



Geological Disposal Review of Alternative Radioactive Waste Management Options

Main Report

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List of Acronyms

ANSTO	Australian Nuclear Science & Technology Organisation
CoRWM	Committee on Radioactive Waste Management
DBD	Deep Borehole Disposal
EDRAM	International Association for Environmentally Safe Disposal of Radioactive Materials
ERDO	Association for Multinational Radioactive Waste Solutions
EURAD	European Joint Programme on Radioactive Waste Management
FA	Fly Ash
GDF	Geological Disposal Facility
GBFS	Ground Blast-Furnace Slag
GWe	Gigawatt Electric
HAWTT	Higher Activity Waste Thermal Treatment
HIP	Hot Isostatic Pressing
IAEA	International Atomic Energy Agency
IGD	Inventory for Geological Disposal
HLW	High Level Waste
HSR	Higher Strength Rock
ICV	In-Container Vitrification
ILW	Intermediate Level Waste
IPT	Integrated Project Team
ISO	International Standards Organisation
IWMP	Integrated Waste Management Programme
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LSSR	Lower Strength Sedimentary Rock
MKPC	Magnesium Potassium Phosphate Cement
NDA	Nuclear Decommissioning Authority
NDPB	Non-Departmental Public Body
NNL	National Nuclear Laboratory
NSD	Near-surface disposal
NWS	Nuclear Waste Services
OPC	Ordinary Portland Cement
PCM	Plutonium Contaminated Material
PVC	Polyvinyl chloride
RWM	Radioactive Waste Management Limited
R&D	Research and Development
SCM	Supplementary Cementitious Materials
SIXEP	Sellafield Site Ion Exchange Effluent Plant
SKB	Svensk Kärnbränslehantering Ab: Swedish Nuclear Fuel and Waste Management Ltd.
SFR	SKB Final Repository for short-lived radioactive waste
VLLW	Very Low Level Waste

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Executive Summary

This report summarises the consideration of Nuclear Waste Services on alternative options to geological disposal for managing the Inventory for Geological Disposal (1). It seeks to update our last consideration in 2017 (2), reflecting the policy set out in the UK policy framework for managing radioactive substances and nuclear decommissioning (3), and recommendations from the Committee on Radioactive Waste Management, which identify the need to maintain an understanding of alternatives to geological disposal.

The scope of the review has been limited to highlighting some key developments in:

- Alternative disposal options for some wastes destined for geological disposal. These include near-surface disposal and deep borehole disposal concepts.
- Alternative treatment techniques for radioactive wastes which could alter the nature and/or reduce the volume of waste requiring geological disposal (referred to in this report as “Waste Treatment Techniques”).

The key conclusions and considerations that emerge from this report are:

- Near-surface disposal facilities have been operated by several nations to manage Low Level Waste and shorter-lived Intermediate Level Waste in their radioactive waste inventories. Although several such facilities are operational globally, Centre de la Manche in France is the only facility, managing Intermediate Level Waste, to have completed its disposal mission and have been brought to a state of closure. In the UK, near-surface facilities are currently operated for the disposal of Low Level Waste, primarily the Low Level Waste Repository. However, the Nuclear Decommissioning Authority have considered the benefits of developing further near-surface facilities for disposing of a proportion of the less hazardous waste in the Intermediate Level Waste category. Development of such facilities in England and Wales was enabled by the UK policy framework for managing radioactive substances and nuclear decommissioning, published in May 2024 (3). Therefore, whilst near-surface disposal does not remove the need for a Geological Disposal Facility, a proportion of the less hazardous waste in the Intermediate Level Waste category, currently destined for geological disposal, could potentially be re-routed to near-surface disposal. This may enable earlier hazard reduction and overall cost savings, subject to a rigorous safety case. NWS recognises the need to further optimise use of the disposal capacity at the Low Level Waste Repository site, in parallel with capping of the Repository which commenced in 2024.
- Progress in deep borehole technology has comprised primarily laboratory and concept based studies, and the approach remains at an early stage of technical maturity. Strategic studies have considered the potential application against some national radioactive waste inventories. Our assessment is that deep borehole disposal could not displace the need for a Geological Disposal Facility in the UK as it could not realistically accommodate the entire inventory for disposal. At the time of writing, deployment of deep borehole disposal in a nuclear environment would require further policy and legislative consideration, both of which would be a matter for government, and subsequent strategy development. Regulatory approaches would also have to be



developed to ensure proper regulation of operational safety and security alongside long-term environmental protection. Nevertheless, in nation states with a smaller inventory of higher activity waste, comprising primarily spent nuclear fuel, deep borehole disposal may be a future alternative to development of a Geological Disposal Facility. However, deep borehole disposal remains conceptual and has no licensing basis at the current time. Nuclear Waste Services will maintain its position of keeping a watching brief on future developments in deep borehole disposal.

- Notable progress has been made in development of alternative treatment options for radioactive wastes, particularly using thermal treatment and enhanced encapsulation technology. These technologies are compatible, in principle, with treatment of a significant fraction of the Intermediate Level Waste inventory, in particular, problematic waste streams for which no, or sub optimal, treatment options are currently available. Thermal treatment methods have the potential to reduce the volume of some wastes requiring interim storage and emplacement in a Geological Disposal Facility, and achieve improvement in passive safety. Inactive and active variants of the technology have been operated at full scale, or demonstrated at pilot scale, for radioactive waste management. Together with laboratory proof of concept work, this research has established an understanding of the key benefits, opportunities and challenges of deployment of thermal treatment technology in a UK context.

Notwithstanding the advances in all the alternative radioactive waste management options discussed herein, none remove the need for a Geological Disposal Facility. The alternative radioactive waste management options reviewed here, may, in the future, enable the Inventory for Geological Disposal to be further optimised or reduced. The associated cost savings clearly need to be considered against the marginal cost of emplacement of the waste inventory in a Geological Disposal Facility, in addition to due consideration of potential earlier waste emplacement, whole lifecycle costs and other strategic drivers. In developing alternative radioactive waste management options, it will be important to engage public awareness and understanding of technology developments.

Nuclear Waste Services maintains a watching brief on the development of technologies that might have the potential to reduce the Inventory for Geological Disposal and/or act as alternative management/disposal options. If any such technologies emerge and reach a suitable level of maturity, then Nuclear Waste Services will initiate an internal process to engage with and actively support their development, if it is felt that they offer suitable potential benefits to the management of the Inventory for Geological Disposal. The development of potential thermal treatment and near-surface disposal options has followed this process.

1. Introduction

1.1 Background to radioactive waste management in the UK

Responsibility for Government policy on radioactive waste management is devolved to the Welsh and Scottish Governments and the Northern Ireland Executive in their respective countries. The UK Government has responsibility for the policy in England only, with the Department for Energy Security and Net Zero being the government department with policy responsibility for the management of radioactive waste and materials.

The Nuclear Decommissioning Authority (NDA), a non-departmental public body (NDPB) sponsored by the Department for Energy Security and Net Zero, implements policy on the management of radioactive waste from the nuclear industry, through its strategy, on behalf of the UK Government and the devolved administrations of Scotland and Wales. There are no nuclear facilities in Northern Ireland.

Nuclear Waste Services (NWS), as a wholly owned subsidiary, forms part of the NDA group, and is tasked with supporting the safe and secure long-term management of the UK's nuclear waste.

In England, Wales and Northern Ireland the policy is for geological disposal of the UK inventory of High Level Waste (HLW) and Intermediate Level Waste (ILW), and a small amount of Low Level Waste (LLW) not suitable for disposal at the LLWR site. Geological disposal involves isolating radioactive waste deep inside a suitable rock volume to ensure that no harmful quantities of radioactivity ever reach the surface environment. This is achieved using multiple barriers, that work together to provide the necessary containment and protection, including:

- the form of the radioactive waste itself, for example solid HLW glass
- the waste packaging, including the container and any overpack
- the engineered barriers (buffer, backfill, plugs and seals)
- the stable host rock, within which the facility is developed

Delivery of a GDF, up to the point of closure, will be coupled with safe and secure interim storage of wastes and ongoing Research and Development (R&D) to support optimised implementation.

Development of a GDF will be achieved through a consent-based siting process, with willing community partners. This is set out in the Working with Communities policies for England and Wales in Appendices 1 and 2 of the UK policy framework for managing radioactive substances and nuclear decommissioning (3).

There are currently no plans to site a GDF in Northern Ireland and future policy decisions in relation to geological disposal in Northern Ireland would be a matter for the Northern Ireland Executive.



Current Scottish Government policy is that the long-term management of higher activity waste (excluding HLW, which is not located in Scotland) should be in facilities which are located as near to the site where the waste is produced as possible, and in the near-surface environment¹ (4).

As outlined in the NDA Strategy published in 2021 (5), there is an ongoing commitment to optimise the management of radioactive waste to reduce costs and disposal timelines, alongside the development of a GDF (3):

“...NDA [Nuclear Decommissioning Authority] and NWS [Nuclear Waste Services] continue to review appropriate solutions including learning from and engaging with overseas programmes, which could have the potential to improve the long-term management of some of the UK’s radioactive waste.”

The need to review appropriate solutions and learning is also set out in the UK policy framework for managing radioactive substances and nuclear decommissioning, published in May 2024 (3), and stems from the government response to advice provided by the original Committee on Radioactive Waste Management (CoRWM)², an advisory committee to the UK Government and devolved administrations (6). In response to CoRWM Recommendation 5, the joint government response stated:

Recommendation 5: The commitment to ensuring flexibility in decision making should leave open the possibility that other long-term management options (for example, borehole disposal) could emerge as practical alternatives. Developments in alternative management options should be actively pursued through monitoring of and/or participation in national or international R&D programmes.

Government recognises the need to take account of developments in storage and disposal options, as well as possible new technologies and solutions, and the need therefore to ensure that there is flexibility of decision-making in a process which is expected to take a number of decades. Government and the NDA will develop a framework which allows for the flexible delivery of a geological disposal programme. The framework will include monitoring of international Research & Development (R&D) programmes into safe and secure treatment and storage of waste and geological disposal technology as well as any new options that emerge (e.g. the use of deep boreholes for the disposal of some wastes).

¹ Scottish Government policy defined Higher Activity waste as “Radioactive waste defined in current UK categorisations as Intermediate Level Waste. Intermediate Level Waste is waste which has radioactivity levels exceeding the upper boundaries for Low Level Waste and which does not generate enough heat for this to need to be taken into account in the design of treatment or storage or disposal facilities” (4).

² CoRWM is an advisory NDPB. The original CoRWM remit, up until 2007, was to oversee a review of options for managing solid radioactive waste in the UK and to recommend the option, or combination of options, that could provide a long-term solution, providing protection for people and the environment. Its objective was to arrive at recommendations that could inspire public confidence and be practicable. Its current remit is to provide independent scrutiny and advice to UK government and devolved administrations on the long-term management of radioactive waste, including storage and geological disposal.

This report summarises our consideration of some alternative options to geological disposal for managing the Inventory for Geological Disposal (IGD) (1), building on our last published consideration in 2017 (2), and taking account of developments in the policy framework, summarised above. Our review has built on existing knowledge and understanding and our report highlights some key developments of significance, communicating our position in three broad domains of Near-Surface Disposal (NSD), Deep Borehole Disposal (DBD), and alternative treatment options. Our focus is on alternative approaches that are relevant to the UK context and have received the most significant attention since our last consideration. This review is not intended, therefore, to be a complete and comprehensive account of technology developments and our consideration over this period.

1.2 Waste Inventory

In the UK, types of radioactive wastes are classified according to the nature and quantity of radioactivity, and the heat produced. The wastes which comprise the IGD (1) and which are therefore pertinent to this report, are:

- HLW: waste in which the temperature may rise significantly as a result of its radioactivity and has to be taken into account in the design of storage or disposal facilities. This waste results from reprocessing activities.
- ILW: waste exceeding the upper boundaries for LLW which does not generate sufficient heat to be taken into account in the design of storage or disposal facilities.
- The small proportion of LLW which is not suitable for disposal in a near-surface facility such as at the LLWR site.

Another component of the current IGD, published in 2025, is nuclear material that is not currently classified as waste but could be in the future, if deemed to have no further use. These materials are included in the IGD for planning purposes:

- Potential wastes such as spent fuel, plutonium inventory and uranium stocks.

In January 2025, The Department for Energy Security and Net Zero announced that it will work with the Nuclear Decommissioning Authority (NDA) to immobilise the UK-owned civil separated plutonium inventory at Sellafield (7).

The current IGD also includes estimated waste arisings from a new nuclear build programme, currently up to 24 GWe capacity. The IGD will be updated in the future to reflect additional new nuclear projects.

Note that wastes managed under Scottish Government policy are not included within the IGD.

1.3 Optimisation of waste management

NDA published a Radioactive Waste Management Strategy in 2019 (5). Going forward, NDA will seek to take a more risk-informed approach to radioactive waste management based on the nature of the waste and the hazard that the properties of the waste pose to people and the environment. This is in contrast to earlier practice, where the waste classification system prescriptively guided the waste management solution. The risk-informed approach will enable wider and deeper application of the “*waste hierarchy*” (Figure 1) and will allow NDA to optimise

the use of appropriate treatment, storage, and disposal infrastructure. This approach is reflected in Nuclear Waste Services' strategic objectives to:

- achieve the right waste form, in the right package, managed or disposed of at the right facility
- enable accelerated decommissioning through innovation, working with waste producers to overcome challenges and to capitalise on new opportunities
- facilitate integration across the NDA group and the supply chain, delivering our mission in an optimised way, by providing industry-wide solutions to radioactive waste challenges

In accordance with the waste hierarchy, for existing and any new facilities, it is a requirement to minimise the occurrence of future waste. However, it is recognised that while disposal is the least preferred option, without disposal capability the NDA's mission would be impossible to complete (5).

The IGD shows that only 8% of the UK Radioactive Waste Inventory is destined for a GDF, based on stored volume. Viable alternative management options for the LLW and ILW not yet packaged for GDF, could optimise the IGD, for example by reducing the volume and number of packages to be emplaced in the GDF, and divert wastes appropriate for any future NSD facility. This approach is an implementation of the waste hierarchy illustrated in **Figure 1**.



Figure 1 The Waste Hierarchy (5).

1.4 Scope

This report focuses on alternative options to geological disposal that are currently under investigation or development by external actors, both in the UK and internationally.

As defined by the UK Government (3), geological disposal “involves isolating radioactive waste deep inside a suitable rock formation to ensure that no harmful quantities of radioactivity ever reach the surface environment. This is achieved through the use of multiple barriers that work together to provide protection over hundreds of thousands of years”.



The alternative management options discussed in this report include waste treatment techniques and two of the alternatives to geological disposal that were highlighted by CoRWM (8) for potential ongoing review of developments:

- Alternative treatment techniques to waste which could alter the nature and/or reduce the volume of waste requiring geological disposal (referred to in this report as “Waste Treatment Techniques”)
- Alternative disposal options for waste destined for geological disposal. These include NSD and DBD concepts.

The International Atomic Energy Agency (IAEA) safety standards (9) recognise three options, tailored to risk, for the disposal of the radioactive waste categories reviewed in this report, i.e. HLW, ILW and some LLW. These options are near-surface, deep borehole, and geological disposal. Note that borehole disposal is considered by IAEA in the context of sealed sources and at shallow depth, this contrasts with the specification of DBD considered here for the most hazardous radioactive waste at depths greater than 1 km.

If an alternative option to geological disposal is identified and assessed as being sufficiently mature that it warrants further consideration, an internal gated decision-making process would be initiated. This would be escalated by the appropriate governance route which includes relevant stakeholders such as NDA. The first stage of the process would involve a case for change being created. If it was assessed as being beneficial to the mission, NDA could then consider the need for an alternative strategy. If necessary, NDA would also work with the UK Government and relevant devolved administrations to understand if a policy change and/or regulatory change was required.

A future approach to managing radioactive waste through partitioning and transmutation is not considered in detail in this report, given the early stage of conceptual development and incompatibility with UK legacy wastes. The current international consensus is that the opportunity for application of this technology is in combination with future closed fuel cycles, enabling more sustainable nuclear power generation (10). Feasibility studies of partitioning and transmutation are continuing, aiming for an industrialisation of the process by 2050. The nature and complexity of the UK’s radioactive waste legacy means it is unlikely that partitioning and transmutation could be applied to improve radioactive waste management outcomes. The application of such technology for treating spent nuclear fuels from future nuclear energy generation would be a matter for the owner of the fuel, however, a GDF will likely be required to manage the residual fraction of long-lived radionuclides arising from this process.

1.5 Objectives

This report summarises the outcome of our periodic review of technological developments of alternative radioactive waste management solutions to a GDF, in the context of the Working with Communities policies for England and Wales in Appendices 1 and 2 of the UK policy framework for managing radioactive substances and nuclear decommissioning (3). It builds upon work undertaken in 2017 (2) and is informed by subsequent technical developments in alternative waste management techniques, within the updated framework of NDA Strategy 4 (5), NDA Radioactive Waste Strategy 2019 (11), and NDA Strategic Position on Radioactive Waste Treatment (12). An assessment of technology readiness level is made in broad terms as (13):



- Low: technology readiness level 1-3, meaning proof of concept or earlier stage of technology development.
- Medium: technology readiness level 4-6, meaning between bench scale and near full scale demonstration using simulants.
- High: technology readiness level 7-9, meaning between full scale demonstration of a final design using inactive simulants under expected operating conditions, to an operational active facility.

A less granular scale is applied here to reflect the uncertainty in assessment and in the maturity of the integration of component sub-systems.

This report is published for the benefit of stakeholders and organisations interested in radioactive waste disposal, including waste producers, facility operators, local planning authorities, academic research groups, communities where radioactive wastes are generated or are managed or those involved in the GDF siting process.

The options discussed in this report are presented in the context of providing potential alternatives to geological disposal. At the present time, no option reviewed, alone or in combination, could avoid the need for a GDF for the majority of the most hazardous radioactive waste in the IGD (by volume). As discussed in more detail in subsequent chapters, the different options could be suited to specific categories of the radioactive waste destined for geological disposal or specific waste types within these categories. The waste treatment techniques described are currently investigated for their application to certain ILW, providing a potential alternative to the current typical practice of cementation. NSD is anticipated to only be appropriate for a proportion of the less hazardous waste in the ILW category, due to the nature of NSD, and its ability to provide long-term containment from the environment. DBD is considered only for the possible disposal of HLW, spent nuclear fuel, and plutonium which are potentially or conceivably dimensionally compatible with borehole technology.

1.6 Report structure

The report is structured as follows:

- Section 2 reviews the development and maturity of NSD concepts.
- Section 3 reviews the development and maturity of DBD concepts.
- Section 4 reviews emerging waste treatment technologies and methodologies of radioactive waste treatment for the benefit of the IGD.

2. Near Surface Disposal

2.1 Position Statement

Near-surface disposal for some higher activity wastes is already within policy on radioactive waste management in Scotland. A single UK policy framework for managing radioactive substances and nuclear decommissioning, was published in May 2024 (3). The policy framework sets out requirements for the management of all solid radioactive waste, that include:

- For those responsible for creating and managing solid radioactive waste to apply a risk-informed approach to decision making.
- For the waste hierarchy to be used as a framework for decision-making for all radioactive waste across the UK.
- For less hazardous ILW to be disposed of in a near surface facility where it is safe to do so, within England or Wales.

In principle, near-surface disposal could be compatible with a proportion of the less hazardous waste in the ILW category, within the IGD. Therefore, a Geological Disposal Facility will always be required, even if near-surface disposal of some ILW wastes were to be implemented.

Near-surface disposal at or just below the surface, such as the vault system at LLWR or adjacent to Dounreay, is a mature option for the management of LLW; this is referred to as at-surface near-surface disposal. Internationally there is also a mature understanding of the options for adopting near surface-disposal approaches for the management of short-lived or less hazardous ILW that does not require the isolation and containment afforded by a GDF. There is significant UK and overseas experience of at-surface near-surface disposal systems.

Near-surface disposal at a depth of several tens of metres, has been licensed and used across Europe; this is referred to as at-depth near-surface disposal. For example, the Svensk Kärnbränslehantering Final Repository (SFR) for short-lived radioactive waste. SFR, in Sweden, was developed for the disposal of LLW and short-lived ILW; this disposal concept is considered to be of high maturity.

It is noted that although at-surface and at-depth near-surface disposal concepts are at high maturity, further optimisation and development may be desirable or required.

Current Scottish Government policy is for long-term management of higher-activity radioactive waste to be in near-surface facilities located as near as possible to the site where the waste was produced.

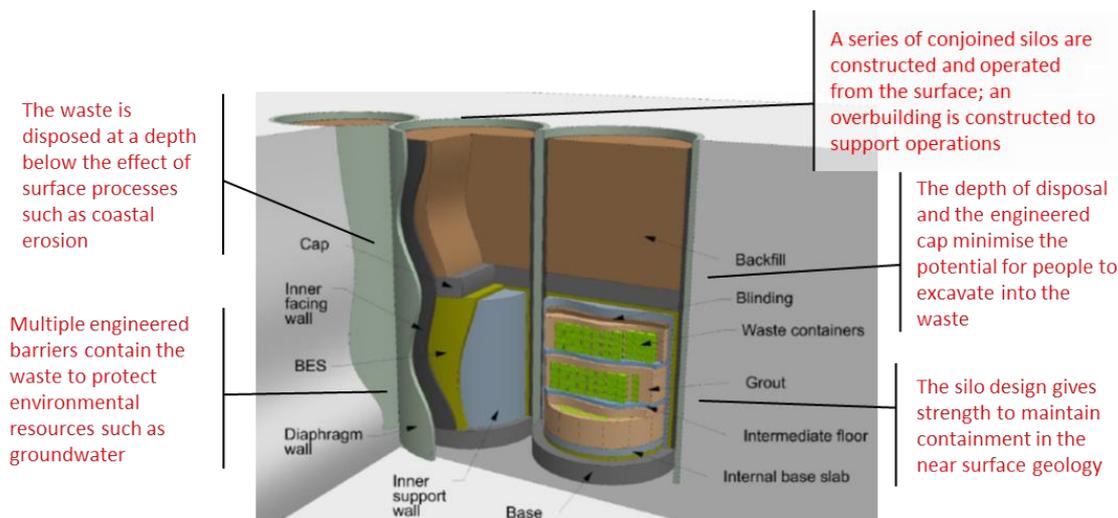


Figure 2 Schematic illustration of an at-depth near-surface disposal silo. “BES” refers to Bentonite Enhanced Sand (14).

2.2 Generic Concept overview

An NSD facility for ILW is a facility that can be located at or below the surface (up to 200 metres, the minimum depth of a GDF), and may make use of existing structures (3). It differs from a GDF in the degree of isolation provided by the facility, including host geology, depth and design (15) (14). The facility design, depth and dimensions will depend on the site for implementation. Two generic facility concepts are considered to be viable credible options for implementation in the UK:

- Disposal vaults at-surface level, referred to as at-surface NSD.
- Disposal vaults or silos several tens of metres below ground, referred to as at-depth NSD.

These generic concepts, which were explored to assess the potential environmental and cost benefits are further described in Sections 2.5 and 2.7.

In the case of geological disposal, the depth of the facility comprises an important component of the safety concept, providing isolation over much longer timescales than is feasible for near-surface facilities. For example, providing protection against surface processes such as the effects of climate change, coastal erosion, glaciation, and human intrusion. The engineered barriers of a geological facility, work together with the geological barrier, to provide containment and protection of the radioactive waste packages. In NSD facilities, long-term safety of the facility relies on engineered barriers to contain the radioactive content of the waste for as long as necessary to demonstrate safety. The packaging, backfill and concrete structure of the disposal vaults reduce the amount of radioactivity that can be released into the environment.

2.3 Potential opportunities

UK classification of radioactive wastes, as LLW or ILW, covers a broad range of nuclear, radiological and chemical hazard, but does not always reflect the full nature or risks posed by the wastes. Adopting a risk informed approach to wastes at the LLW / ILW boundary will be essential for optimising the management of large volumes of waste arising from dismantling of nuclear power stations and facilities. NSD facilities could provide opportunities for a proportion



of the less hazardous waste in the ILW category. This could potentially enable the earlier disposal of stored wastes thereby releasing storage space from Nuclear Licensed Sites. NSD facilities could, in principle, divert a proportion of the less hazardous waste in the ILW category from the GDF, reducing the volume of waste that needs to be disposed of in a GDF.

2.4 Compatibility with current UK Government and Devolved Administration policies

As outlined in its Strategy, for wastes originating in England and Wales, the NDA is committed to exploring NSD options for a proportion of the less hazardous waste in the ILW category (11) (5).

Current Scottish Government policy is that the long-term management of higher-activity radioactive waste should be in near-surface facilities located as near as possible to the site where the waste was produced (note that HLW is not present in Scotland).

Currently, NDA use near-surface facilities for the disposal of LLW and Very Low Level Waste (VLLW). Commercially operated landfill sites are used for VLLW and some lower-activity LLW disposal. Both the LLWR site in Cumbria and the LLW Disposal Facility in Dounreay are used for the disposal of LLW.

NDA has been investigating the benefit of near-surface capability and implementation of this approach in England and Wales was enabled by the single UK policy framework for managing radioactive substances and nuclear decommissioning, published in May 2024 (3).

2.5 Near-surface disposal facility concepts

The NDA NSD facility programme explored four general concepts of NSD facility suitable for a proportion of the less hazardous waste in the ILW category (14); all could offer safe, secure isolation and containment for specific types of waste, with facility depth and dimension being dependent on local site conditions.

Disposal vaults at surface level

In this at-surface NSD concept, waste packages are stacked in shallow engineered concrete vaults up to the approximate level of the surface, such as the vault already constructed at the LLWR site. The closure concept is an engineered cap, designed to minimise the amount of rainwater entering the vault and deter inadvertent intrusion, such as that planned for the LLWR site.



Figure 3 Surface vault system at the LLWR site (16).

Disposal vaults several tens of metres below ground

In this at-depth NSD concept, waste packages are placed in a series of rectangular vaults constructed up 200 metres below the surface. The vaults consist of multiple barriers to provide containment and resistance to groundwater. Once complete, an isolation layer would cover the waste together with thick reinforced inner and outer caps, mass backfill and earth landscaping.

Disposal silos several tens of metres below ground

In this at-depth NSD concept, waste packages are placed in a series of silos tens of metres below the surface, employing multi-barriers and an isolation layer. A number of cylindrical silos could be constructed with a multiple barrier system to provide containment and minimise contact with groundwater, an example being the proposed facility at *Vrbina-Krsko, Slovenia*.

Disposal caverns several tens of metres below ground³

In this at-depth NSD concept, waste packages are placed in rock caverns, for example via horizontal tunnels that extend from the base of shafts. Each cavern has independent multiple-barrier containment and, once full, each cavern is sealed to prevent future intrusion and limit groundwater inflow. The Loviisa facility, in Finland, and SFR facility, in Sweden, both have operational cavern concepts.

2.6 Current experience

NSD facilities for radioactive waste have been licensed and operated in several countries for over 50 years (17). NSD facilities suitable for the disposal of VLLW and LLW include trenches,

³ Construction of the underground caverns is likely to be more difficult and could pose additional health and safety issues for the workforce. As such, NDA is not currently exploring disposal to caverns as a viable option, for the ongoing investigatory work in England and Wales, although they will continue to keep possible options under review.



vaults, shafts, and direct access silos, as well as natural and engineered subsurface structures such as caverns, drifts, and tunnels.

Currently the existing UK at-surface NSD facilities are only used for LLW (or VLLW) disposal. At-depth NSD has not, so far, been implemented in the UK.

Near-surface engineered facilities that can accept less hazardous waste in the ILW category are also in operation in several countries, such as vaults and near-surface silos (14) (17). There is, therefore, worldwide experience of the construction and operation of such facilities for VLLW, LLW and ILW.

2.7 Technical maturity

2.7.1 At-surface disposal

There is significant UK experience of LLW disposal in at-surface NSD facilities both at the LLWR site and LLW Disposal Facility in Dounreay. This technology is considered to be of high technical maturity.

Low Level Waste Repository site

Waste has been disposed of at the LLWR site since 1959 (18), firstly in trenches and latterly in concrete vaults, situated above the water table. Wastes are disposed of in half-height ISO containers with any accessible voids within the containers being filled with a cement-based grout prior to final disposal. The main element of the closure concept is the installation of an engineered cap that will be progressively constructed over the disposal vaults and adjacent trenches. The cap serves to reduce the infiltration of rainwater into the wastes and to maintain unsaturated conditions for several hundred years.

LLW Disposal Facility in Dounreay

Waste has been disposed of at the site since April 2015 in two engineered concrete vaults, constructed approximately 16 metres below the ground surface (19). Groundwater level is controlled by pumping whilst the vaults are being filled. The first vault is used for LLW arising from demolition, which mainly comprises lightly contaminated building rubble and soils, contained within one-tonne bags that are placed directly into the vault. The second vault receives various contaminated items of bulk LLW as compacted LLW in half height ISO containers. Any accessible voids in the containers are filled with a cement-based grout. When both vaults are full and closed, groundwater control will cease, and an engineered cap and low-permeability layer will be constructed (19).



Figure 4 Aerial photograph of Dounreay LLW Disposal Facility (19).

There are several examples of similar international NSD facilities that accept LLW and short-lived ILW. For example:

Centre de l'Aube, France

The facility comprises a series of above-ground concrete vaults in which containers of waste are stacked by overhead crane. The facility receives both LLW and short-lived or less hazardous ILW. On completion, each of the vaults is backfilled with gravel. A reinforced concrete cap is poured once each vault has been filled and an engineered cover system will be installed once waste placement is complete at the facility.



Figure 5 Centre de l'Aube, France (20); image by permission of and © Andra / ADN.

El Cabril facility, Spain

Disposal cells are constructed of concrete and protected by a mobile roof during operations. The facility receives both LLW and short-lived or less hazardous ILW. Once the disposal cell is full, the space between the disposal units is filled with gravel and a slab is constructed over the top. When a set of cells is filled, a multi-layered engineered cap is constructed enclosing the whole disposal volume.



Figure 6 Disposal of LLW and ILW at El Cabril, Spain (21); image by permission of and owned by Enresa.

Mochovce, Slovakia

The surface disposal vaults are constructed of concrete and drums of compacted waste are grouted into concrete containers and placed in the vaults. The facility receives both LLW and short-lived or less hazardous ILW. Once filled, the vault is backfilled and capped with concrete. At closure, the entire repository will be capped with a multi-layered engineered capping system (17).

2.7.2 At-depth disposal

Although there is less direct experience of at-depth NSD facilities when compared with at-surface NSD facilities, both the at-depth vaults and at-depth silos concepts have been proven internationally with facilities in Finland and Sweden operating since the 1980s or early 1990s. At-depth NSD is an internationally proven concept for the management of LLW and short-lived or less hazardous ILW and is considered to be of high technical maturity.

SFR Forsmark, Sweden

The SFR facility (22) is located at a depth of approximately 50 metres beneath the Baltic Sea, with LLW being disposed of in ISO containers in a concrete vault and ILW in a concrete silo. The silos have a base of 1 m thick reinforced concrete over a layer of sand-bentonite. The silo has a diameter of approximately 30 m and an outer wall constructed from reinforced concrete. It is 70 m in height, with a height of 50 m available for disposal, the remaining head space will be backfilled with bentonite. After emplacement of waste packages in the shafts, concrete is used to seal void spaces, allowing disposal operations to continue. At closure, the remaining void space in the silo will be filled with concrete.



Figure 7 Disposal vaults for LLW at the SFR, Sweden (23); image by permission of and owned by SKB.

Loviisa disposal facility, Finland

The Loviisa disposal facility (22), is at a depth of approximately 110 m below ground, with the facility comprising three caverns for solid LLW and a cavern for immobilised ILW. Inside the cavern for ILW, the waste packages are emplaced in a structure made of reinforced concrete. After emplacement the waste caverns will be filled with crushed rock and concrete plugs will be installed to seal the caverns, access tunnels and shafts.



Figure 8 Disposal vaults for ILW at Loviisa, Finland (24); image by permission of and owned by Fortum.

Vrbina-Krsko, Slovenia

A silo facility in Slovenia is proposed for the disposal of LLW and short-lived or less hazardous ILW. The silos will be approximately 55 m deep with an inner diameter of 27 m and will be excavated from the surface. An initial 1.2 m thick reinforced concrete diaphragm wall will provide stability, and a second 1 m thick reinforced concrete liner will provide isolation. A temporary operations hall will be constructed over the silo. Waste containers will be placed into the silo by crane and the voids between containers backfilled with concrete. Filling of the silo will be complete when the level of waste is approximately 15 m below ground surface. A 1 m thick layer of concrete will be cast over the silo and clay will be placed over the top to ground level, which is intended to limit water ingress into the silo (22).

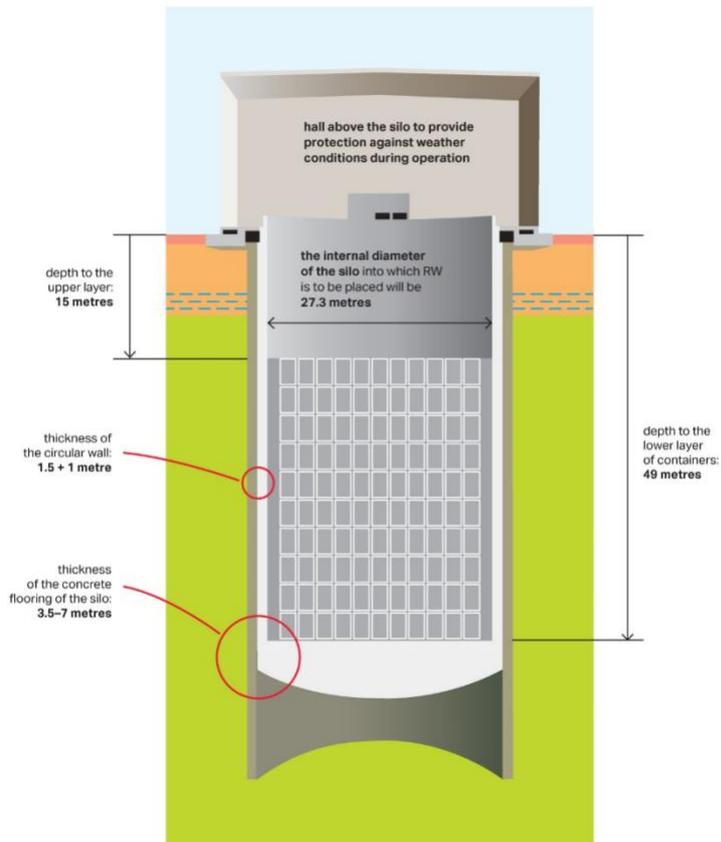


Figure 9 Vrbina repository, Slovenia, vertical cross-section (25), image by permission of and owned by ARAO.

3. Deep Borehole Disposal

3.1 Position statement

In principle, deep borehole disposal could be compatible with some of the most hazardous radioactive wastes within the Inventory for Geological Disposal but is not compatible with the full inventory. Therefore, a Geological Disposal Facility will always be required, even if deep borehole disposal of radioactive wastes were to be implemented in the future. At time of writing, deployment of deep borehole disposal technology in a nuclear environment would require further policy and legislative consideration, both of which would be a matter for government. Regulatory approaches would also have to be developed to ensure proper regulation of operational safety and security alongside long-term environmental protection. It is recognised that development of generic deep borehole concepts has progressed over the last decade. Nevertheless, substantial further development is required to increase the maturity of the approach for potential application, including consideration of operational and post closure safety. Whilst some components of the required technology are considered mature in their current implementation, e.g. drilling of boreholes within the oil and gas sector, they have neither been deployed to nuclear safety and security standards, nor fully integrated as a technological solution at the scale required for reliable disposal of solid waste packages anywhere in the world. Waste packaging designs are also of a conceptual nature. At present the technology readiness of deep borehole disposal is considered low to medium, it is yet to be licenced or implemented anywhere in the world.

3.2 Recent developments

3.2.1 Consideration by international implementers

The status of DBD technology was reviewed by EDRAM, the International Association for Environmentally Safe Disposal of Radioactive Materials, involving the implementing waste management organisations from 11 national programmes (26). This review identified that substantial work would be required to mature DBD to be an implementable alternative to GDF, including on issues of legislation, regulation, site characterisation, and safety case; including borehole sealing and recovery from maloperations. EDRAM adopted an evidenced position on DBD concepts, expressed as three key principles:

- DBD can never replace a GDF in most countries with significant inventories of radioactive waste.
- DBD may have potential as an alternative to a GDF for countries with smaller waste inventories (which can be easily converted to small diameter packages).
- DBD technology requires significant development to reach a sufficient level of maturity for nuclear waste disposal.

ERDO, the Association for Multinational Radioactive Waste Solutions, initiated a project to assess the potential of DBD (27). It concluded that DBD is a “*technologically feasible and potentially cost-efficient solution for high-level or long-lived ILW from Croatia, Slovenia, Denmark, The Netherlands, and Norway*”, but that “*DBD is a less mature concept than mined repositories*”. It was recommended that a full-scale inactive demonstration of the approach be undertaken, for which an initial estimate of €97 million was cited (28). In a study commissioned

on behalf of ERDO, Deep Isolation (a developer and implementer of directional deep borehole technology) concluded that its borehole technology could dispose of the full inventory of high heat generating waste across the ERDO countries, at a cost of €542-684 million, equivalent to around one half or two thirds of the cost of a suitable GDF (28). Nevertheless, the ERDO nations would still need to develop a suitable disposal solution for their ILW and LLW, not suited for DBD.

Norwegian Nuclear Decommissioning has evaluated a DBD concept, as an alternative to GDF, for disposal of its relatively small volume of HLW, comprising spent nuclear fuels from experimental and test reactors (29). An intermediate depth repository would still be required for ILW and LLW. It concluded that *“deep boreholes as a disposal option is demonstrably practical and feasible with currently existing technology used in other mining and civil engineering practices”*. However, this conclusion is tempered by the recognition that such existing drilling technology and waste packaging design would require “adaptation”, which would need to be addressed in the context of a particular site and geology. The crystalline basement rock prevalent in Norway was considered suitable and advantageous for implementation of DBD, it was claimed that several advantages favour the implementation of DBD over GDF technology, including: reduced potential of accessways as return pathways, easier fabrication of seals, reduced excavation, and greater operational safety.

A conceptual design was produced for a DBD disposal canister suitable for disposal of Norway’s small spent nuclear fuel inventory; the canister would not be self-shielding (30). The conceptual design uses austenitic and duplex steel for construction, acknowledging that the corrosion resistance of such steels is significantly reduced in brines with a high chloride content, as may be expected in the borehole deposition zone. Consideration has been given to potential materials and design of plugs and seals (31).

The US DOE terminated its Deep Borehole Field Test experiment in 2017, citing changes in budget priorities (32). This followed opposition by communities local to the potential test site in North Dakota, founded on concern that the communities would be required to accept the disposal of wastes in the future. Site selection for this DBD demonstration test followed a consent-based process; the failure of this process highlights the need for DBD projects to learn from international GDF siting experience in planning a realistic timeframe to build trust and confidence of potential host communities and progressing development at a pace they are comfortable with.

3.2.2 Reviews of DBD technology and implementation

Several reviews have considered the maturity and implementation of DBD technology, since 2017. A review of the potential uptake of DBD technology showed that this is likely to be highest for those nations with only a small amount of spent nuclear fuel to manage (with no reprocessing) (32). It is suggested that implementation of DBD in nations with large and complex inventories, which will anyway require a GDF, may be favoured if disposal of high heat generating waste could be deferred pending satisfactory demonstration of DBD (32). However, this would evidently propagate the hazard, risk and cost of extended surface storage and waste repackaging arrangements. A critical review of the geological and geochemical considerations of DBD, with a focus on post closure safety, challenged the broad assumptions of favourable density stratification of groundwater, low permeability of basement rocks, and reducing geochemistry, at depth (33). It was emphasised that, to address the relevant uncertainties, greater consideration should be given to the specific geological setting for deployment and site characterisation programmes in DBD implementation. A review of the potential for DBD

implementation in Germany concluded that this is a highly unlikely prospect and foreseeable only in the context of the failure to successfully site a GDF; the need for considerable research and development was highlighted (34) (35).

3.2.3 UK focussed research and development

An initial assessment of the potential suitability of Deep Isolation's directional borehole technology for disposal of elements of the UK's IGD was recently published (36). This study showed that:

- Directional borehole disposal could not replace the need for development of a GDF in the UK, since it is not suited to the full diversity of the UK's waste inventory.
- A GDF will always be required, but directional borehole disposal could conceivably be considered in the future to dispose of some of the UK's high heat generating waste (e.g. HLW glass, and spent fuel and plutonium if classified as waste)
- Further development of directional borehole technology is required to increase the maturity for potential application to the conceivable inventory, including consideration of operational and post closure safety.

The most recent consideration of DBD technology by CoRWM, concluded that in the UK context, the approach "is unlikely to be practical for a number of reasons including the size and range of wastes in the inventory, the size of packaging and the as yet unproven practicalities of drilling boreholes of sufficient diameter to appropriate depths" (37). However, CoRWM noted that, from a technical perspective, DBD may be a suitable option for disposal of certain materials, such as plutonium or highly enriched uranium, or for countries with smaller waste inventories.

DBD of the UK HLW inventory has been proposed to avoid the cost of replacing the current Vitrified Product Store, pending the availability of a GDF (38). However, the current expectation is that the operational lifetime of the Vitrified Product Store may be extended, subject to a robust and rigorous safety case, to allow export of vitrified HLW product to the GDF.

Whilst the conceptual DBD approach draws, to some extent, on commercially available technologies for implementation, these have not been adequately integrated and nuclearised to be confident of the technical maturity and readiness of the approach. The required drilling technology to achieve a suitable diameter hole to the required depth is stated to be "within or only very marginally outside" or "only slightly outside current experience" (38), and substantial cost savings were claimed against disposal of the HLW glass inventory in a GDF. Such an assessment needs to be considered with caution: any such comparison should be made against the marginal additional cost of HLW glass disposal, since a GDF will be required for the remainder of the UK Radioactive Waste Inventory, and use a comparable methodology.

3.2.4 Wider research and development

The peer reviewed literature has reported advancements in several areas required to develop the technical maturity and readiness of DBD technology:

- Coupled thermal and transport modelling of radionuclide migration (39);
- Optimisation of support and sealing systems (40) (41) (42) (43);

- Generic post closure safety assessment (44);
- Horizontal drill holes for waste emplacement (45);
- Socio-technical analysis of the acceptance of DBD implementation (46).

Several of these reports were published in a special issue of the journal “Energies”, the guest editorial expressed an opinion that there existed a “sufficient knowledge base to plan and execute field-based demonstration projects that address key elements of borehole disposal, including deep drilling of wide-diameter holes, waste emplacement testing and seal emplacement and performance monitoring” (47). Nevertheless, it is considered that there remain important uncertainties in these sub-system components and critical gaps in the demonstration of their integration at pilot scale, within the constraints of a nuclear operational environment, as highlighted below.

3.3 Basis of Position

3.3.1 Basis of alternative concept

Disposal of radioactive wastes in deep boreholes was first proposed as an alternative to geological disposal in the 1980s (48) (49).⁴ The concept is to deposit radioactive waste packages at a depth of up to several kilometres, in an environment where groundwater is effectively stagnant due to density stratification and isolated from near-surface hydraulic gradients. Two distinctive approaches to implementation have emerged:

- Disposal in vertical deep boreholes, up to 5 km deep, with disposal of waste packages in a deposition zone below 2 km.
- Disposal in deviated horizontal deep boreholes, up to 2 km deep, with disposal of waste packages in a deviated horizontal section up to 1.5 km long.

These concepts require a suitable host rock for implementation, in the case of vertical deep boreholes, this is generally a crystalline basement rock.

3.3.2 Compatibility with current UK government and devolved administration policies

UK Government, Welsh Government and Northern Ireland Executive policy is to dispose of the most hazardous radioactive wastes in a Geological Disposal Facility, comprised of a series of tunnels and vaults, at a depth of 200 – 1000 m, with safety assured using a system of multiple barriers (engineered and geological) (3). No credible alternative to geological disposal was considered feasible when the policy was adopted following the original CoRWM review in 2006, which was based on an extensive review of evidence to identify solutions that could be deployed using currently available technology, could carry public confidence and be legal within the UK. That position had not changed in any of the subsequent policy statements on GDF implementation in 2008, 2014, 2018 or the UK policy framework for managing radioactive substances and nuclear decommissioning (3). As noted earlier, any further policy or legislative

⁴ Here, “deep” is taken to exceed the feasible depth of construction of a Geological Disposal Facility, of the order of 1 km, for the application to disposal of long lived higher activity radioactive wastes.



considerations required to enable DBD would be a matter for government and an appropriate licensing and permitting basis would need to be created to enable implementation.

3.3.3 Compatibility with UK IGD

A primary constraint on compatibility of DBD concepts with the IGD (1), is the diameter of the borehole relative to that of the potential waste package, allowing for any required overpack, clearance tolerance, and sealing material. Vertical borehole diameters with a diameter of 0.5m to 3km depth are considered routinely feasible, whereas a diameter of 0.75m to 4km depth, would require significant investment in R&D to achieve routinely (50). Deviated horizontal boreholes with a diameter of 0.4m are considered routinely feasible up to 3km depth, with confidence in 0.6m diameter (28).

In principle, both vertical and deviated horizontal boreholes could be dimensionally compatible with disposal of existing and future HLW glass packages, with a canister diameter of 0.5m, assuming an adequate deposition container. Both concepts are also, in principle, compatible with the disposal of spent nuclear fuel, for which an unconsolidated Pressurised Water Reactor fuel assembly is likely to be limiting, measuring just over 0.2m x 0.2m on a cross section. Disposal of immobilised separated plutonium would also be feasible, in principle, if produced in packages commensurate with achievable borehole diameter. It would need to be demonstrated that the deposition container could be safely transported, handled, and emplaced. The dimensions of ILW packages in the IGD are not compatible with either vertical or deviated horizontal disposal concepts (36).

It is clear that DBD concepts could only provide a solution, in principle, for the high heat generating waste component of the UK IGD, in agreement with a study by Deep Isolation (36). These wastes comprise approximately 10% of packaged volume and 96% of activity (at 2200) of the IGD (1). A GDF will always be required to manage the substantial volume of the ILW inventory.

3.3.4 Technical maturity

Implementation of DBD requires the integration of component technologies which all require at least some development and proven application within the constraints of a nuclear safety environment.

The basic drilling technology required to develop a deep borehole is well established in the oil and gas sector (36) (50) (51) (52), but, crucially, routine operation in a nuclear safety environment has not been demonstrated. Drilling of boreholes with a diameter potentially compatible with high heat generating waste packages, to relevant depths, has been achieved, but such applications are not yet routine (52).

Some consideration has been given to a high-level concept of operations for DBD (53), but considerable maturation is required. Waste package receipt, storage and handling operations on a surface site are likely to be similar to those required for a GDF, which are already proven. Design solutions for emplacing waste packages in a deep borehole (and retrieving trapped or misplaced waste packages) are currently generic in nature. Extensive development will be required to mature these designs for demonstration to nuclear safety standards, including fault schedules and hazard management strategies.

The potential for waste packages to become trapped in the borehole during emplacement or retrieval operations is of concern, particularly the occurrence in a zone of groundwater flow in

the near-surface section of a borehole. To reduce the likelihood of such an event a smooth borehole casing is required and the use of callipers or suitable sensors in the deployment system to detect borehole deformation or obstruction. It is suggested that specialised “fishing” tools, developed to recover or move material in oil and gas wells, could be applied to dislodge and recover trapped waste packages (54). However, such a method would risk (further) damage to the trapped waste package, complicating recovery. Deposition and recovery of a simulated waste package has been demonstrated for horizontal DBD (55), but not under relevant accident or nuclear safety constraints.

In the event that a waste package was irretrievably trapped above the waste deposition zone, it has been suggested that the borehole could be sealed and abandoned (53). This would require demonstration and safety case assessment to show that the system can be managed to an acceptable end point for long term safety. Recovery of the system or its management to a safe closure state also needs to be considered for scenarios involving dropped waste packages and partial borehole collapse, which could lead to damage of waste packages, release of radioactivity and potentially criticality. All of these scenarios represent additional scope for loss of control of highly active waste packages beyond the realistic scenarios considered in relation to existing GDF concepts and these will all need to be addressed with the nuclear and environmental safety regulators before a DBD concept can be considered feasible.

In the UK, policy for radioactive waste disposal advocates the multi-barrier concept to assure long-term safety (3). IAEA guidance also prescribes a requirement for complete physical containment of radioactive waste during the thermal period after disposal, to exclude the role of heat-driven transport of radionuclides (56). Implementation in the UK context will therefore require demonstration of container performance and an engineered barrier material, to fill the space between waste packages and the confining borehole and minimise groundwater access and radionuclide migration. This material may also serve as a support matrix for waste packages under load stress in the disposal zone. Waste packaging will need to be designed to fit the constraint of borehole diameter and provide sufficient structural integrity to prevent damage during handling and emplacement, and under load stress.

Options proposed for engineered barrier materials include a high-density support matrix and cement grouts (57). Lead-tin alloy is proposed as a high-density support matrix, deployed in granular form; radiogenic heating from waste packages leads to melting and eventually solidification to form a seal (57). The use of such a hazardous substance is likely to prove challenging to justify. Cementitious grouts have been developed as derivatives of oil well sealing cements, with adequate workability and curing behaviour to enable down hole deployment, in principle (40) (41) (42) (43). Nevertheless, evidence to demonstrate the long-term efficacy of container materials and engineered barriers, in the comparatively hostile environment of a deep borehole (elevated salinity, temperature and pressure), is presently immature. Methods to verify and validate successful emplacement of engineered barriers, sufficiently filling open voidage, also need to be developed and demonstrated to nuclear standards.

It will be necessary to evidence effective sealing of the borehole, at least for the thermal period after disposal, consistent with the principles of achieving isolation and containment, and to avoid a direct pathway for radionuclide migration from the deposition zone to the surface. In the upper section of the borehole, conventional sealing approaches may be applicable, for example, cement, bentonite, and swell packer seals. Evidence for the long-term adequacy of such seals at greater depth requires research. An alternative approach to borehole sealing in crystalline

rock involves melting crushed material and the adjacent rock body, followed by re-crystallisation, either by electrical or chemical means (58), or a ceramic plug formed *in situ* by thermite reaction (44). These approaches are currently at the stage of laboratory scale proof of concept and require considerable maturation. Exhaustive testing of engineered barriers and seals, over long timescales, in a demonstration borehole are likely to be required to demonstrate adequate performance before construction could be licensed.

3.3.5 Post closure safety considerations

In post closure safety assessment of GDF concepts, long term passive safety is primarily assured by the isolation afforded by the geological barrier, imposing long return time and dilution of radionuclides, combined with effectively stagnant groundwater. Generic total system models show that the peak risk arising from DBD concepts may be within guidance levels, under conservative assumptions (44) (59). Nevertheless, supporting safety narratives require development in consideration of the containment afforded by the engineered barriers during the thermal phase after disposal and evolution under coupled thermo-hydro-mechanical-chemical processes, and the performance of plugs and seals. This is arguably of greater significance in the context of DBD concepts because long term *in situ* experiments are likely to prove difficult, due to the relatively hostile environment, and there are unlikely to be relevant natural analogues to validate the predictions of mechanistic models. Existing thermochemical and mechanistic data sets, which were developed for conditions relevant to geological disposal, are unlikely to be transferrable and will need to be established for DBD environments. Management of criticality in waste package emplacement, retrieval and post closure safety has received very little consideration in DBD proposals. The containment assured by the engineered barriers in generic geological disposal system designs, over 100,000 years, underpins criticality safety. Since deep boreholes place less reliance on such barriers for containment, in favour of the isolation afforded by the geological barrier, it may be challenging to demonstrate the criticality safety for disposal of spent fuel and fissile material.

3.3.6 Timescale for implementation

It has been argued that construction, operation, and sealing of a deep borehole could be achieved within a few years, and could be deployed more rapidly than a GDF (38) (52). These arguments are generally concerned with reduced requirements, and hence timescale, for adequate site characterisation, coupled with postulated greater societal acceptance due to siting flexibility.

Although access to the waste disposal zone could conceivably be achieved more quickly for deep boreholes, the surface site and transport infrastructure requirements are likely to be comparable to those of a GDF. It is plausible that reliance on essentially static groundwaters and low permeability basement rocks, at depth, with a small deposition volume, may enable site characterisation to be achieved more rapidly. However, comparatively little is known about the deep subsurface geology of the UK, which was not considered in the National Geological Screening exercise. Reliance on essentially a single (geological) barrier and the presence of a potential direct return pathway to the biosphere, will increase characterisation requirements to adequately reduce uncertainties.

The failure of the US Deep Borehole Field Test programme, in siting a demonstration deep borehole, due to public opposition, does not evidence greater public acceptance, of this technology (60). Strong international consensus for implementation is also yet to develop, in contrast to that established in support of geological disposal, which formed over several decades. There is no compelling evidence at the present time to suggest earlier implementation



of a deep borehole option could be achieved, given the need to navigate required policy, regulatory and development consent processes prior to siting and construction.

3.3.7 Cost

Since DBD is compatible in principle with only a fraction of the UK IGD, implementation could only complement rather than substitute the need for a GDF. Consequently, the cost of DBD must be evaluated against the marginal cost of disposing of these radioactive wastes in a GDF. At present cost estimates are too immature to draw meaningful conclusions on the comparative cost effectiveness or otherwise of DBD concepts. The number of boreholes required to dispose of compatible waste inventory depends sensitively on the usable volume of the deposition zone and waste packaging assumptions, which are highly uncertain. Realistic cost data are required for construction, operation, and closure of deep boreholes, within constraints of environmental permitting and nuclear licencing.

4. Waste Treatment Techniques

4.1 Position Statement

Alternative waste treatment techniques for ILW could have a positive impact on the Inventory for Geological Disposal, by reducing the hazard, volume, cost and timescale of treatment and disposal of some waste streams. There is growing evidence that thermal treatment, if demonstrated to be viable, could radically reduce the volume and number of waste packages consigned to a GDF, and enhance passive safety through destruction of organic material and oxidation of reactive metals. The present maturity of these technologies is assessed to be medium to high, based on successful laboratory and full scale demonstration with simulant wastes relevant to the UK context. The available evidence, from short term experiments, is that the conceptual products of thermal treatment present a high chemical durability and will undergo only very slow dissolution in a GDF. However, there is a need to develop understanding and evidence for the long term stability of the products of thermal treatment in a cementitious backfill. It is recognised that there is an opportunity to develop a more optimal concept for disposal of thermal treatment products, which do not necessarily require conservation of alkaline conditions by a cementitious backfill to assure chemical containment; a low pH cement or bentonite backfill could be more appropriate.

Geopolymer and magnesium phosphate cement encapsulants are, respectively, considered to be of medium-high and low-medium technical maturity. These enhanced encapsulants are unlikely to deliver benefits of reduction in the volume and number of waste packages, but improved compatibility with the waste material may contribute to post closure safety by reducing complexant, non-aqueous phase and gas phase mediated transport of radionuclides, relative to the current baseline. Considerable further investigation is required to understand the long term stability, hydration and alteration processes of enhanced encapsulants, the mechanism and rate of encapsulated waste corrosion and degradation processes, and interaction with backfill materials.

Going forward, the NDA group will continue to consider the opportunity of thermal treatment and enhanced encapsulant technologies, and development of GDF vault and backfill concepts, that enable the full lifecycle waste management benefits of these technologies to be captured.

4.2 Recent developments

4.2.1 Thermal treatment technology

Higher Activity Waste Thermal Treatment Programme

Sellafield Ltd have undertaken the Higher Activity Waste Thermal Treatment (HAWTT) programme, to evaluate and mature this technology for potential application in a UK context. The Programme was structured in three tranches:

- Tranche 1: Plutonium Contaminated Material (PCM), treatment by plasma technology
- Tranche 2: Pumpable ILW, treatment by Joule Heated Ceramic Melter
- Tranche 3: Solid Beta/Gamma Wastes, treatment technology to be determined



Tranche 1 delivered desk based, laboratory and mid-scale development work to assess the potential of thermal treatment of PCM. Desk based studies reviewed the readiness of plasma system technology, to identify the system components requiring maturation for deployment. Crucible scale studies assessed the suitability of glass formulations for treatment of a PCM simulant. Mid-scale trials were undertaken to understand the interaction of plasma with glass and waste simulant materials and potential particulate carry over to the off gas system. NWS has prepared an Expert View to Sellafield Ltd, highlighting the uncertainties and risks in conceptual waste package acceptance for geological disposal, to be addressed in further research.

Tranche 2 delivered laboratory and bench scale tests of Magnox sludge vitrification, focusing on crucible scale glass formulation studies, with bench scale melter validation. The melter feed arrangement has also been considered.

Joule Heated In Container Melter Technology

The National Nuclear Laboratory (NNL) and Kurion Veolia (now Veolia Nuclear Solutions) installed a demonstration GeoMelt In-Container Vitrification (ICV) system at the NNL Central Laboratory Active Rig Hall, which completed commissioning in 2016 (61). In outline, the melter comprises a pre-cast 200 L refractory crucible, to contain the melt, within a steel container, with a feed system that enables waste addition during processing. The system has a comprehensive off gas management systems with pre-treatment, cooling, wet scrubbing, final conditioning and discharge stages. Three successful demonstration melts were undertaken during commissioning, involving inactive soil with / without Cs-137 tracer, and active soil from UK nuclear sites co-processed with Magnox sludge simulant. Vitrified products were produced with Cs retention estimated at 99%.

The demonstration system was subsequently used for a series of trials on generic UK waste types to understand the potential for thermal treatment, with a focus on the impact of the feed and configuration of the melter on the development of the vitrified product and retention of key radionuclides (62). The trials were carried out with / without the addition of radioactive tracers to simulate wastes representing: Magnox sludge (with and without skip containment), clinoptilolite ion exchange material, mixed decommissioning wastes, cement encapsulated uranium metal in a steel container, and polyvinyl chloride (PVC) (analogous to soft PCM waste). Vitrified products were obtained, with a separate metal phase or remnant metal items depending on the waste feed (63). The PVC feed required shredding to afford a greater surface area for interaction with the melt. Acknowledging the trials were not optimised, higher waste loadings and volume reduction factors were obtained for waste streams with suitable glass forming chemistry (e.g. clinoptilolite: 75wt.% waste loading, 65% volume reduction), compared to waste streams without glass formers (e.g. Magnox sludge and skip: 10wt.% waste loading, 20% volume reduction); volume reduction was assessed against the unconditioned waste feed.

More recent trials demonstrated successful thermal treatment of simulants representing failed cement waste packages and a mixture of Magnox sludge and clinoptilolite (64), and satisfactory management of process issues during the operational cycle. This study estimated the technology readiness level of GeoMelt technology to be TRL 3-6. In these sets of trials it was not possible to reliably estimate radionuclide retention rates, most likely due to the challenge of adequately sampling heterogeneous products (64) (63).

Published characterisation data concerning the illustrative, but not optimised, products arising from the aforementioned trials, was limited to visual observation, and partial chemical and



radiometric analysis of the vitrified material, with no published data on chemical durability. However, a recent study performed a comprehensive characterisation of unoptimised vitrified Sellafield Pile Fuel Cladding Silo and clinoptilolite / Sellafield Site Ion Exchange Effluent Plant (SIXEP) sand waste simulants (65), produced by a GeoMelt ICV system in 2009 (66). Surrogate radionuclides were determined to partition overwhelmingly into the glass phase, which was shown to be highly resistant to dissolution. Indeed, although very different in composition, the elemental releases from the two vitrified products were an order of magnitude lower than UK HLW wasteforms under comparable accelerated dissolution test conditions. This study highlights the opportunity and value in utilising the illustrative products from such trials, to provide evidence to support later disposability assessments.

A recent strategic study considered the applicability of GeoMelt ICV technology to problematic UK LLW, with holistic consideration of the constraint of site licencing and permitting, transport, and waste acceptance criteria (67). A parametric tool was developed to support decision making for the implementation and siting of such technology for thermal treatment of LLW. The key finding arising from this work was the need to consider discrete item constraints to open up the potential for NSD of vitrified LLW products.

A GeoMelt® ICV™ system with a 10 ton melter capacity entered operation in 2018, at the Perma-Fix Northwest facility, located in Richland, WA, USA (68). The system was developed to treat mixed reactive metal wastes, with low level radioactivity, produced from the historic fast reactor programme at the Idaho National Laboratory site. These wastes incorporate trace or bulk quantities of sodium or sodium-potassium alloy, used as liquid metal coolant in fast reactors. The reactive metal component is passively oxidised and incorporated into a durable vitrified product, together with other waste constituents, during the thermal treatment process. To 2020, this plant had successfully treated 934 drums of waste to produce 16 vitrified monoliths that were accepted for disposal at the Nevada National Security Site. This advancement is of particular interest in a UK context, given the historic fast reactor programme and associated reactive metal waste streams.

In 2020, a preconceptual design was reported for a GeoMelt® ICV™ system with a 10 ton melter capacity to treat decommissioning wastes at the Fukushima Daiichi sites (69), with a throughput of 93 ICV containers per year. The system would immobilise ion exchange and sludge wastes arising from treatment of contaminated water from the Fukushima Daiichi reactors. This advancement is of significance in a UK context given the existence of waste streams of a similar nature.

The development of these large scale GeoMelt systems was predicated on a phased campaign of experiments and trials, progressing from laboratory scale investigation of formulation and melting behaviour, through bench scale ICV, to engineering scale ICV trials (68) (69) (70) (71) (72). Importantly, this provided data and evidence to understand the influence of the scale of the melter system on waste incorporation and passivation, radionuclide retention and off gas behaviour, product disposability, and other issues, as a function of feed composition and process parameters. These studies provide an exemplar of an effective phased approach to development and successful implementation of thermal treatment technologies in the UK.

Joule Heated Ceramic Melter Technology

Vitrification of Magnox sludge waste from the SIXEP was further demonstrated using a small scale Joule Heated Ceramic Melter (73). This study used an inactive magnesium hydroxide simulant, incorporating a full spectrum of radionuclide surrogates, and magnesium metal as a

surrogate for metallic Magnox fuel cladding. It was demonstrated that the reactive magnesium metal could be passively oxidised and incorporated into the glass melts, in both laboratory and small scale melter experiments, addressing a key uncertainty for application of this technology. Tailored glass formulations were developed for this purpose based on a sodium aluminoborosilicate system, achieving a waste loading of 35 wt.%, acceptable processing properties, and moderate leach resistance. Waste loading was reported to be limited by crystallisation of forsterite, Mg_2SiO_4 . It was noted that hydrogen was evolved as a result of the reaction between magnesium metal and the alkaline sludge feed during processing. Particulate carry over in waste feeds containing reactive magnesium metal was greater than, in feeds without a metal component, as a result of more vigorous reaction in the cold cap.

Hot Isostatic Pressing Technology

Laboratory scale research and engineering scale trials have advanced the maturity of Hot Isostatic Pressing (HIP), demonstrating the opportunity and compatibility with treatment of radioactive wastes relevant to the UK Radioactive Waste Inventory. An active bench scale HIP facility is operated at The University of Sheffield, with the capability to process radioactive simulants using a patented Active Furnace Isolation Chamber (74) (75). An inactive engineering scale HIP facility is operated at the NNL (76). These UK facilities were established primarily to support the NDA plutonium disposition programme. Considerable experience has been gained in the formulation, operation, and scale up of HIP products and processes, with significant investment in active trials (77) (78) (79) (80). Key advancements include:

- Design of effective pre-treatment and bake out steps, tailored to feed characteristics.
- Formulation design with active and inactive simulants, and plutonium validation.
- System engineering, integration and safety case development.

This work has greatly strengthened UK knowledge, understanding and experience in the application of HIP technology for radioactive waste treatment.

Several investigations have considered the application of HIP technology to the treatment of Magnox sludge and inorganic ion exchange materials in the UK Radioactive Waste Inventory, either as single or blended waste streams. HIP was demonstrated, at bench scale, to be effective in treatment of spent zeolite ion exchange materials to produce highly durable glass-ceramic wasteforms, with control over radionuclide partitioning between glass and ceramic phases in a wide processing envelope (81) (82) (83) (84) (85). Treatment to achieve 100 wt.% waste loading may be possible, although a minor addition of fluxing additives, to reduce the liquidus temperature, may be preferable to tailor the product microstructure and processing envelope. Volume reduction factors estimated on partially optimised bench scale wasteforms were reported in the range 40 – 80%, relative to the feed material. For spent ion exchange materials, the output thermal power of the waste package is a potential consideration for storage and disposal, given the high waste loading and volume reduction factor, and typically short lived nature of the radionuclides. Consideration of a pessimistic case for Cs-exchanged clinoptilolite, estimated the thermal output of a 3m³ box waste package to be within limits specified for ILW transport and disposal (81).

HIP technology was further demonstrated, at bench scale, to be effective for the treatment of simulant Magnox sludge, to produce a (primarily) ceramic waste form, or a glass-ceramic wasteform if co-processed with simulant clinoptilolite ion exchange material (81) (86). A waste



loading of at least 60 wt.% Magnox sludge was shown to be achievable, with volume reduction factors of 75-80%, and a wide process envelope. This approach was later proven at engineering scale, with successful HIPing of 0.4 L, 8 L and 102 L canisters, utilising a formulation developed at bench scale, and achieving a volume reduction factor of 75% for the 102 L product (87).

A full scale plant utilising HIP for treatment of ILW, arising from medical isotope production, is in the final stage of commissioning in Australia, designed and constructed by ANSTO (the Australian Nuclear Science and Technology Organisation) (88). This will be a world first facility to implement HIP technology at industrial scale for radioactive waste processing. The plant is fully automated and operated in a shielded enclosure, liquid ILW is combined with formulation additives, prior to introduction to a rotary calciner; the calcined feed is packed into a stainless steel can, that is sealed, evacuated and welded closed, prior to the HIP cycle which produces a glass-ceramic product. The plant is designed for a throughput of around 200 packages per year, at 30 L scale, achieving a volume reduction factor of approximately 60%.

ANSTO emphasises the importance of a staged development plan, as instrumental to successful technology maturation, proceeding from laboratory scale, through bench scale, to engineering scale demonstration (89). A substantial programme of early stage work to develop, test, improve, and optimise waste form formulation and performance, and an inactive engineering scale demonstrator for technology integration and nuclearization, were identified as critical to the successful development of the full scale plant.

Plasma Melter Technology

A full scale plasma thermal treatment facility entered operation in 2018, at the Kozloduy Nuclear Power Plant, Bulgaria (90). The facility was designed and built as a joint venture between Iberdrola Ingeniería y Construcción and Belgoprocess (91). The facility consists of a tilting, refractory lined, plasma furnace operating with a 500 kW non-transferable torch as the heat source. The system has a design capacity to process 250 tonnes of waste, operating over 40 weeks, per year.

The solid waste feed comprises bags of organic waste, 200 L drums containing organic and metallic waste, and supercompacted 200 L drums containing mixed wastes of concrete and organic material. Typical activity of the waste feed is around 0.5 MBq / kg. The waste feed is passed through a shredder and passed into the melter by a transfer tube, the melt temperature is typically 1400°C. When the melter has accumulated 200 L of slag it is discharged into a mould, leaving around 50 L in the melter to initiate the next melt. The system has an integrated off gas handling system and afterburner chamber.

The plasma melter system design was approved in 2012, with component manufacture and factory acceptance testing completed in 2013. Construction of the facility started in 2015, with integration testing of the complete facility in 2017 (90). The facility entered operation in 2018, processing the full spectrum of anticipated waste types. In the first period of operation, it was reported that 6.4 tons of waste was successfully processed to afford 170 L of slag product. The volume reduction factor was around 90%, relative to the feed.

More recently, the thermal treatment of simulant organic ion exchange resins and bitumen wastes was successfully demonstrated using a plasma test facility, comprising a tilting, refractory lined furnace, with a 1.2 MW non-transferred arc plasma torch (90) (93). This produced a slag product with a volume reduction factor of around 85%.

Laboratory formulation studies

Thermal treatment of Magnox sludge from the First Generation Magnox Storage Ponds was demonstrated as a proof of concept at laboratory scale, with a view to implementation in Joule heated melter systems (94) (95).

Mixed alkali – alkaline earth borosilicate glasses were designed for the immobilisation of a simple simulant Magnox sludge waste (magnesium metal and magnesium hydroxide) (95). Homogeneous glass wastefoms were produced at 1200°C, with up to 30wt.% of Magnox sludge, limited by crystallisation of forsterite. Retention of up to 90% of the spiked Cs₂O content was reported and the products demonstrated dissolution behaviour typical of borosilicate glasses.

Vitrification of more complex bounding sludge waste simulants was also recently reported (94). These simulants represented: a corroded waste stream, comprising principally magnesium hydroxide sludge, with minor Magnox fuel cladding, uranium fuel, and uranium oxide; and a metallic waste stream comprising mostly uranium fuel and Magnox fuel cladding, with minor magnesium hydroxide sludge. To accommodate the magnesium content of the waste, magnesium aluminosilicate and magnesium borosilicate glasses were designed, from knowledge of the pseudo ternary phase diagrams, with suitably low melting point.

Passive oxidation of the metal components was observed during the vitrification process, which produced glassy products at melting temperatures of 1200°C for the borosilicate system and 1500°C for the aluminosilicate system. The heterogeneity of the products was comparable to that of wastefoms produced at pilot scale. Dissolution experiments in deionised water evidenced lower dissolution rates for the borosilicate products, broadly comparable with simulant UK HLW glasses tested under comparable conditions. The study considered implementation of the approach using Joule heated ICV to be plausible, and estimated a volume reduction of 60% could be achieved, against the baseline of cement encapsulation of the waste streams.

4.2.2 Enhanced encapsulant technology

Cement encapsulation, using Ordinary Portland Cement (OPC) and Supplementary Cementitious Materials (SCM), is the baseline technology for treatment of compatible legacy and operational ILW and LLW in the UK. However, there are legacy wastes that are not compatible with this approach or for which a more optimal encapsulation technology is desirable. Geopolymer and Magnesium Potassium Phosphate Cements have been the focus for development as enhanced encapsulants and selected recent progress, relevant to the UK context, is reviewed here.

Problematic Waste Integrated Project Team

The Problematic Waste – Integrated Project Team (IPT), has taken a more coordinated approach to the management of problematic wastes across the NDA estate. Many sites have small volumes of radioactive wastes that are considered problematic, due to their age, chemical, physical, or radiological properties, or as a consequence of their storage arrangements. The development of enhanced encapsulants is one approach to realising this aim, enabling wastes to be treated that are not compatible with baseline OPC encapsulation or achieving desirable improvement in package safety over the current baseline. The IPT has progressed this, by:



- Collating an inventory of problematic wastes to improve understanding and identify opportunities;
- Delivering a programme of R&D to identify and implement waste management solutions for problematic waste;
- Acting as a focal point for the sharing and exchange of knowledge between waste producers, and other stakeholders, on problematic waste management issues.

Geopolymers

Sellafield Ltd sponsored a programme of research to investigate the potential of geopolymers for encapsulation of ILW (96). The potential advantages of geopolymer encapsulation were considered to be:

- Low viscosity, enabling higher incorporation rate of sludge and flood grouting;
- Reactive incorporation of silicate wastes, yielding a high waste loading;
- Chemical similarity to OPC, and expected compatibility with disposal systems;
- Flexibility in formulation, affording a wide processing envelope.

The initial development programme was designed to assess the potential for geopolymer encapsulation of inactive waste simulants of Magnox sludge, and clinoptilolite and synthetic AW500 zeolite ion exchange materials.

Geopolymer formulations based on potassium hydroxide activated potassium silicate or metakaolin were assessed for encapsulation of Magnox sludge and clinoptilolite, targeting an incorporation rate of around 70 vol.%. The proprietary MIDAR[®] geopolymer product, marketed by Lucideon, was assessed for encapsulation of AW500, targeting an incorporation rate of 70 wt.%. Satisfactory products were produced at laboratory scale by a process intended to simulate in drum mixing, the waste incorporation rates were higher than achievable with conventional OPC formulations. The compressive strength and gas permeability of geopolymer products were reported to be comparable to those of OPC based counterparts. The rheology of geopolymer formulations was highlighted as being advantageous for encapsulation of AW500 zeolite in sealed containers, with over standing water.

Jacobs reported an assessment of its proprietary SIAL[®] geopolymer technology for potential application to encapsulation of radioactive wastes in the UK (97), drawing on more than 20 years of operational experience developed in Czechia and Slovakia (98). This central European experience has applied geopolymer technology to the successful encapsulation of:

- More than 3000 tons of radioactive sludge, resins, borate salt and contaminated organic waste from Bohunice and Mochovce, Slovakia; and
- Approximately 300 m³ of spent ion exchange resins and sludge from tanks at the Dukovany Nuclear Power Plant, Czechia.

This established encapsulation process involves in-drum mixing the waste feed with the alkali activated SIAL[®] matrix, a proprietary mixture of aluminosilicate and other inorganic materials;



the process operates on a 200 L drum scale. The encapsulated products meet the waste acceptance criteria for subsequent NSD.

The potential for SIAL® application to encapsulation of UK radioactive wastes was investigated in laboratory scale trials with inactive simulant wastes, it was demonstrated that:

- Solidification of SIAL® geopolymer was only modestly retarded by high zinc concentration, in contrast to the prohibitive retardation of OPC systems, which is of relevance to waste arising on the Dounreay site.
- SIAL® geopolymer was effective in encapsulation of Magnox sludge, arising on the Sellafield site. Satisfactory products were obtained with a sludge incorporation rate of 30 wt.%, with improved workability over OPC systems.
- Mixed solid decommissioning wastes were effectively encapsulated by SIAL® geopolymer on the scale of a 200 L drum, the low viscosity of the geopolymer system being advantageous for infilling of voids, without vibration or agitation.

Effective geopolymer encapsulation of a simulant hydrocarbon oil was demonstrated in a two step process, at laboratory scale, using a metakaolin based formulation, in the context of radioactive waste management in the UK and France (99) (100). The approach exploits formation of an oil emulsion *in situ*, which is encapsulated by the geopolymer matrix as micron sized droplets, achieving an incorporation rate of up to 30 vol.%. Partial replacement of the metakaolin substrate by Ground Blast-Furnace Slag (GBFS) or Fly Ash (FA) was investigated, with the aim of reducing the overall environment and CO₂ footprint of metakaolin based geopolymer formulations. It was shown that such modified formulations afforded satisfactory products at laboratory scale, and enabled an increase in the oil incorporation rate to 60 vol.%.

Laboratory scale investigation showed metakaolin based geopolymers to be compatible with encapsulation of ion exchange resins, achieving superior waste incorporation rate and Sr retention (based on stable isotope studies) (101). Mechanistic studies established the potential for Sr, Co, and Cs to be chemically incorporated within the aluminosilicate gel of the geopolymer binder, exchanging for alkali cations, in contrast to retention by surface sorption in counterpart OPC systems (102) (103), leading to comparatively lower leaching.

Magnesium phosphate cements

Research and development of Magnesium Potassium Phosphate Cements (MKPCs) has focused on understanding the fundamental mechanisms of binder formation and setting, the role of SCMs, innovation of formulations appropriate for the encapsulation of reactive metals, and conceptual fire performance. These MKPC cements use MgKPO₄.6H₂O, struvite-K, as a binder phase.

MKPC composites with up to 50 wt.% replacement by FA or GBFS were successfully developed to reduce heat evolution, water demand and cost, in the context of application to radioactive waste immobilisation (104). It was demonstrated that minor reaction of FA and GBFS with the MKPC matrix, contributed to strength development of the composites, however, the primary role of the FA and GBFS was to act as a moderately inert diluent.

MKPC formulations with GBFS or FA were demonstrated to show superior performance for encapsulation of uranium, the corrosion rate, measured by hydrogen production, was determined to be 50% lower than a conventional 3:1 GBFS/OPC system used for encapsulation

of UK ILW (105). The lower corrosion rate was attributed to the lower water / solids ratio (w/s) of around 0.24 w/s in the MKPC systems, compared to a typical 0.35 w/s ratio used in comparable OPC composite, and incorporation of water within the crystal structure of the struvite binder phase. This investigation also applied time resolved synchrotron X-ray diffraction, to demonstrate delayed crystallisation of the struvite-K binder phase in the presence of GBFS, relative to FA at the same w/s ratio. This offers further potential to optimise workability and setting behaviour in ternary formulations. More recently, MKPC based formulations were shown to have potential for encapsulation of aluminium and aluminium-magnesium alloys, with a rate of hydrogen production an order of magnitude lower than OPC systems (106).

MKPCs typically require the use of boric acid to retard setting and maintain sufficient workability of the paste during encapsulation. A mechanism of this retarding reaction has been proposed (107), which involves precipitation of an amorphous borate compound with Mg and/or K during early hydration, and also stabilisation of Mg in solution by complexation, or as a charge compensating species for polyborate ions, all of which delay formation of struvite-K.

The high temperature transformations of MKPC composite formulations were investigated up to 1200°C, to assess performance relevant to a fire accident scenario (108), (109), (110). Irreversible dehydration of the struvite-K binder phase was found to yield a previously undocumented amorphous polymorph of MgKPO₄, in the range 100 – 350°C, above which the known α-MgKPO₄ polymorph crystallised. Solid state reaction between the dehydrated MgKPO₄ binder and the FA or GBFS matrix was initiated at 1000°C, leading to the formation of potassium aluminosilicate phases leucite and kalsilite, together with a glassy phase. Although the test specimens evidenced a considerable change in phase assemblage and microstructure, they retained structural integrity, with no evidence of cracking or spalling. It was concluded that the high temperature behaviour of MKPC composites did not present any prohibitive concern for performance in a fire accident scenario, in terms of challenging reactivity or loss of structural integrity, acknowledging the small scale nature of the experiments and recognising the need for further investigation at waste package scale.

The radiolysis behaviour of MKPC cements was investigated under external gamma and internal alpha irradiation (from incorporation of mixed uranium plutonium oxide) (111). In comparison to OPC based materials, MKPC formulations demonstrated lower radiolytic gas production. It was reported that the radiolytic yield of hydrogen was reduced by a factor of 2-3, provided that the water / solids ratio is sufficiently low, such that the mix water is effectively fully consumed to form the struvite-K binder phase. MKPC specimens exhibited excellent resistance to gamma irradiation up to 10 MGy, with no change in dimensions, compressive strength or mineralogy. It was suggested that radiolysis performance may be further improved by radical scavengers or hydrogen getters.

4.3 Basis of position

The NDA's 2019 Radioactive Waste Strategy (11), established that processing waste should occur as early as practicable, achieving a form that is suitable for long-term storage and/or disposal. However, this strategy still allows for the option to pursue alternative approaches, where there are strong drivers to do so (such as safety or maturity of technology).

4.3.1 Basis of alternative concepts

Alternative waste treatment technologies may enable optimisation of the radioactive waste management lifecycle, by reducing the volume, hazard or risk of waste consigned for disposal

relative to the current baseline approach, or for which no baseline exists. The treatment options considered in this review have been limited to options that are not currently employed for managing the most hazardous radioactive waste in the UK:

- Thermal treatment technology
- Enhanced encapsulation technology

In the context of this report, containerisation, polymer encapsulation and decontamination technology are considered part of the current baseline strategy for the most hazardous radioactive waste (11), and are therefore not considered further.

4.3.2 Basis of thermal treatment concepts

This review considered thermal treatment technology being actively developed for application to the most hazardous radioactive wastes in the UK, or relevant waste streams of international programmes. These technologies all use the application of heat to produce a solid product, through reaction of waste material with chemical additives, and evolution of volatile components to an off gas system. The solid product is typically a glass, slag, glass-ceramic, ceramic, and/or metal material; here, the terms glass-ceramic or slag used to mean, respectively, a material in which a desired crystalline phase assemblage is either purposefully or not purposefully crystallised on cooling. The potential benefits of thermal treatment technology for storage and disposal, include:

- Reduction or elimination of waste chemical reactivity: through (partial) oxidation of reactive metals and other materials, to produce a product with greater passive safety.
- Destruction of organic materials: eliminating the potential for soluble degradation products with potential to increase radionuclide solubility and non-aqueous phase liquids.
- Volume minimisation: by achieving a reduced product volume relative to the raw waste or cement encapsulation baseline, leading to fewer waste packages.
- Product durability: by incorporation of radionuclides into a solid matrix, at the atomic scale, through formation of chemical bonds in a solid material of low solubility.

Nevertheless, the complexity and off gas handling requirements of a high temperature process pose a challenge to implementation, although this is not considered insurmountable, as demonstrated by operational experience of the Sellafield Waste Vitrification Plant. Development of sufficiently robust process and formulation envelopes, tolerant to feed variation, is also a key consideration. A key issue for thermal products is the potential interaction with a high pH lime based backfill, which may adversely impact the long term chemical containment safety function. It is also recognised that there is a need to demonstrate the disposability of thermally treated PCM in terms of operational and post closure criticality safety and that a high pH backfill may not be the most optimal solution.

In the UK context, a toolbox of thermal technologies is of interest for potential application to treatment of a diverse family of waste streams (see section 4.2.1), which are summarised below.

Joule Heated In-Container Vitrification: in which radioactive wastes are melted in a refractory lined container, with suitable additives, to form a product comprising a glass or slag material, with or without a separate metal phase. The melt is heated by passing a current between graphite electrodes; depending on the nature of the waste, it is possible to top up the melt with additional feed during processing. Since the melt is processed inside the refractory lined container, it is not necessary to pour the melt on completion of the thermal cycle.

Joule Heated Ceramic Melters: in which the radioactive waste, typically in slurry form, together with glass forming additives, is fed into a refractory lined melter during continuous process operation. The melt is heated by an electrical current passed between submerged electrodes, and may be agitated by an air sparge or stirrer. An air lift or bottom drain allows discharge of the melt into a container; the product is typically a homogeneous or heterogeneous glassy material.

Plasma melting: in which the radioactive waste, together with additives, is melted in a refractory lined or cold crucible melter; the system may operate as a continuous or batch process depending on the feed and system design. The melt is heated by a plasma torch system operating in either transferred or non-transferred mode, where the electric arc is sustained between electrode to melt or electrode to electrode, respectively. A refractory lined melter requires a bottom drain or other mechanism to discharge the melt into a container. The product is typically a glass or slag material, with a separate metal phase depending on the feed.

Hot Isostatic Pressing: in which the radioactive waste, together with additives, is packed into a can typically fabricated from stainless steel; the can is baked out and evacuated (to remove volatiles), sealed, and the work piece processed at high temperature and hydrostatic pressure (<200 MPa). HIP utilises radiative heat transfer to the work piece and may produce a glass-ceramic, ceramic or metal encapsulated waste form in a hermetically sealed can, that serves as primary packaging.

As noted above, an off gas system is required for management of volatile substances evolved in these thermal treatment technologies, except in the case of HIP.

In addition to this toolbox of thermal treatment technologies, other approaches under development internationally, include pyrolysis, gasification and cold crucible induction melting, however, these are not yet being actively pursued in the UK context.

4.3.3 Basis of enhanced encapsulation concepts

This review considers enhanced encapsulation technology being actively developed for treatment of radioactive wastes in the UK context. Two encapsulation systems are of primary interest, where considerable progress has been demonstrated in recent years:

- Geopolymers and alkali activated slags
- Magnesium phosphate cements

Other encapsulation systems of potential interest in the future may include calcium sulfoaluminate and calcium aluminate cements.

Cement encapsulation technologies all rely on the complex exothermic hydration reaction of materials with cementitious properties, to form a paste, which hardens to bind any supplementary, aggregate, or waste material, into a hard, solid, mass. The binder phase is generally a hydrated crystalline or amorphous compound. SCMs are typically added to improve

the workability of the cement paste and control heat output (by dilution), which may lead to cracking or undesirable phase assemblage. SCMs may exhibit some pozzolanic character, undergoing reaction within the cement paste, to produce additional binder phases. The current baseline technology for cement encapsulation of radioactive wastes is based on OPC, typically blended with FA or GBFS as an SCM; hydration leads to the formation of a poorly crystalline calcium silicate hydrate phase as a binder.

Of particular interest in the UK context are:

Geopolymers and alkali activated cements: which comprise an aluminosilicate source, typically FA, GBFS, or metakaolin, that is activated by addition of an alkaline solution, such as sodium hydroxide or sodium metasilicate (112). The activation process involves dissolution – precipitation reactions to form a disordered calcium-, or highly cross linked sodium-, aluminosilicate hydrate gel, which acts as a binder phase in the set cement, depending on calcium content of the source. Compositions from a low calcium aluminosilicate source are generally referred to as geopolymers.

Magnesium phosphate cements: which utilise an acid-base reaction between magnesium oxide and an acidic phosphate source, to produce a set cement in which the binder phase is a hydrated magnesium phosphate phase (113). For radioactive waste applications, the formation of struvite-K, $\text{KMgPO}_4 \cdot 6\text{H}_2\text{O}$, is preferred, because of the favourable kinetics of formation from suitable MgO and KH_2PO_4 reagents, which may be tuned by GBFS or FA addition (104). Such MKPC cements were originally developed by Argonne National Laboratory, as Ceramicrete, for application to challenging tank wastes on the Hanford and Mayak sites (115) (116). The internal pH of magnesium phosphate cements is near neutral, and, consequently, these materials are of potential interest for reactive metal encapsulation.

The potential benefits of enhanced encapsulation technology for storage and disposal, include:

- Improved chemical compatibility with some waste streams that retard the setting behaviour of OPC blends.
- Improved tolerance to and incorporation rate of non-aqueous liquids, such as oils or solvents.
- Lower internal pH which reduces the corrosion rate of reactive metals such as aluminium or uranium.

A key consideration for enhanced encapsulants is the potential interaction with a high pH lime based backfill, which may adversely impact the long term chemical containment safety function.

4.3.4 Compatibility with Current UK Government and Devolved Administration Policies

The development and implementation of thermal and enhanced encapsulant technologies, as part of a treatment tool kit to optimise outcomes, is considered in NDA strategy (5), and further developed in the NDA Strategic Position on Radioactive Waste Treatment (12), to deliver government policy objectives. These technologies are considered to be opportunities for realising the NDA Strategic Objective to complete treatment of all ILW, and therefore enabling Strategic Objectives of moving all ILW to interim storage and disposal in England and Wales. NDA recognises that development of alternative treatment baselines delivered by its strategy will also help to support implementation of Scottish government policy. NDA Strategy,



recognised that significant progress had been made in research and development of thermal treatment strategies to understand the scale of the potential benefits.

The Thermal Treatment IPT delivered a programme of work, to understand waste applicability, technology capability and economic aspects, with demonstration trials under nuclear site licence conditions. This work confirmed significant volume reduction and waste passivation could be achieved and is continuing through the Sellafield led Higher Activity Waste Thermal Treatment Programme. NDA also committed to coordination of effort across the group to maintain and develop enhanced encapsulants that could offer operational or long-term performance benefits.

In summary, development and implementation of thermal and enhanced encapsulation technology is an integrated priority within NDA Strategy and, therefore, fully compatible with current government policy.

4.3.5 Compatibility with IGD

Thermal treatment

In principle, thermal treatment could be applied to a diversity of waste streams in the UK Radioactive Waste Inventory, including: sludges, fuel debris, ion exchange materials, and PCM, as demonstrated in laboratory scale and pilot scale trials discussed in Section 4.2.1. The primary requirements for waste stream compatibility with thermal treatment, are that it must be in a form appropriate for the feed mechanism of the technology system, and undergo reaction with formulation additives, at high temperature, to afford a disposable product, with acceptable management and recycling of secondary wastes. If the technology system requires a melt to be homogenised, drained or poured, then the melt must also have appropriate characteristics. To achieve the minimum volume, and maximum waste loading, the waste constituents should ideally form part of the immobilisation matrix, minimising the quantity of formulation additives. Overall, the flexibility in technology and formulation envelope, establish a wide technological solution space for application of thermal treatment to the UK Radioactive Waste Inventory.

The impact of thermal treatment on the IGD is difficult to quantify at the present time because it is sensitive to the waste form and packaging solution. The nature of thermal treatment technology is to concentrate and contain radioactivity, as a consequence of volume reduction. Thermal treatment is therefore unlikely to significantly reduce the quantity of radioactivity consigned to the GDF in ILW, by diversion to NSD. An initial appraisal of the potential for volume reduction achieved by thermal treatment of radioactive wastes estimated a reduction factor of between 0-30% for metallic wastes, 0-80% for inorganic wastes and in excess of 90% for organic wastes, relative to the unconditioned waste stream (117). These estimates are broadly validated by early, and more recent, trials at laboratory and pilot scale, reviewed in Section 4.2.1. These estimates should nevertheless be considered indicative, being sensitive to the waste feed, technology and packaging solution, which are not defined. In contrast, baseline cement encapsulation technology would be expected to increase the conditioned waste volume by a factor of 20% or more. The available evidence therefore points to the potential for substantial reduction in packaged waste volume and waste packages, achieved by thermal treatment, relative to the baseline, translating into potentially substantial capital and operational cost savings in geological disposal and interim storage capacity.

Enhanced encapsulants

Enhanced encapsulants are likely to find application to ILW regarded as challenging for conventional encapsulation by OPC blends, either as a consequence of adverse chemical



composition, reactivity, including formation of expansive reaction products, or formation of a separate non-aqueous phase or gas production, to an unacceptable degree. The most likely waste streams for consideration of enhanced encapsulation, as a result of potential compatibility issues with OPC systems, include:

- Inorganic and organic ion exchange materials; issues include potential for reverse ion exchange reactions; degradation to yield complexing agents; adverse reactions with cement matrix.
- Aluminium; issues include: corrosion, hydrogen production, and expansive corrosion products.
- Uranium; issues include: corrosion, pyrophoric hydride formation, hydrogen production, and expansive corrosion products.
- Magnox fuel: issues include: corrosion, pyrophoric hydride formation, hydrogen production, and expansive corrosion products.
- Oils and solvents; issues include: degradation to form complexing agents and non-aqueous phase liquids.

These compatibility issues arise from a combination of the very high internal pH and internal free water of OPC encapsulants, combined with the reactivity of the waste material.

Some key requirements of an enhanced encapsulation system, include:

- Sufficiently fluid to penetrate and encapsulate irregular materials, with minimal bleed.
- Minimal water content for workability to reduce internal corrosion reactions.
- Chemical compatibility with waste, e.g. to avoid expansive reactions.
- Sufficient compressive strength to enable handling and transport.

It is expected that the conditioning factors of enhanced encapsulants would not be significantly different from the OPC baseline, and, therefore, application would not significantly impact the radionuclide inventory, packages waste volume, or number of waste packages consigned to a GDF.

4.3.6 Technical Maturity

In general terms, thermal treatment technologies are considered to be relatively mature, with laboratory and pilot scale studies, and considerable operational experience of some technologies for radioactive waste treatment. Nevertheless, thermal technologies are assessed to be of medium to high maturity, in the context of application to specific UK waste streams with further development required, as summarised in the following table.

Technology	Technical Maturity	Summary
Joule Heated In-Container Vitrification	Medium – high	The GeoMelt® system is an industry-leading example of the approach, developed in 1980. Since then, it has processed over 26,000 tonnes of waste, predominantly through in-situ vitrification. It has been demonstrated within the last 5 years at both pilot plant-scale in the UK (118), and full operational scale in the USA (119).
Continuous Joule Heated Vitrification	Medium – high	The technology has been successfully implemented for the vitrification of HLW at several facilities at United States Department of Energy sites, as well as in Germany, Belgium, Russia and Japan.
Plasma melting	Medium – high	Refractory plasma melting treatment systems have been deployed worldwide for the treatment of radioactive wastes, specifically VLLW and LLW (93). The Zwiilag Plasma Melter in Switzerland has been operating since 2004 for the treatment of 200 tonnes of LLW per year. This is a similar throughput to that for VLLW at the refractory melting system at Kozloduy Plasma Melting Facility in Bulgaria, with the output being a volume reduction of up to eight times.
Hot Isostatic Pressing	Medium	No fully active waste treatment plant employing HIP technology has yet entered operation, however, the first of a kind facility to treat radioactive wastes from Mo-99 production for medical applications has been constructed and is in its pre-commissioning phase in Australia (118).

Cement encapsulation, using OPC blend formulations, is the current baseline technology for ILW treatment, and encapsulation plants are in operation at UK and international nuclear sites. The technology to implement cement encapsulation is therefore considered to be of high maturity. The specific alternative enhanced encapsulants considered here, and under development for application in the UK, are summarised in the following table.

Technology	Technical Maturity	Notes
Geopolymers	Medium – high	Laboratory and full scale trials have established compatibility and application to simulants of waste streams in the UK Radioactive Waste Inventory. Commercial products and solutions are available on the market and some have a long history of implementation.
Magnesium phosphate cements	Low – medium	Recent research and development has focused on understanding the fundamental chemistry and properties of the encapsulation system and compatibility with some niche elements of the UK Radioactive Waste Inventory as simulants. Although commercial products exist, they do not yet have a track record of application to radioactive wastes comparable to geopolymers.

4.3.7 Post Closure Safety Considerations

Thermal treatment offers the potential for treating some radioactive wastes, such as a fraction of the PCM inventory, which are not compatible with the current baseline (compaction and encapsulation). The elimination of organic materials and waste reactivity, by oxidation, removes the need to consider the potential contribution to complexant, non-aqueous phase and gas phase mediated transport of radionuclides in the post closure safety case. The available data, and recent research summarised in Section 4.2, evidence that the conceptual products of thermal treatment undergo slow dissolution under both accelerated and GDF relevant dissolution experiments that emulate the use of a cementitious backfill. Typically, these conceptual products present a chemical durability similar to, or exceeding that of, UK HLW glasses under comparable conditions of testing.

The UK illustrative disposal concept for ILW in Lower Strength Sedimentary Rock (LSSR) or Higher Strength Rock (HSR), involves the use of a cementitious backfill to maintain alkaline conditions over 100,000 years, to assure chemical containment of solubility limited radionuclides. The compatibility of this concept with prototype vitrified products of thermal treated ILW was investigated in a programme of research to understand the corrosion of laboratory and pilot scale products in a simplified cement porewater at 50°C (to accelerate dissolution). In general, glass dissolution was observed to be inhibited at early time periods, as a result of calcium incorporation into the alteration products, thereafter, showing the generalised transition from an initial to relatively slow residual dissolution rate, as a result of solution saturation effects. Further investigation is required to understand the long term behaviour of such vitrified product dissolution in conditions relevant to a cement backfilled vault, with consideration of conditioned groundwater chemistry. The concept and requirements for a vault optimised to accept thermal treatment products are not yet defined, but could conceivably utilise an alternative backfill material.



Enhanced encapsulant based waste forms are expected to undergo long term hydration and alteration processes, such as carbonation, depending on their chemistry. These reactions will result in changes of the mineralogy of the encapsulant material and changes in the compressive strength, which may be beneficial. Radiolysis may result in gas generation. The improved compatibility of enhanced encapsulants with radioactive wastes, should reduce expansive corrosion reactions, corrosion of reactive metals, and degradation of organic materials, relative to the expected performance of counterpart OPC encapsulated wastefoms in the IGD. In principle, implementation of enhanced encapsulants is therefore expected to contribute to post closure safety by reducing potential contribution to complexant, non-aqueous phase and gas phase mediated transport of radionuclides, relative to the current baseline. However, evidence to support this claim is currently of a foundational nature and considerable further investigation is required to understand the long term stability, hydration and alteration processes of enhanced encapsulants, and the mechanism and rate of encapsulated waste corrosion and degradation processes.

Once the waste container is breached, slow dissolution of the enhanced encapsulation matrix and leaching of radionuclides, will result in radionuclide migration into the engineered barrier system. The rate of dissolution and leaching will depend on the composition of the cement matrix, encapsulated waste, buffer material, and groundwater, and chemistry of the radionuclides. There is currently little available knowledge concerning the interaction of enhanced encapsulants with a cementitious backfill, which is currently assumed for ILW vaults in LSSR and HSR geology.

The enhanced encapsulant products investigated to date are at an early stage of development and there remains considerable scope for optimisation of the wasteform product and backfill solution.

4.3.8 Timescale for Implementation

The timescale for implementation of thermal and enhanced encapsulation technology will depend on the nature of the waste to be treated, which will set the requirements on the feed, treatment, effluent and off gas handling, and waste packaging systems. Illustrative examples from international experience, noting the different licensing and permitting basis, include:

- Kozloduy plasma facility: contract to build facility signed in 2009, facility entered operation in 2018.
- Zwiilag plasma facility: facility construction started in 1997, and the facility entered operation in 2004.
- ANSTO Synroc plant: construction commenced in 2018, facility expected to enter operation in 2028.
- Veolia 10 ton Joule Heated ICV facility: designed, constructed, and delivered to operation in 2018, noting a substantial preceding demonstration programme.

Therefore, from acceptance of final business case, an illustrative timescale for implementation of thermal and enhanced encapsulation treatment technology, could be around 10 years. This assumes an adequate foundation of research, development, and demonstration, which was integral to the success of the aforementioned facilities, as discussed in this report. Repurposing

of an existing cement encapsulation facility, if viable, could provide an alternative route to implementation of enhanced encapsulation technology.

4.3.9 Cost

At present, there is not sufficient certainty in the cost and benefit of the implementation of thermal and enhanced encapsulant technology, and the applicable inventory, to be confident of the impact on the whole lifecycle cost for radioactive waste management. Nevertheless, in the case of thermal treatment technology, laboratory and demonstration scale trials with relevant UK waste simulants, evidence volume reduction factors of 50 – 80%, compared to the waste feed, whereas the cement encapsulation baseline would result in a relative increase of packaged volume. Deployment of a mature thermal treatment technology could potentially result in substantial cost savings in the capital and operational cost of stores and disposal facilities, recognising the latter should be considered in terms of marginal cost avoidance. Savings would arise from reduced volume of packaged waste, number of waste packages, and reduced operational mission of facilities. Irrespective, implementation of thermal treatment and enhanced encapsulants may be necessary to adequately treat some wastes not compatible with baseline cement encapsulation. Enhanced encapsulation, being broadly comparable to conventional OPC encapsulation, in terms of waste package volume and numbers, is unlikely to have radical impact on overall lifecycle cost of radioactive waste management, but may enable treatment of wastes with no or sub-optimal treatment route.

5. Conclusion

Our review of alternative options to geological disposal for managing the IGD, has considered key developments relating to:

- Alternative disposal options for waste destined for geological disposal. These include NSD and DBD disposal concepts.
- Alternative waste treatment techniques which could alter the nature and/or reduce the volume of waste requiring geological disposal.

We have concluded that although alternative options may conceivably optimise the future IGD, a GDF will be required to manage the majority of the current IGD to a safe and secure end point. These radioactive wastes will require the long term isolation and containment afforded by the depth and multibarrier system of a GDF. No alternative option, or combination of alternative options, could avoid the need for development of a GDF.

The NDA group model, including NWS, enables acceleration and implementation of alternative management options, to achieve optimisation and integration of radioactive waste management, across the lifecycle. This is reflected in both our guiding strategic priorities which involve the implementation of alternative options for management of appropriate components of the IGD. These programmes involve need and opportunity led research and development, commissioned in the supply chain, to develop the evidence base required to implement alternative options. We also continue to work in collaboration with international projects, for example delivered through the European Joint Programme on Radioactive Waste Management (EURAD), where alignment of objectives, partnership of capability, and sharing of costs, enables us to realise technical and community outcomes, building international consensus, and demonstrating value for money. Our role in networks such as EDRAM and the Network of Waste Management Agencies, and with organisations such as Nuclear Energy Agency (NEA) and IAEA, enables us to benefit from, and contribute to, the developing international consideration of alternative management options.

The summary findings of this review are:

Near-surface Disposal

NSD of radioactive waste is a mature technology, in both at-surface and at-depth variants, with decades of international experience over construction, operation, and closure. There is international consensus and confidence in the safe disposal of appropriate short-lived or less hazardous LLW and ILW, underpinned by a rigorous environmental safety case. In common with other nations having nuclear power programmes, the UK already has decades of experience of LLW disposal in at-surface NSD facilities both at the LLWR site and LLW Disposal Facility in Dounreay. The UK has not so far adopted at-depth NSD for the more hazardous wastes, as vaults or silos, although there is considerable relevant international experience in Finland and Sweden. NSD is already covered by current Scottish Government policy on the management of the most hazardous radioactive waste and a future implementation of NSD in England and Wales for a proportion of the less hazardous waste was enabled by the UK policy framework for managing radioactive substances and nuclear decommissioning, published in May 2024 (3). The availability of an NSD capability for a



proportion of the less hazardous waste in the ILW category, may enable earlier hazard reduction and overall cost savings, subject to a rigorous environmental safety case.

Deep Borehole Disposal

Disposal of radioactive waste in deep boreholes has been advocated for decades, as an alternative to geological disposal. It has been argued that the approach is sufficiently mature for deployment, drawing on borehole drilling technology currently applied in the oil and gas sector. However, although components of the technology have been demonstrated in isolation, without the constraints of a nuclear operating environment, demonstration of the fully integrated solution, involving borehole construction, waste package emplacement / retrieval, sealing and closure, monitoring, and *in situ* validation of post closure safety has not been realised. This stands in contrast to the body of evidence for the safe implementation of geological disposal achieved in national research programmes and underground research laboratories. Therefore, the feasibility and operational readiness of DBD technology remains of low to medium maturity. Substantial further development is required to increase the maturity of the approach for potential application, including consideration of operational and post closure safety. Notwithstanding these challenges, there is consensus that DBD could be applicable, in principle, to the dimensionally compatible elements of the high heat generating waste in the IGD, from a conceptual stand point. Therefore, DBD could only supplement, and not replace, the need for a GDF for the components of the IGD which are certainly not compatible with DBD. Cost and schedule estimates for DBD are not sufficiently certain to have confidence in the potential marginal cost savings achievable, versus geological disposal. Nevertheless, the demonstrator development and initial deployment costs seem within the reach of private equity investors, which offers an alternative route to market. At time of writing, deployment of DBD technology in a nuclear environment would require further policy and legislative consideration, both of which would be a matter for government. Regulatory approaches would also have to be developed to ensure proper regulation of operational safety and security alongside long-term environmental protection. Going forward, NDA and NWS will continue to maintain a watching brief on maturation of DBD technology.

Alternative Treatment Technology

Considerable progress has been made in the advancement of alternative treatment technology, to increase readiness for implementation. Laboratory scale investigation has effectively achieved proof of concept application of thermal treatment and enhanced encapsulants to problematic UK wastes, using simulant and surrogate feeds, including PCM, Magnox sludge and fuel debris, ion exchange materials and reactive metals. These investigations have provided insight on the nature of the product materials, volume reduction factors, passive safety and suitability for geological disposal. In parallel, successful trials to demonstrate pilot scale thermal treatment of simulant ILW wastes have established the compatibility and feasibility of technology implementation, in principle. Alternative thermal and enhanced encapsulant treatment technologies will not displace the need for a GDF, but can contribute to realising a more optimal end point in terms of operational and post closure safety, and reduced timescales and cost.

Looking Forward and Next Steps

NWS has updated its awareness of alternative options for management of the IGD, to potentially optimise disposal of radioactive wastes to any future near-surface and deep geological disposal facilities. It is evident that these alternative options cannot fully displace the need for a GDF, but may enable earlier hazard reduction and overall cost savings, without any



compromise on safety, and a more optimal end point for the decommissioning mission. Going forward, we will undertake systematic horizon scanning to ensure good awareness of alternative options and technology developments to ensure we achieve greater effectiveness and realise potentially transformative benefits.

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