	M	Capture in general uncertainty analysis. Obtaining new data for the bioshield would reduce this uncertainty, once access is possible. More detailed assessment of the impact of shielding could be captured if required. The adopted conservatism in the inventory could be reduced if necessary.
INV-SGHWR-005	Bioshield activation modelling	Activation modelling of the bioshield is relied upon for the inventory; however, this is based on a number of generic parameters (e.g. NRC concrete composition). Furthermore, neutron	Use of the modelling data has been restricted to providing concrete fingerprints, but rebar estimates rely on Bq/g model predictions.	M	Reliance on the modelling would be lessened if new bioshield characterisation data

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
		activation data is averaged over each radial interval of the bioshield and it is unclear whether the higher flux associated with penetrations through the bioshield have been included in the average. The fits to characterisation data are poor with the modelling over-estimating activities relative to the characterisation data.			become available. Capture in general uncertainty analysis.
INV-SGHWR-006	Adequateness of characterisation data	A significant quantity of radiological characterisation data has been incorporated into the calculations; however, no statistical analysis of the likely robustness of the dataset including number of results and spatial distribution has been undertaken. It is likely that if such a test was undertaken a number of areas would fail. The following specific uncertainties remain regarding measured activities in the bioshield cores: - In the ATS core, a fibrous layer assumed to be the flexcell joint contained elevated ^{137}Cs activity. - Rise in activities in the bioshield LSD core beyond the flexcell joint attributed to contamination from the secondary containment.	The radiological dataset, including values derived using fingerprints, is assumed to be representative of the SGHWR features. This includes assumption of the rubble mound material as OoS (the Rubble Mounds will undergo further characterisation prior to their potential disposal in the SGHWR voids). Where other relevant variables need to be determined (e.g. the depth of penetration of contamination), these typically take a precautionary approach by selecting cautiously realistic value or are based on the depth of the cores taken. Alternative inventories have been derived to capture remaining uncertainties, including in fingerprints where appropriate. All bioshield core data beyond and including those from the assumed flexcell layer are used to calculate average measured activities for	M	Capture in general uncertainty analysis of the SGHWR inventory during the assessment process. Refine fingerprints and inventory on receipt of new information/data.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
		The mechanisms leading to these readings are not fully understood. The tritium ingress inventory of the primary containment relies on the two bioshield cores which were positioned at locations of expected high neutron flux rather than to capture tritium ingress; they may therefore not be representative.	the contaminated concrete component of the bioshield and applied across its whole volume. This approach is conservative as it does not exclude any poorly-understood high measurements. The bioshield cores (over the relevant interval) are assumed to be representative of tritium ingress across the whole primary containment; this is believed to be reasonable given the high mobility of tritium.		
INV-SGHWR-007	Impact of demolition	The SGHWR demolition strategy is still in development. As such, the fate and thus location in the in-situ or backfill inventory of some walls/floors and voidage is not known. This could affect the total inventory (e.g. more material can be disposed of if densely compressed) and it could affect activity concentrations (e.g. surface contamination is re-distributed by crushing into bulk material), as well as the form and location of the activity. There is also a small uncertainty regarding the SGHWR ground level, which increases from 40.53 mAOD on the north side to 41.61 mAOD on the south side.	The assessment assumes all structures on Levels 1-3 remain in-situ. The inventory assumes all the activity from the SGHWR above-ground structure and the majority of the Rubble Mounds may be disposed of into the SGHWR voids and thus is cautious. Maximum activities are taken as maximum in-situ activities and, as such, do not allow for the effects of crushing. The higher ground level has been used in the calculations in this report, but in reality it is more likely that the demolition / final ground profile will transition across the demolition cut-line than step down. This will not change the overall inventory activity, but may slightly change the proportions assigned to in-situ structures and backfill.	M	Review impact of final demolition strategy on assumption.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
INV-SGHWR-008	SGHWR backfill	The SGHWR backfill strategy is not complete and as such it is uncertain as to what materials may be used as backfill.	It is currently assumed that SGHWR backfill will be limited to: 1. Concrete blocks and brick/concrete rubble from demolition of the above-ground (L4-10) SGHWR structure. 2. Rubble from stockpiles already outside the SGHWR. It is also assumed that the rubble stockpile material is at OoS levels.	M	Refine inventory on receipt of new information/data.
INV-SGHWR-009	SGHWR ongoing and future contamination	There are a number of activities still occurring in the SGHWR or planned to occur that may contribute to the overall disposal inventory. The most consequential of these activities is expected to be the segmentation of the reactor core, which will involve activities in a number of areas spanning the primary and secondary containments as well as parts of the ancillary areas. The contribution to the final inventory of these activities is not accounted for in the disposal inventory.	For a number of rooms with ongoing or planned active operations, no inventory contribution is derived. It is assumed any contamination arising in these areas will be decontaminated to OoS prior to demolition and disposal. In other areas, the inventory is based on what is currently known and will need to be reviewed once remaining activities are complete.	M	Refine inventory on receipt of new information/data.
INV-SGHWR-010	Uncharacterised rooms in the SGHWR	Some rooms in the SGHWR have not been characterised at all; however, these are typically low-risk rooms in relation to	Uncharacterised rooms of low risk are assumed to be inactive. Where possible, uncharacterised rooms of higher risk have inventories derived based on data for rooms	M	Refine inventory on receipt of new information/data.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
		their potential radiological impact on the inventory as most have no process history. Uncharacterised rooms that have a process history or other likely contamination pathways are subject to health physics monitoring and surveys which show activities are commensurate with the typical background for outside areas of the SGHWR.	expected to have a similar contamination profile and pathway. Uncharacterised rooms are included in the inventory estimate for the bulk structure (for tritium only in the reference inventory, and using ancillary areas average activities for all radionuclides in the alternative inventory).		
INV-SGHWR-011	SGHWR mortuary tubes characterisation	The SGHWR mortuary tubes contain active items that are yet to be removed. There are no sampling data from the mortuary tubes on which to base an inventory and the amount of contamination remaining following the removal of the items is unknown.	The presented mortuary tube inventory estimate is regarded as preliminary, a speculative inventory conservatively derived based on the potential sources of contamination. An alternative estimate considers a more conservative (in terms of radionuclides likely to be significant in the PA) fingerprint.	H	Obtain samples from the inside of the mortuary tubes. Review inventory on receipt of new information/data.
INV-DRAGON-001	Remaining Dragon structures	Other plant, outbuildings and structures exist as part of the Dragon Complex. These structures and their surrounding land are not included in the inventory calculations in this report.	It is assumed that the remaining plant and structures comprising the Dragon Complex are either radiologically uncontaminated, OoS of RSR, or will be decontaminated prior to their demolition and removal from site; there is no expectation that any other Dragon Complex below-ground concrete structures will be left in-situ at the IEP. Similarly, no inventory associated with external areas of the Dragon Complex or contaminated land is	L	Continue routine monitoring. Ensure coordination with Site Remediation Zone Close Out.

UMP Reference #	Feature, Event or Process subject to Uncertainty	Description of Uncertainty	Treatment of Uncertainty / Statement of Assumption	Originator's Rating of Potential Significance/ Impact (H/M/L)[23]	Originator's Recommended Action
			captured in this report. It is assumed that any such contamination, if present, will be removed or is OoS. Within B70 there is the potential for some low-level actinide contamination beneath the fuel carousel and fission product contamination in the steel-lined sump beneath the reactor [109, §2.1]; these areas will be characterised once they are accessible, but it is assumed here that they will be decontaminated as appropriate and so are not included in the end state inventory estimate.		
INV-DRAGON-002	Impact of demolition	The Dragon demolition strategy is not fully determined. As such, the fate and thus location in the in-situ or backfill inventory of some walls/floors and voidage is not known. This is not likely to affect the total inventory, but could affect activity concentrations (e.g. surface contamination is re-distributed by crushing into bulk material), as well as the form and location of the activity.	It is assumed that the Dragon Reactor building will be demolished to ground level and then the ground in this area re-profiled and/or additional material added to ensure at least 1 m of clean cover. Any change in this assumption will change the proportion of radioactivity in the below-ground components of the bioshield and building structure, as compared to the backfill inventory.	M	Refine inventory on receipt of new information/data.
INV-DRAGON-003	Dragon backfill	The Dragon backfill strategy is not fixed and as such it is uncertain as to what materials may be used as backfill.	The concrete and masonry waste from demolition of the above-ground portion of the Dragon Reactor (B70) and Fuel Store (B78) buildings will be used to backfill the below-	M	Capture in general uncertainty analysis of the Dragon inventory in

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			ground voids left by demolition of the Dragon building. A remaining void space of 1,099 m^3 is anticipated, which will be filled with material from the existing D630 rubble stockpiles. A density of compacted rubble (1,829 kg/m^3) has been assumed as an intermediate value between stacked blocks (2,400 kg/m^3) and loose rubble (1,500 kg/m^3). Rebar from the above-ground portion of the bioshield is currently included in the inventory and it is conservative to include it, but it would be expected that rebar that is freed during demolition would be removed for off-site management/recycling.		sensitivity analysis and during the assessment process. Refine inventory on receipt of new information/data.
INV-DRAGON-004	Adequateness of characterisation data	A significant quantity of radiological characterisation data has been incorporated into the calculations; however, no statistical analysis of the likely robustness of the dataset including number of results and spatial distribution has been undertaken. It is likely that if such a test was undertaken a number of areas would fail. Some areas have not been characterised at all; however, these are typically low-risk	The radiological dataset, including values derived using fingerprints, is assumed to be representative of the Dragon features. The fingerprint derived for application to the entire building structure is particularly dependent on the distribution and robustness of the characterisation data, and there has been no radiological characterisation of the mortuary holes. Further characterisation to reduce these uncertainties could be undertaken as decommissioning proceeds. Nonetheless, the approach taken in	M	Capture in general uncertainty analysis of the Dragon inventory in sensitivity analysis and during the assessment process. Refine fingerprints and inventory on receipt of new information/data.

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		in relation to their potential radiological impact on the inventory. There are no recent data from B78 that can be used to constrain the contamination level. Regarding the bioshield, no samples have been taken from areas known to contain barytes concrete, which may be indicative of higher activation levels. Thus, potentially high activation areas may not be accounted for in the inventory estimate.	developing fingerprints and assumptions made regarding their application are generally conservative. Where other relevant variables need to be determined (e.g. the depth of penetration of contamination), these typically take a precautionary approach by selecting a cautiously realistic value. Alternative inventories have been derived to capture remaining uncertainties, including in fingerprints where appropriate.		
INV-DRAGON-005	Dragon bioshield activation modelling	No activation modelling of the Dragon bioshield has been undertaken. Activation modelling of the SGHWR bioshield is used to support the inventory; however, this is for a different reactor and is based on a number of generic parameters.	Activation modelling results produced for the SGHWR have been used to inform derivation of the Dragon bioshield fingerprint. However, the SGHWR is a different reactor type to that of Dragon, used different fuel and operated at different energies and over different periods, and has been shut down for considerably longer. In addition, the type of concrete and steel used in the model derive from generic USA specifications. Therefore, the activation data can only be considered to give a general indication of possible activation in the Dragon bioshield. Use of the modelling data has been restricted to providing fingerprints only.	L	Reliance on the modelling is limited and, given the much smaller total inventory compared to SGHWR, the overall impact will be small. Capture in general uncertainty analysis.

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INV-DRAGON-006	Dragon bioshield material composition	The exact specification and extent of the ordinary concrete, barytes concrete and rebar in the bioshield is not known.	The bioshield fingerprints have been developed assuming generic material specifications and informed guesses, but different material specifications will have a significant impact on the rebar and barytes concrete fingerprints given the lack of characterisation data and limited material knowledge. The volume of barytes concrete modelled is thought to be bounding as it is believed to be limited to surrounding penetrations through the bioshield, but only a few technical drawings indicate this.	M	Capture in general uncertainty analysis of the Dragon inventory in sensitivity analysis and during the assessment process. Refine fingerprints and inventory on receipt of new information/data.
INV-DRAGON-007	Dragon (B70 and B78) building surface contamination	The proportion of surface contamination inside B70 and B78 is thought to be low, but a systematic survey to confirm this has not been identified. There is potential for Pu isotopes to be included in the fingerprint, but no direct evidence for this from the sample results available. All approaches to including them are subject to significant uncertainty and would result in a significant Pu inventory that is not justified. It is unclear whether one anomalously high ^{3}H result from the Betalite store area	The approach applied to calculating the Dragon building surface contamination inventory has been to use the derived fingerprints with the measured highest activity patch. This is an extremely pessimistic approach as it applies the highest measured hotspot contamination to the entire Dragon building. Therefore, an assumption is made that only 5% of the surface activity is present – this assumption is made on the basis that the building does not have any significant contamination, but underpinning with additional sampling or uncertainty analysis in the PA is advised, particularly as the building	M	Capture in general uncertainty analysis of the Dragon inventory via a range of sensitivity analyses (in this report) and during the assessment process. Review inventory on receipt of new information/data.

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		should be included in the Betalite area fingerprint.	contamination contributes significantly to the total Dragon inventory. The presence of Pu has not been inferred and it is excluded from the Dragon general building contamination fingerprint at this time. However, the updated FP-002 Dragon General Area fingerprint has been used to develop an alternative inventory that does include an estimate for Pu. For the reference inventory estimate, the single anomalously high ^{3}H result is assumed to be in error and is not included in fingerprint. However, it is included in the alternative inventory for the Betalite store area.		
INV-DRAGON-008 (incorporates UMP/MH/001 and UMP/MH/002 from [160])	Dragon mortuary hole system characterisation	a) The fingerprint of the fixed contamination and the ratio of loose to fixed contamination in the Dragon mortuary holes is uncertain. b) The pick-up efficiency of smears and the appropriate surface area for the full-height smears are uncertain. c) There has been no direct characterisation of some parts of the system including bottom cross vents.	a) It is assumed that the fingerprint of the fixed contamination is equivalent to the loose contamination. b) The assumed pick-up efficiency of 10% and the adopted full-height smear area of 300 cm^2 are both considered to be conservative. c) No account has been taken of the potential for accumulations or increased contamination in the bottom corners of the system or in the bottom horizontal ventilation linking ducts.	L	Collect and analyse a metal coupon from a bottom cross vent. If required, complete a separate assessment of the fixed component of MH contamination. Capture remaining uncertainty in general uncertainty

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		d) It is unclear how to characterise the contamination volume in order to calculate activity concentration.	d) There alternative contamination volumes have been considered: i) the first 1 mm of the entire steel structure; ii) loose contamination averaged over the infill volume; iii) the planned concrete monolith comprising the entire pit in which the steel structure and grout infill will sit.		analysis of the Dragon inventory in sensitivity analysis and during the assessment process.
INV-DRAGON-009	B78 below-ground features	This inventory assessment does not include i) the metal tube system for additional mortuary holes (tubes 1-40, used for the storage of fresh fuel), ii) the metal lining of the storage pit, or iii) the bulk system concrete into which the mortuary holes are set.	i) It is assumed that the metal tube system for additional holes will be removed and then cleaned should any contamination remain. ii) It is assumed that the metal lining of the storage pit will be removed and the area cleaned. iii) Given that the mortuary holes, sump and storage pit are metal-lined, it is expected and assumed that there has been negligible radionuclide migration in to the bulk system concrete.	L	Refine inventory if/as needed on receipt of new information/data.
INV-DRAGON-010	Residual contamination from the PGPC contaminated water spill	As characterisation and clean-up of the contamination is ongoing (the spill occurred in March 2021), there are many uncertainties relating to this feature, primarily: - Areal extent and penetration depth into the concrete floor of the contaminated water. - Total activity of the contamination.	The present estimate of the PGPC spill inventory is regarded as preliminary pending further characterisation data. - Area and penetration depth are assumed to be 3.3 m^3 and 10 mm respectively, based on current best (conservative) estimates.	H	Refine inventory if/as needed on receipt of new information/data.

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		- Radionuclides present/fingerprint. - Extent of decontamination to be carried out during decommissioning.	- Total activity is taken to be the upper value of the range currently estimated (24 GBq). - Dragon primary coolant fingerprint is assumed to be representative. - Inventories are presented assuming 0%, 90% and 99% of contamination is removed, as well as a level of decontamination sufficient to reduce activity concentration to the upper LLW limit (equivalent to 95.5% removal). The 95.5% decontamination case has been taken forward for inclusion in the total Winfrith end state radiological inventory.		
INV-DRAGON-011	Contaminated surface area and other dimensions	To simplify calculations (and/or where precise information is not available), several dimensional approximations have been made.	Approximations include: - Assumption that there is equal surface area on every floor of the Dragon Reactor building, in order to calculate below-ground surface area. - Betalite store area floor calculated as a rectangle. - - Simplified wall pattern assumed for B78.	L	None
INV-A59-001	Remaining contamination in A59 area	Although the current reference inventory estimate suggests that all A59 features are already at OoS levels, there remain many uncertainties in this estimate (listed in	The reference inventory estimate (derived in [10]) is already OoS for all three A59 features. An alternative inventory estimate has been derived assuming maximum activity	L	Refine inventory if/as needed on receipt of new

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		[10]) and future characterisation may show otherwise. If needed, the A59 area (in particular the two current APCs) will be remediated to OoS levels.	concentrations, then scaling so that activity concentrations just meet OoS criteria.		information/data or if strategy changes.

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