Executive Summary

The United Kingdom has operated a wide range of nuclear facilities for over half a century and these activities have led to the generation of radioactive wastes that are stored at various locations around the country. Every three years an inventory is taken to record the characteristics and quantities of the different waste streams, along with details of how they are contained and the plans for their eventual disposal. Included in this inventory is the spent fuel from the UK's fleet of nuclear reactors, which needs to be safely managed following removal from the reactor cores. One option for storing and transporting wastes, including spent fuel, is a dual-purpose cask (DPC). Using a DPC for interim storage and then transport can simplify surface operations, but a DPC may not be suitable for deep disposal. This report considers relevant guidance, case studies and inventory data to investigate the storage, transportation and ALARP arguments for the implementation of a DPC system.

The 2013 Derived Inventory is explored to provide a broad perspective of the different waste streams that are in stock and are expected to arise in the future; breaking down the data to investigate how these wastes are packaged and how many of these packages will need to be disposed of at a Geological Disposal Facility (GDF). The top ten materials expected to make up the UK's inventory of radioactive waste are demonstrated, with plots included to illustrate when these materials are expected to arise in the future, as certain stations and facilities are decommissioned. Most waste packages have associated transport containers that they are designed to be compatible with; the number and type of these anticipated transport containers is also included.

The transportation of the radioactive waste packages identified in the inventory from their storage location to their disposal location will be a significant undertaking, especially considering how the country's nuclear facilities are scattered around the UK and many are likely to be hundreds of miles from a potential GDF. The routes and methods for how these packages may be transported to a GDF are discussed, with rail and road likely to be favoured options in most circumstances. The transport regulations are explained and deconstructed, with a detailed explanation of each of the criteria that are required to demonstrate compliance.

DPC systems have been implemented in several countries and the IAEA have published some guidance with respect to the methodology that is required to build a safety case for the storage and transportation of spent fuel in DPCs. This guidance, along with a range of research from other relevant organisations, is summarised to demonstrate the range of sources available to guide ONR on potential future safety cases for DPCs to store and transport radioactive materials, including spent nuclear fuel, in the UK.

A section is also included to review the ALARP considerations around the justifications of implementing a DPC system, which include: handling systems and repackaging; containment barriers; the fuel assembly capacity of a DPC; the benefits of DPCs for failed fuel assemblies; the importance of ageing management and monitoring of DPCs during interim storage; and also the vulnerability of ALARP justifications to future changes to both regulation and the GDF design.

Abbreviations

ADR	European Agreement for the International Carriage of Dangerous Goods by Road
AGR	Advanced Gas-cooled Reactor
ALARP	As Low As Reasonably Practicable
AMPs	Ageing Management Plans
BEIS	Department for Business, Energy and Industrial Strategy
Bq	becquerel
BWR	Boiling Water Reactor
C&U	Construction and Use
CDG	Carriage of Dangerous Goods
CoA	Certificate of Approval
CSA	Criticality Safety Assessments
CSI	Criticality Safety Index
DCICs	Ductile Cast Iron Containers
DCTC	Disposal Container Transport Container
DDP	Design Decision Panel
DI	Derived Inventory
DNLEU	Depleted, Natural and Low Enriched Uranium
DoE	US Department of Energy
DPC	Dual-purpose Cask
DPCSC	Dual-purpose Cask safety Case
DSR	Design Safety Report
EDF	Électricité de France Energy Nuclear Generation Limited

EPRI	Electric Power Research Institute
ESCP	Extended Storage Collaboration Project
EST	Extended Storage and Transport
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GDSS	Generic Disposal System Specification
GNS	Gesellschaft für Nuklear-Service
GTSD	Generic Transport System Design
GWPS	Generic Waste Package Specification
HAW	Higher Activity Wastes
HBU	High Burn-up
HGV	Heavy Goods Vehicle
HHGW	High-Heat-Generating Waste
HI-STORM	Holtec International Storage Module
HI-STORM MIC	Holtec International Storage Module Mega Impact Capable
HI-TRAC	Holtec International Transfer Cask
HLW	High Level Waste
IAEA	International Atomic Energy Agency
IGD	Inventory for Geological Disposal.
ILW	Intermediate-Level Waste
IMDG	International Maritime Dangerous Goods
IPs	Industrial Packages
ISFSI's	Independent Spent Fuel Storage Installations
LAW	Lower Activity Waste

LC	Licence Condition
LHGW	Low-Heat-Generating Waste
LI	Licence Instrument
LLW	Low-Level Waste
LLWR	Low-Level Waste Repository
LoC	Letter of Compliance
LSA	Low Specific Activity
MBGWS	Miscellaneous Beta Gamma Waste Store
MoD	Ministry of Defence
MPCs	Multi-purpose Containers
NDA	Nuclear Decommissioning Authority
NDE	Non-destructive Evaluation
ONR	Office for Nuclear Regulation
PATRAM	Packaging and Transportation of Radioactive Materials
PNTL	Pacific Nuclear Transport Limited
PSRG	Project Safety Review Group
PWR	Pressurized Water Reactor
RAMTRANS	Radioactive Materials Transport and Storage
RID	International Carriage of Dangerous Goods by Rail
RWM	Radioactive Waste Management Limited
SAPs	Safety Assessment Principles
SCO	Surface Contaminated Object
SF	Spent Fuel
SFAIRP	So Far As Is Reasonably Practicable

SFM	Safe Fissile Mass
SLC	Site Licence Company
SNM	Special Nuclear Materials
SRN	Strategic Road Network
SSG	Specific Safety Guide
SSR	Specific Safety Requirement
STGO	Special Types (General Order)
SWTC	Standard Waste Transport Container
T&CP	Testing and Commissioning Panel
TAGs	Technical Assessment Guides
TCA	Transport Competent Authority
TDC	Transport and Disposal Container
TRANSSC	Transport Safety Standards Committee
TRNs	Trunk Roads Networks
TSA	Transport Safety Assessment
TSC	Transport Safety Case
TSSA	Transport System Safety Assessment
UKRWI	UK Radioactive Waste Inventory
US / USA	United States of America
USNRC	United States Nuclear Regulatory Commission
WAC	Waste Acceptance Criteria
WAGR	Windscale Advanced Gas-cooled Reactor
WASSC	Waste Safety Standards Committee

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Appendices

1 Introduction

The United Kingdom of Great Britain and Northern Ireland (UK) was a pioneer in the development of nuclear technology for use in power production, defence, medicine, research and other industrial applications. These activities have led to the accumulation of radioactive wastes and materials at various locations around the country. The wastes and materials have been stored at interim surface facilities since their production but require a consolidated solution to ensure their long-term isolation and containment. Most Low-Level Waste (LLW) can be disposed of at the Low-Level Waste Repository (LLWR) in Cumbria. However, the Higher Activity Wastes (HAW), which comprises High-Level Waste (HLW), Intermediate-Level Waste (ILW) and some LLW that cannot be disposed of at LLWR or the LLW facility at Dounreay, require a more robust solution.

The policy of UK government and the devolved administrations of Wales and Northern Ireland for management of HAW is set out in the Implementing Geological Disposal White Paper [1], which details how England, Wales and Northern Ireland will dispose of this waste via a Geological Disposal Facility (GDF). Radioactive Waste Management Limited (RWM) is a subsidiary of the Nuclear Decommissioning Authority (NDA) and has been established as the delivery organisation responsible for the implementation of a GDF. The current policy of the Scottish government is to store HAW in near-surface facilities located as near as possible to the site where the waste is produced.

Information on radioactive wastes in the UK is compiled in the UK Radioactive Waste Inventory (UKRWI) which is updated on a three-yearly cycle, with the last published version being the 2016 UKRWI [2]. The wastes in the UKRWI that will require geological disposal are set out in the White Paper [1] and are recorded in a separate inventory, previously known as the Derived Inventory (DI) up to 2013 [3] but now called the Inventory for Geological Disposal (IGD). The IGD is composed of a subset of UKRWI wastes (generally the more highly active wastes), plus some materials such as spent fuel and stocks of uranium that have not yet been classified as waste. The IGD also includes additional information on the waste streams that is relevant to disposal. The 2016 IGD [4] was published in December 2018. By then, the analysis phase of the work described in this report (using data from the 2013 DI) was already complete. This report refers to previous versions of this inventory as the DI and future versions as the IGD. Differences between the 2016 IGD and the 2013 DI should not affect the conclusions of this work.

Both the UKRWI and DI/IGD report radioactive wastes in 'waste streams'. Each waste stream is assigned an identification code. A **waste stream** is a convenient grouping of wastes with common properties such as origin (reactor or processing plant) and material composition. **Radioactive materials** are radioactive items that are not classed as waste now but may be in the future if no further use can be found for them. In general, data on radioactive materials are collected separately from the UKRWI. The DI and IGD include materials, such as uranium and plutonium recovered from spent fuel, that are not currently categorised as waste but may be so in the future, or will require packaging and transporting as such. The data for the UKRWI and the DI are stored in RWM's DIQuest database.

Containers for radioactive waste are designed to provide a physical barrier and enable safe handling. DIQuest records which of the many container types is assigned to each waste stream.

The term 'waste package' refers to the unit that will be disposed of in the GDF and includes the waste, the encapsulation or immobilisation material, and the disposal container (Figure 1) [5]. The waste packages are designed to be compatible with safe transport to, handling at, and disposal in the GDF, and expected to be able to provide the following five operational safety functions:

- Provide containment of radionuclides and other hazardous materials during routine, normal and accident conditions;
- Limit radiation dose to workers and members of the public;
- Preclude criticality;
- Provide the means of safe handling; and
- Withstand internal and external loads [5].

The barriers provided by the waste package will play key roles in achieving the required degree of safety during transport and will continue to do so during the operational period of the GDF [5].

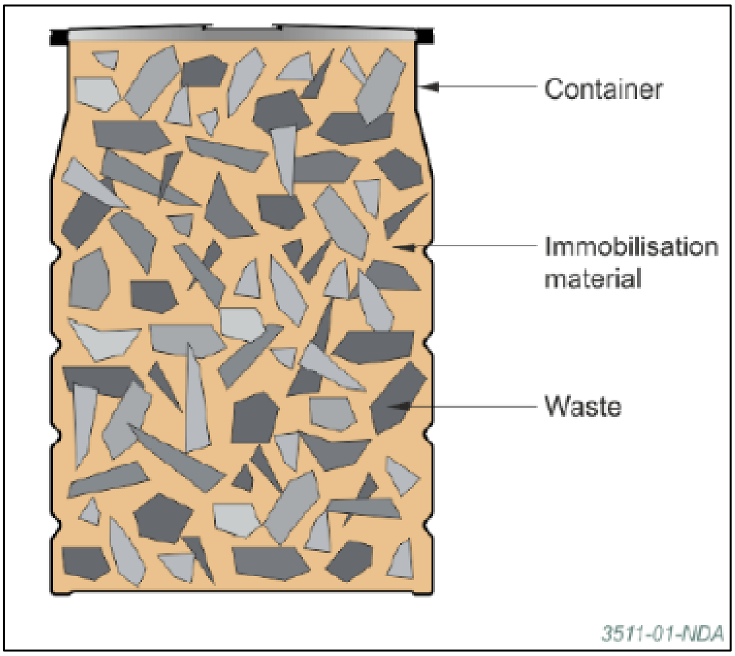

Figure 1 The components that make up a waste package [5]

Specific requirements for wasteforms and waste packages are set out in the Generic Disposal System Specification document published by RWM [5], and consider properties such as:

- Activity content;
- Gross mass;
- External dose rate;
- Heat output;
- Surface contamination;
- Gas generation;
- Criticality safety; and
- Accident performance.

The specification document [5] includes a list of containers that are available for use by waste producers. These containers vary in design to provide packaging solutions for a wide range of wastes, and many container types have multiple variants that are designed for specific purposes. Subsequent sections of this report provide further detail on these containers.

The storage locations for radioactive waste packages in the UK are illustrated in Figure 2 which is taken from the Generic Waste Transport System Design [6]. In the figure, the wastes are categorised as either high-heat generating (e.g. spent fuel or vitrified HLW, but also including plutonium and highly enriched uranium) or low-heat generating waste (typically ILW from operations or decommissioning). It is useful to differentiate these two groups, as different disposal solutions need to be provided at a GDF for the two waste categories.

Figure 2 Storage locations for the wastes identified for disposal at a GDF [6]

The Ministry of Defence (MoD) sites in Scotland at Vulcan (Dounreay) and Rosyth host a relatively small number of waste packages (approximately 100) that are included in the 2013 DI. However, in accordance with Scottish Government policy, waste packages arising at *civil* nuclear sites in Scotland are not included in the 2013 DI.

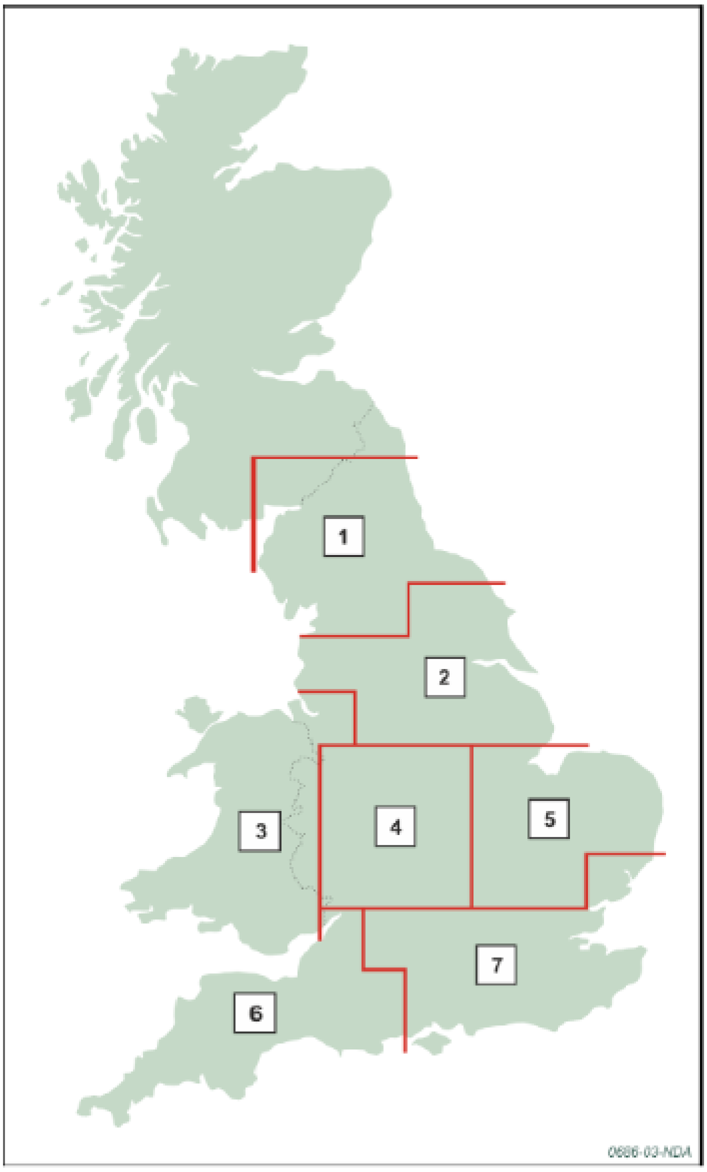

Figure 3 Zones and notional GDF locations considered in the generic TSD [6].

The waste packages identified in the IGD will require transport from their interim storage locations to the GDF once it is constructed and becomes available. It is currently envisaged that most waste packages will be transported inside a re-usable transport container that will not form part of the disposal unit. The safety aspects of this transport operation are set out in RWM's generic Transport Safety Case (TSC) [7]. It is assumed in the generic TSC that a single GDF will be developed, capable of receiving all the wastes in the IGD. Wherever a GDF is located, it will be developed with connections to the national rail network and the strategic road network.

The TSC is generic because no site has yet been selected for a GDF. Therefore, all design and planning work must be done in a non-specific way so as to make the preparations applicable to all potential locations. In order to scope assessment of transport safety, RWM has adopted an approach whereby the land area of England and Wales is divided into seven zones of approximately equal area. A GDF is notionally located at the centres of these zones. These zones and notional GDF locations are illustrated in Figure 3.

The generic TSC was first published in 2010 [7]. A number of changes, including the inventory of wastes requiring disposal, led to an update to the generic TSC being published in 2016 [8] which was based on data from the 2013 DI.

With the design and operation of a future geological disposal solution in the conceptual stage and hence open to modification, the consideration of new types of containers to store waste or transfer packages is regularly investigated. The dispersed locations of the UK's nuclear sites, illustrated in Figure 2, will require the transport of hundreds of thousands of waste packages stored around the country to the GDF. One option for simplifying this transport is to use a Dual-Purpose Cask (DPC) for both interim storage and transport, and possibly also for disposal. Currently, many of the waste packages identified in the IGD must be transported inside a transport container.

There are many drivers behind the motivation to develop a regulatory approach to the use of DPCs, as they have the potential to reduce both the number of containers and the handling facilities required for geological disposal. This could in turn increase the speed of construction and implementation and significantly reduce the cost of the programme. However, the process of approving the use of a container that performs multiple functions (and must do so over a long time period) is more challenging than approving a container that performs a single function soon after approval has been given. Significant research, testing and regulatory checks must be conducted before such a container can be considered for use.

The Office for Nuclear Regulation (ONR) has begun to consider how the UK will approach these points from a regulatory perspective. This report investigates the current situation in the UK and internationally to understand the advantages and disadvantages of DPC-based solutions. Guidance documents and ALARP arguments are considered to help explore the implications of introducing such a system in the UK.

The research in this report is limited to radioactive waste packages that require Competent Authority approval prior to transport, i.e. Type C packages, Type B packages and any packages that are fissile (and not exempt from fissile requirements).

2 Review of Packages, Waste Streams and Sites

The UK Government is responsible for taking stock of the country's radioactive waste packages every three years and this is sponsored through the Department for Business, Energy and Industrial Strategy (BEIS) and delivered via the NDA. This stocktake is reported as the UKRWI which accounts for the latest national record on radioactive wastes and materials in the UK.

A second supplementary inventory called the IGD (previously known as the DI) is produced by RWM and focuses on the HAW that will require geological disposal. This inventory builds on the information in the UKRWI and includes further information that assists the planning for a GDF. Both the UKRWI and IGD cover existing waste, future expected arisings, and items that are not currently classed as waste but may be so in the future if no further use can be found for them.

The waste information is recorded and reported in different waste streams that are given unique identification codes to enable tracking. Individual waste streams usually arise at a site from a particular facility or process and tend to have similar material properties, making their consolidation appropriate for similar treatment or storage routes. A considerable amount of information is recorded in each waste stream datasheet (see [8] for an example), which includes:

- Description of waste stream;
- Waste stream identification code;
- Waste classification;
- Volume in stock (existing volumes);
- Volumes forecast to arise and associated time periods;
- Specific activity; and
- Current or planned waste treatment and packaging.

The following sub-sections utilise the information from the 2013 DI to illustrate the current estimated quantity, type, and scheduled arising of waste packages from each nuclear site in the UK, with an assessment of the quantity and type of transport containers required for these waste packages.

2.1 Package Types

To consider the implications of implementing a container that can be used to both store and transport radioactive material, it is appropriate and relevant to understand the current situation in terms of the number of waste packages that have and will arise and extrapolate how many transport packages would be required to transfer these waste packages to an interim or final storage site. In this section, attention focuses on the transport packages included in the 2013 DI in order to show how many of each type of package have arisen or are expected to arise and where they are currently stored.

2.1.1 Waste Containers

The specifications for waste containers are based on the International Atomic Energy Agency (IAEA) safety standards for storing radioactive waste [9]. RWM is responsible for the Generic Waste Package Specification (GWPS), which defines the requirements for all waste packages destined for geological disposal. The GWPS provides the basis for the definition of the Generic Specifications, which define the standards and specifications for waste packages containing specific categories of waste. This specification is split into three categories defined by the properties of the waste:

- High-Heat-Generating Waste (HHGW);
- Low-Heat-Generating Waste (LHGW); and
- Depleted, Natural and Low Enriched Uranium (DNLEU) [10].

The Generic Disposal System Specification (GDSS) [4] outlines a list of waste containers expected to require disposal at a GDF, some of which include specific variants[1]. These containers are:

- 500-litre drum;
- 3 cubic metre box;
- 3 cubic metre drum;
- 6 cubic metre concrete box[2];
- 2 metre box;
- 4 metre box;
- Miscellaneous Beta Gamma Waste Store (MBGWS) box (currently used to store MBGW at the Sellafield site and which is expected to be suitable for disposal, but has not yet been endorsed);
- 500-litre concrete drum;
- Robust Shielded Containers (for example, cubical (Type 6) and cylindrical (Type 2) ductile cast iron containers (DCICs));
- 1 cubic metre concrete drum;
- DNLEU Transport and Disposal Container (TDC); and
- Disposal Containers for HLW and spent fuel.

This list represents the range of containers available for use by waste producers. The details for each of these waste containers are available in the specific Waste Package Specifications [10] and a future disposal system will need to handle these waste package designs.

The 2013 DI data include the expected waste package for each associated waste stream, including the specific container variants. However, this study only considered the high-level design and grouped the variants together to keep the analysis simple.

[1] Only containers that have been adopted through RWM's Change Management procedure and for which RWM has developed a Level 3 waste package specification are considered in the GDF design.
[2] Previously known as the Windscale Advanced Gas-cooled Reactor (WAGR) box.

2.1.2 Transport Containers

Transport containers are designed to provide shielding, containment and criticality safety for the radioactive contents, during routine, normal and accident conditions of transport. The detailed requirements for transporting packages that contain different categories of radioactive waste are set out in IAEA Regulations and Guidance documents for the Safe Transport of Radioactive Material. These international regulations help guide domestic laws and set recommended regulatory standards for national and international transport activities. They define six families of package:

- Excepted;
- Industrial (IP-1, IP-2, IP-3);
- Type A;
- Type B;
- Type C; and
- Uranium hexafluoride.

A brief definition of each package type is provided in Appendix C [11]. No specific package type is defined for fissile material, instead the regulations set out detailed requirements that must be met by packages containing fissile material and that need to be considered on a case-by-case basis.

The research in this report is limited to packages that require Competent Authority approval prior to transport, i.e. Type C packages, Type B packages and any packages that are fissile (and not exempt from fissile requirements).

The 2013 DI outlines the transport packages expected to be used for most of the waste streams in the UK inventory. Some waste streams still have undefined package types, such as those for spent fuel or new build, as decisions are yet to be made on how they will be managed in the future.

The Generic Transport System Design (GTSD) [6] defines what comprises the transport system and how the system will be operated. The document details the container designs that can be used for transporting radioactive wastes.

Less hazardous wastes (usually LHGW) are assumed to be packaged in 'industrial packages'; typically large steel or concrete boxes of waste. These waste packages qualify as transport packages in their own right. The types of waste container include:

- 2 metre box;
- 4 metre box;
- 6 cubic metre concrete box; and
- Transport and Disposal Container (TDC).

Designs have also been developed for two types of reusable transport container to transport waste to the GDF. The first is a family of Standard Waste Transport Container (SWTC) designs with a range of shielding thicknesses that will be used to transport the more hazardous LHGW packages. The second is a Disposal Container Transport Container (DCTC) design, which will be used for the transport of waste packages containing HHGW. These reusable transport containers are:

- SWTC-70;
- SWTC-150;
- SWTC-285; and
- DCTC.

An additional option of a transport overpack will be developed for transporting LHGW packages, however this enclosure will only be used for ease of handling and stowage during transport and will not form part of the transport package [6]. This option is not considered any further in this study.

Table 1 An illustration of the packaging options for low- and high-heat generating wastes for storage and transport [6]

Waste type	Waste category	Waste package	Transport arrangement
LHGW	ILW	500 litre drum	SWTC-70, SWTC-285
		3 cubic metre box	
		3 cubic metre drum	
		6 cubic metre concrete box	Transported as is
		2 metre box	Transported as is
		4 metre box	Transported as is
		MBGWS box	SWTC-150
		500 litre robust shielded drum	Transport overpack, SWTC-150
		3 cubic metre robust shielded box	Transport overpack
		1 cubic metre concrete drum	Transport overpack
		500 litre concrete drum	Transport overpack
	LLW	500 litre drum	SWTC-70, SWTC-285
		4 metre box	Transported as is
	DNLEU	Transport and Disposal Container	Transported as is
		500 litre drum	SWTC-70, SWTC-285
HHGW	HLW	Disposal container	DCTC
	SF[4]		
	Pu		
	HEU		
	MOX SF		

A more detailed analysis of the transport considerations is dealt with in subsequent chapters of this report.

2.2 Package Arisings

The radioactive waste inventories hold information on the volume of waste in stock and an estimate of the volume of waste expected to arise in each year[3] in the future. This data is provided by Site Licence Companies (SLCs) and provides the best estimate of the UK's total inventory of radioactive wastes. This data is useful for planning long-term decommissioning and disposal strategies, such as understanding the number and type of packages required, what materials are expected to require storage, and the volume of vault space required in a GDF.

The 2013 DI has been compiled using data sourced predominantly from the 2013 UKRWI. The data presented in the UKRWI for future waste arisings are projections made by the organisations that operate the sites where radioactive waste is generated. The projections are based on assumptions as to the nature, scale and timing of future operations and activities. With data based on projections in this way, it should be noted that there are limitations to the data in the 2013 DI. New build data is based on out-of-date assumptions on the expected future nuclear programme of three new EPR stations and three new AP1000 stations. Also, recent changes to the Toshiba and Hitachi programmes are not taken into account, neither is the proposed UK HPR1000 plant at Bradwell. The DI also has several waste streams that have no allocated storage location. Some of these limitations are touched upon in the subsequent sections.

2.2.1 Waste Package Arisings

Data from the 2013 DI were analysed to understand the number of waste packages required for waste that is in stock and expected to arise in the future. As this dataset was published in 2013, 'in stock' is defined as waste that was in existence on the DI reference date of 31 March 2013; 'expected arisings' refers to waste that is anticipated to be produced after this date. Some of the new-build arisings stretch out to 2137.

Table 2 Extrapolated radioactive waste packages for waste in stock and for expected future arisings from 2013 to 2137

Waste package	In Stock	Expected arisings
500-litre drum	196,493	113,689
$3m^3$ box	21,412	55,383
$3m^3$ drum	340	7,490
6 cubic metre concrete box	115	311
2m box	0	75
4m box	511	3,902
MBGWS box	851	655
Spent Fuel Disposal Container	4,387	14,246

[3] Reporting is conducted on a financial year basis (April to March).

Waste package	In Stock	Expected arisings
DCIC Cubical	794	247
DCIC Cylindrical	982	252
C1 Container	0	6,840
C4 Container	0	3,240
TDC	0	0
Total	**225,885**	**206,330**

Table 2 illustrates that over half of the waste packages from the UK's nuclear decommissioning programme are already in stock, and that the vast majority (>70%) of waste packages in stock and expected to arise in the future are in the form of 500-litre drums.

The 2013 DI shows zero TDC packages in stock or expected to arise. This is due to the nature of the waste these packages are expected to contain, namely DNLEU. This waste has a variety of origins from a range of fuel-cycle processes. This range of materials will have been packaged in a number of ways for storage, none of which are currently deemed suitable by RWM for disposal. It is currently envisaged that these temporary storage packages will be overpacked inside the TDC packages. However, the detail on TDC packages was not available in the 2013 DI.

It is possible to break the data down to understand when the packages in Table 2 are expected to arise. Figure 4, Figure 5 and Figure 6 plot the number of each type of waste package expected to arise over five-year periods up to 2040. It should be noted that the scales differ on these plots, as some package types are more numerous than others.

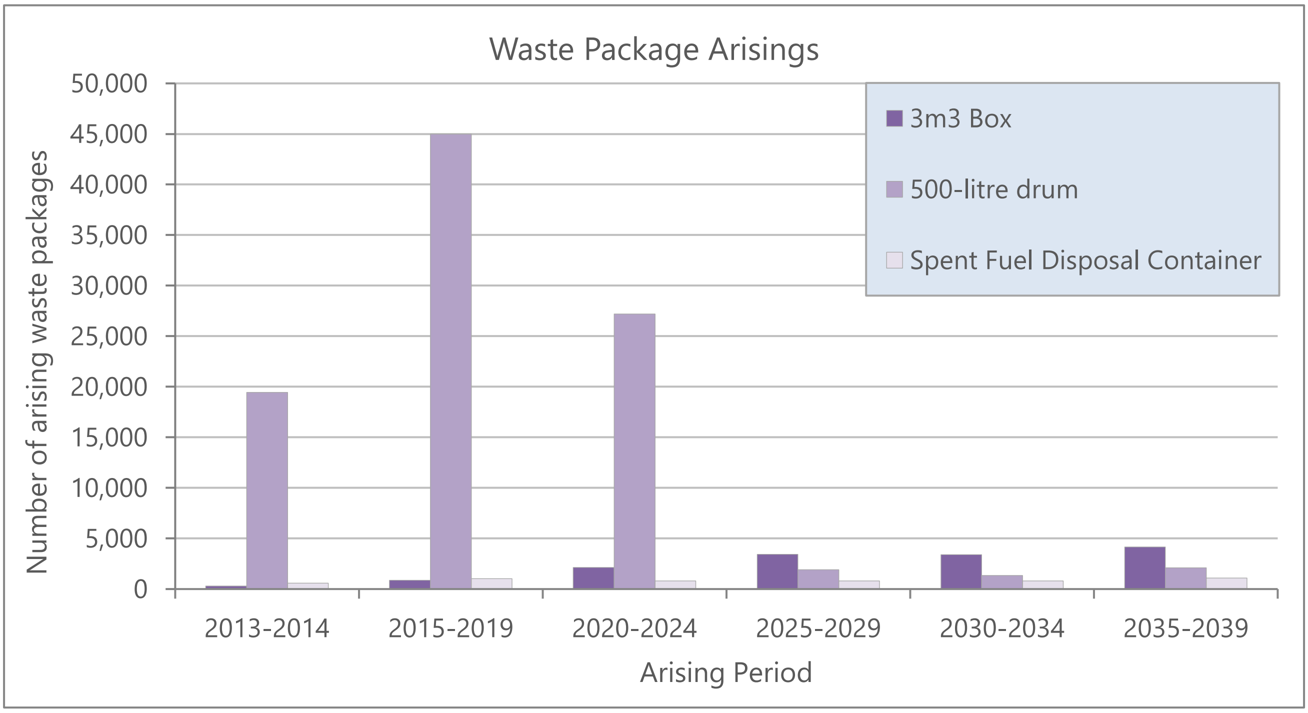

Figure 4 Waste package arisings for 3m^3 box, 500-litre drum and spent fuel disposal container from 2013 up to 2040

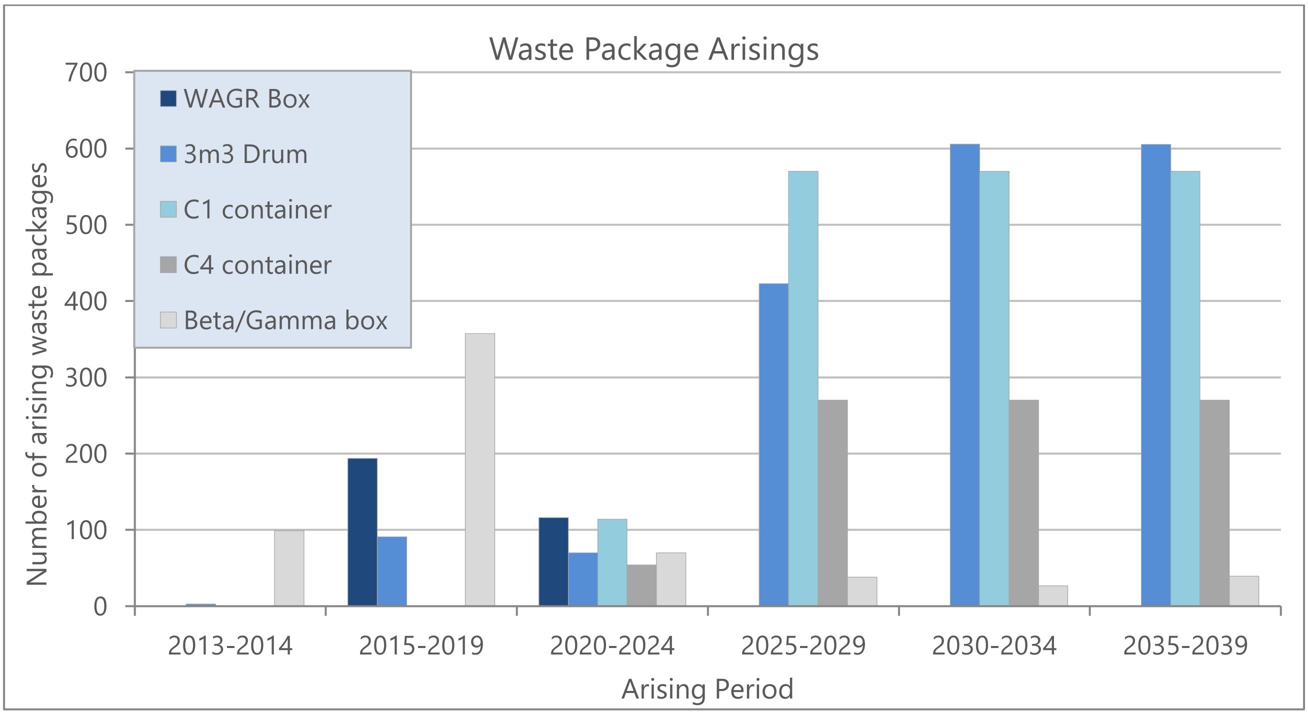

Figure 5 Waste package arisings for WAGR box, 3m^3 drum, C1 container, C4 container and beta/gamma box from 2013 up to 2040

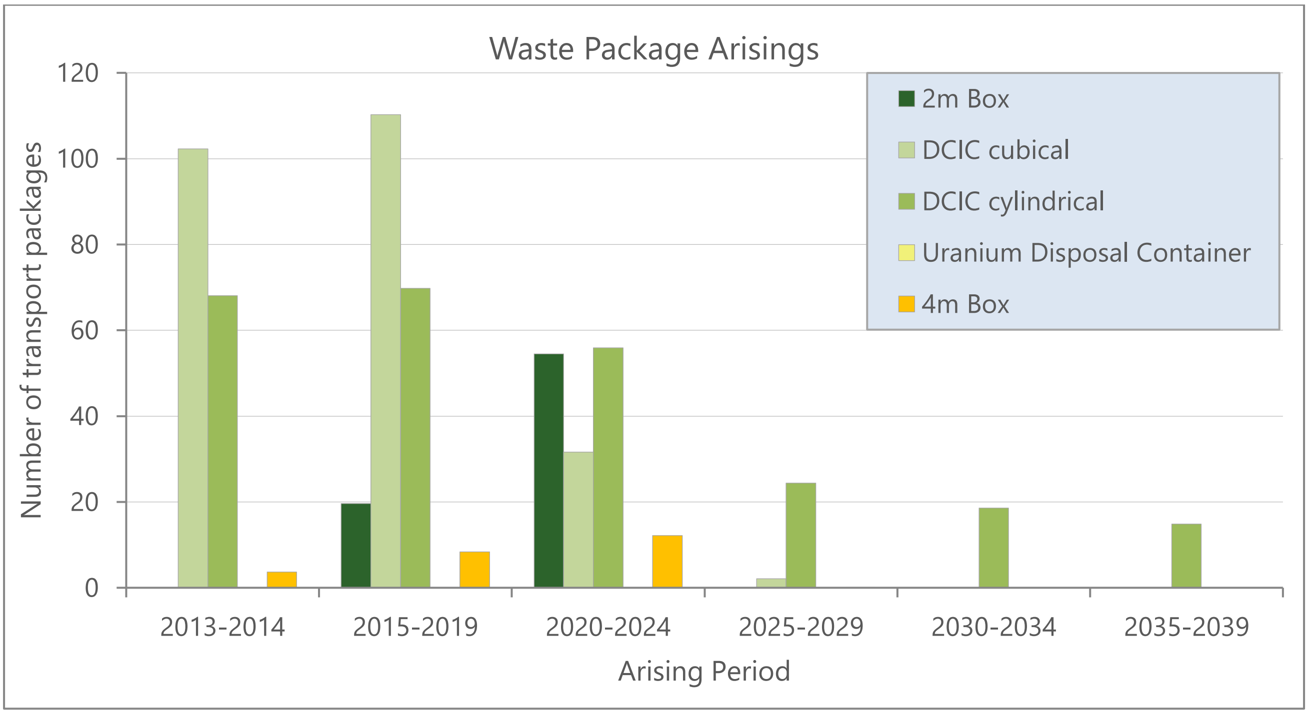

Figure 6 Waste package arisings for 2m box, DCIC cubical, DCIC cylindrical, uranium disposal container and 4m box from 2013 up to 2040

The arising plots illustrate the processes and activities that require specific waste packages, such as the decommissioning of the Sellafield legacy ponds between 2015-2019 that requires the encapsulation of sludge in thousands of 500-litre drums.

2.2.2 Transport Container Arisings

Although the GDF is expected not to be operating until 2040 or later, understanding the expected numbers and types of transport containers helps waste package and GDF designers to develop a detailed plan for how an operating system may work.

Table 3 is based upon an extrapolation of the information in Table 2 and provides an estimate for the number of transport containers required for the stated number of waste packages in stock and expected to arise up to 2040. The data in Table 3 are based on the transport packages identified in the 2013 DI and may not perfectly correlate with the transport package options illustrated in Table 1.

Table 3 Estimated number of transport packages required between 2013 – 2040 based on the 2013 Derived Inventory

Arising Period	SWTC-070	SWTC-150	SWTC-285	2m Box	4m Box	$6m^3$ box	DCTC
Stock	9,120	1,830	30,945	511	0	115	794
2013-2014	473	168	1,250	4	0	0	103
2015-2019	1,444	427	2,526	9	20	194	111
2020-2024	1,308	126	612	13	55	116	32
2025-2029	2,640	63	973	1	0	0	3
2030-2034	2,649	45	1,298	0	0	0	0
2035-2039	3,905	50	1,314	0	0	1	0

The transport container data takes into consideration waste packages that require grouping for transport (e.g. 500-litre drums are transported in groups of four in a stillage). Because SWTC and DCTC transport containers are reusable, the numbers do not equal the number of transport containers required: each container may be used many times.

2.3 Waste Materials and Arising Timeline

The decommissioning of nuclear facilities produces an unusual mix of hazardous materials that are difficult to handle and challenging to dispose of. Common construction materials such as concrete and steel are present in enormous volumes and can be highly contaminated or activated following potentially decades of presence in highly radioactive environments. Smaller volumes of exotic materials present a different challenge, with their complex properties requiring unique science and engineering solutions. Some materials can be loaded straight into containers for long-term storage, whereas other materials require multiple treatment steps before an adequate disposal solution is viable. It is this mix of complicated materials, varying volumes and extreme storage periods that make the waste packaging and transport for nuclear decommissioning such a complex challenge. A broad set of data based on accurate estimates helps organisations plan how and when they must deal with these issues, and the UKRWI and DI (IGD in the future) provides this database of information.

The materials data from the UKRWI and 2013 DI were analysed to understand the breakdown of waste by specific material groups. To provide an overview, information was collated for the top ten materials by mass for the radioactive waste in stock and expected to arise up to 2137. Most of UK's nuclear reactor plants have been based on large graphite-moderated cores with reinforced concrete pressure vessels and foundations, and this is illustrated in the breakdown in Table 4 with 'graphite', 'ferrous-based alloys' and 'cement concrete and sand' representing the vast majority of waste mass.

Table 4 List of the top ten materials by mass estimated to make up the UK's radioactive waste inventory by 2137

Top 10 materials by mass	Mass (tonnes)
Graphite	76,826
Other ferrous-based alloys	54,850
Cement Concrete and Sand	53,462
Stainless Steel	32,406
Inorganic Sludges and Flocs	21,721
Magnox alloy	6,374
Inorganic Ion Exchange	5,464
Halogenated Plastics	4,766
Aqueous Liquids	4,483
Ion Exchange Resins	3,632

Linking together the temporal elements of the inventory data illustrates how the individual material groups arise over time. Figure 7 and Figure 8 demonstrate the current estimate of when the materials listed in Table 4 will arise between 2013 and 2137. The plots highlight when individual stations approach specific points in the decommissioning stage, for example the peaks arising for graphite shown in Figure 7 correlate to the reactor core dismantling after the care and maintenance period.

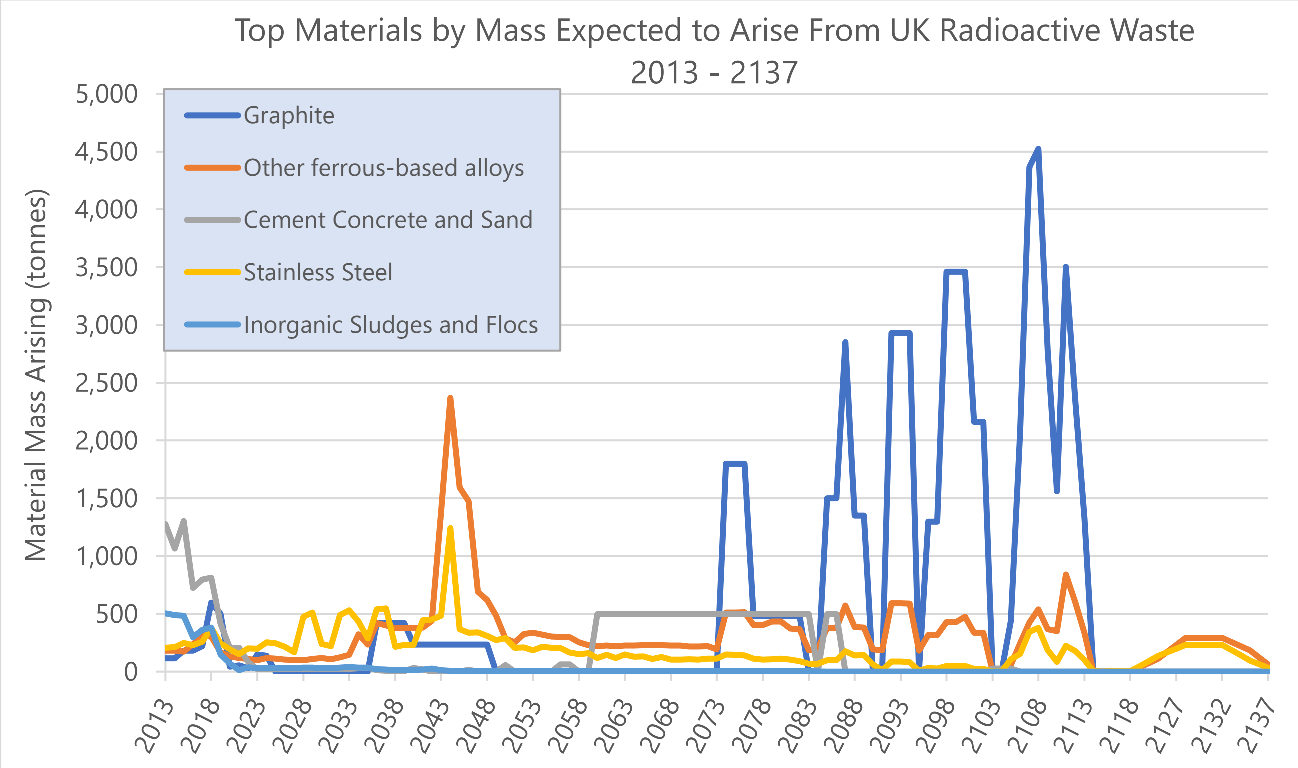

Figure 7 Annual breakdown of the top five materials by mass arising from the 2013 DI between 2013 - 2137

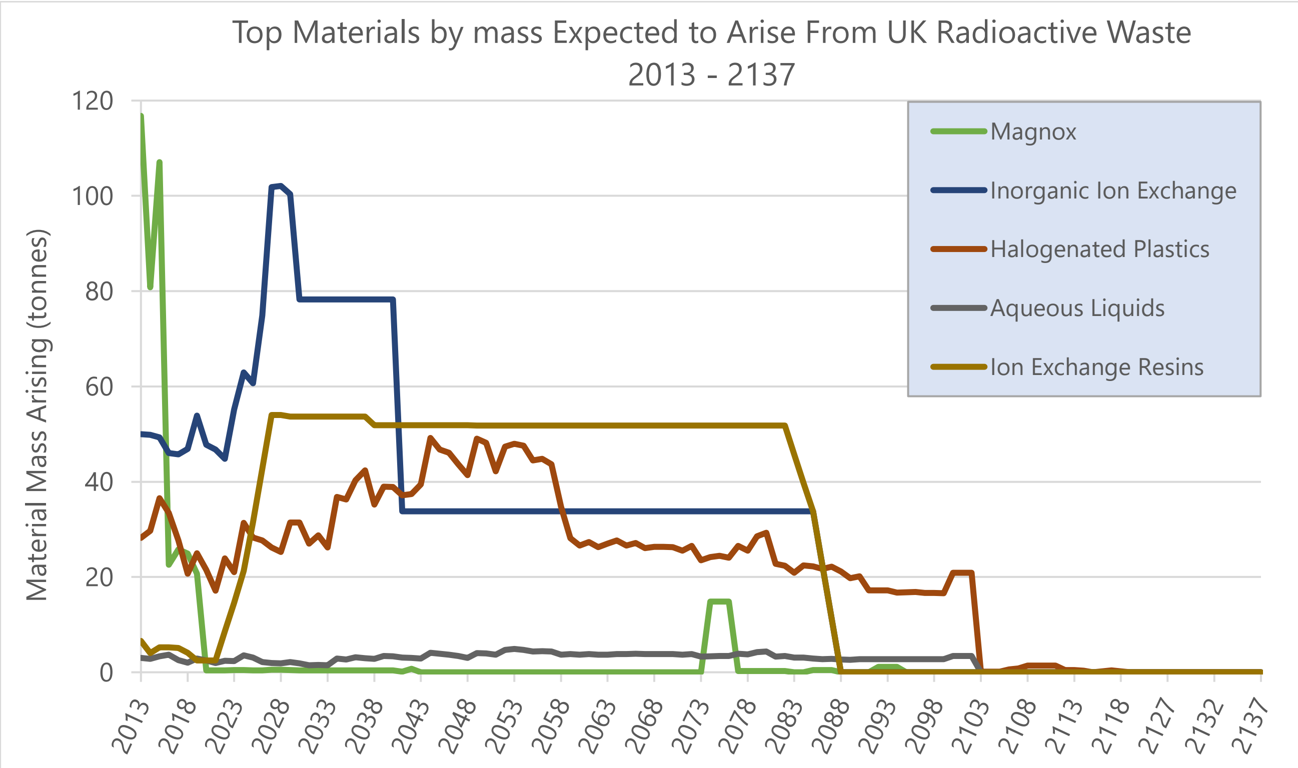

Figure 8 Annual breakdown of the second five materials by mass arising from the 2013 DI between 2013 - 2137

The data behind Figure 7 and Figure 8, as well as the many other material categories, help the NDA and SLCs to plan when numbers and types of waste containers may be required. This in turn enables RWM to design a final disposal solution with estimates of the waste packages expected to require disposal and when they are likely to arise.

2.4 Sites Affected

The 2013 DI records data on waste packages located at 27 locations, with many of these shown in Figure 2. The waste package information for each location is an important aspect to consider as the transport distances and routes vary significantly for the many sites around the country.

Each waste stream in the 2013 DI has an associated site of origin. This enables the estimation of number and types of waste packages from each individual site, which in turn aids the TSC required for moving the waste packages from each individual site to the UK's final disposal facility. The *Review of Transport Modes* chapter of this report will elaborate further on this.

Figure 9 illustrates the number of waste packages estimated to arise from the UK's nuclear decommissioning programme up to 2137 for each of the 27 locations.

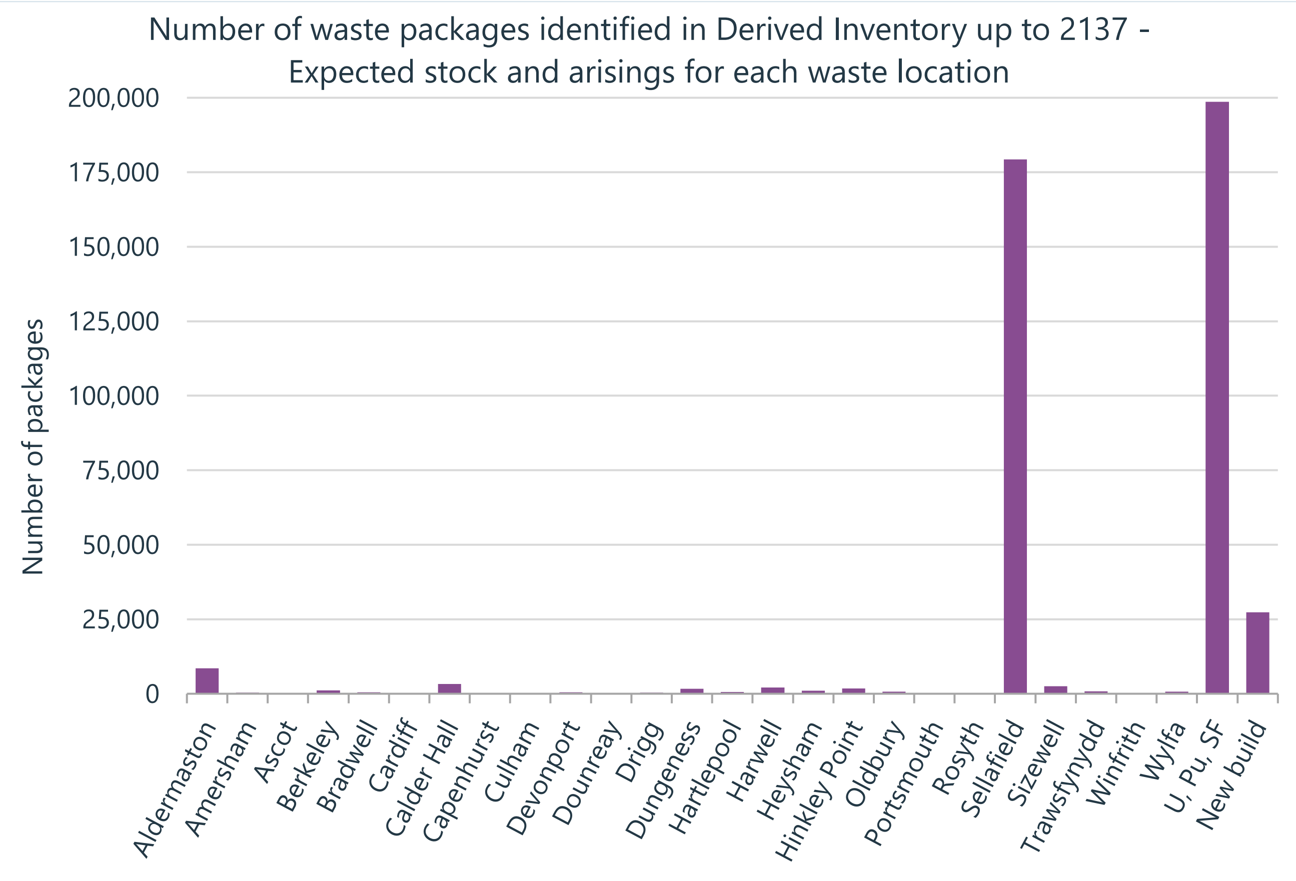

Figure 9 Plot of the number of radioactive waste packages estimated to arise up to 2137 for locations identified in the 2013 Derived Inventory

The data illustrates how the vast majority of waste packages arise from Sellafield and from the activities associated with uranium, plutonium and spent fuel (of which many will end up at Sellafield).

One of the limitations of the 2013 DI data is that some waste streams do not have an associated storage location. While the number of waste streams without a location is only a modest number, they represent around half the total number of waste packages. These location-less waste streams are represented by the 'U/Pu/SF' and 'New build' bars in Figure 9.

'U/Pu/SF' represents the packages from the uranium, plutonium and spent fuel waste streams generated from all the UK's civil nuclear programmes and most of the defence programmes. The spent fuel from Magnox, AGR, PFR, MOX, legacy ponds and submarine programmes are expected to go to Sellafield. Spent fuel from the PWR station at Sizewell B is expected to be stored on site. Highly-enriched uranium from civil and defence programmes is expected to go to Sellafield, as will plutonium from civil fuel reprocessing and Magnox depleted uranium. Depleted uranium packages from defence enrichment, and irradiated and unirradiated tails are expected to be stored at Capenhurst.

'New build' represents the estimated number of waste packages that would arise from new nuclear power stations not yet constructed. This estimate was made in 2013 when only the EPR and AP1000 designs had published expected inventory information. At the time, the expected number of new build reactors was three twin-reactor EPR stations and three twin-reactor AP1000 stations, i.e. a fleet of 12 reactors. Waste packages generated from these stations were assigned representative sites: Hinkley Point (EPR) and Oldbury (AP-1000). However, considering the new build programme in the UK has drastically changed since the 2013 DI data was recorded, it is unclear how representative these sites are.

A breakdown of the data in Figure 9 is provided in tabular form in Appendix B, which illustrates the number of different waste packages expected to arise from each of the locations.

3 Review of Transport Modes and Routes

The generic TSC includes a dose assessment. For the 2010 generic TSC, this is presented in the generic Transport System Safety Assessment (TSSA) [12]; for the 2016 generic TSC, it is presented in the generic Transport Safety Assessment (TSA) [13]. The transport modes and routes from the storage locations to the notional GDF locations are considered in these safety assessment reports.

Five transport modes that could be used to transport wastes from their storage location to a GDF have been identified:

- Air
- Road
- Rail
- Sea
- Inland waterway

The transport packages are too heavy for air transport to be practical. It would also require a GDF to be sited near a suitable airport.

At present transport by inland waterway is not considered. There is no history of transporting radioactive material by inland waterway and it would be a viable option only if a GDF were located near a suitable waterway. Nevertheless, this transport mode may be considered if a suitable GDF location is identified.

Transport packages need to be transported from each storage location to the GDF. In each case, the choice of route and transport mode will be made based on the transport infrastructure. RWM has adopted a transport safety strategy [14] in which transport by rail is to be used in preference to road transport wherever possible. Thus, if a storage location has an on-site railhead, packages will be transported directly from the storage location to the GDF by rail. For other storage locations, packages will be transported to a suitable transhipment site for onward transport to the GDF by rail.

A careful analysis of operator doses needs to justify use of sea transport. Apart from a very small number of packages stored at naval dockyards, the transport packages would need to be transported from their storage locations to a despatch port. If a storage location has an onsite railhead, transport to the despatch port would be by rail, otherwise it would be by road. Using road transport direct to the port avoids the doses that would arise from transhipment from road to rail at an off-site railhead. From the despatch port, packages would be transported by sea to a port serving the GDF from where they would be transported to the GDF by rail.

As far as radiological safety is concerned, sea transport is justified only if there is a significant saving in operator dose relative to the equivalent land journey. In addition, there may be economic drawbacks to using sea transport – construction of the necessary infrastructure may not be justified if only a small fraction of the transport packages is likely to be conveyed by ship. In the generic TSSA, operator doses were calculated according to four transport scenarios two of which involved sea transport (a different dose assessment methodology was used in the generic TSA). The rules adopted in the sea transport scenarios illustrate the issues that need to be considered. In these scenarios, transport by sea is only selected if:

- The distance travelled by sea is greater than 75km.
- The saving in land transport distance is greater than 50km.
- There are at least 9 packages requiring transport from the storage location.

Transport by road, rail and sea are considered further in the following subsections. This information captures the limitations of UK infrastructure and should be considered by stakeholders in design specifications and/or planning exercises to ensure UK infrastructure can handle the intended movements.

3.1 Road transport

The core provision for road transport is provided by the Strategic Road Network (SRN) in England and the Trunk Road Networks (TRNs) in Scotland and Wales. These are illustrated in Figure 10.

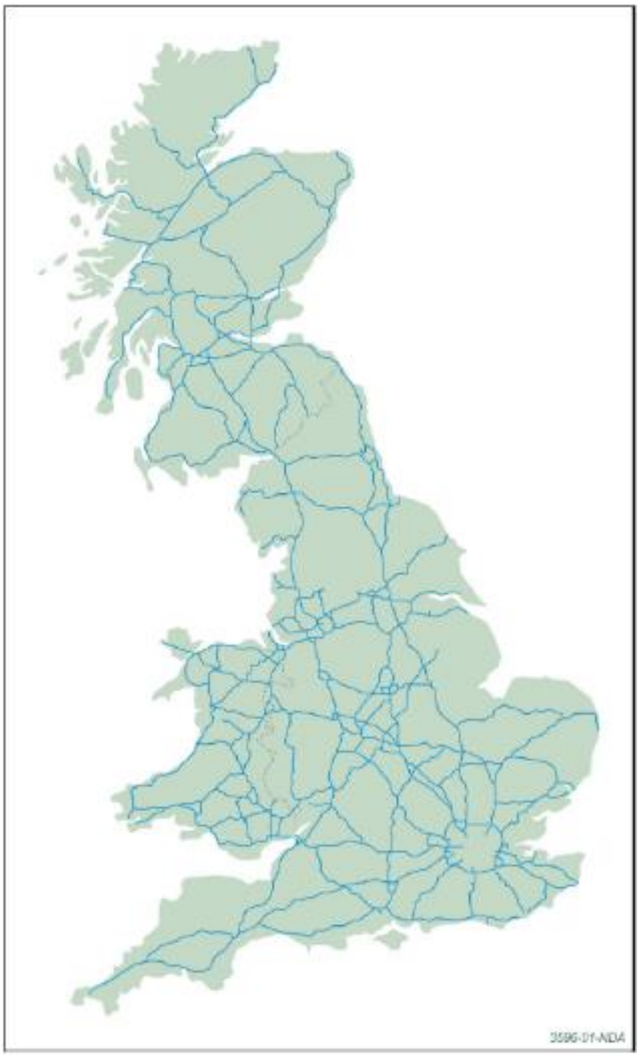

Figure 10 Strategic Road Network in England and the Trunk Road Networks in Scotland and Wales.

The GDF will be developed to have a direct connection to the SRN or TRN. Where necessary, a connecting route from a waste storage location to the SRN or TRN will be derived using local roads. Factors affecting the choice of route include the impact on settlements and the capability of the roads to bear heavy loads.

Within the UK, regulations for road vehicles are set out in the Road Vehicle (Construction and Use) (C&U) regulations [15]. With a few exceptions, these regulations allow:

- A maximum gross weight of 44t, requiring a six-axle articulated vehicle. A five-axle articulated vehicle has a gross weight limit of 40t. The weight limit for non-drive axles is 10t with a weight limit of up to 11.5t for a drive axle with road-friendly (pneumatic) suspension.
- A maximum individual truck length of 12m.
- A maximum length for an articulated vehicle of 16.5m.
- A maximum length for a road train (a rigid road vehicle pulling a trailer) of 18.5m.
- In all cases, a maximum width of 2.55m.

Transport of abnormal indivisible loads is permitted through the Special Types (General Order) (STGO) regulations [16]. An abnormal indivisible load is defined as one that cannot without undue expense or damage be divided into two or more loads that can be carried on a vehicle complying with the C&U regulations on account of its dimensions or weight. Transport packages that exceed the C&U regulations are abnormal indivisible loads.

There are three categories of STGO vehicle:

- STGO Category 1 has a gross weight limit of 50t with the same axle-weight limits as set out in the C&U regulations.
- STGO Category 2 has a gross weight limit of 80t with a maximum axle weight on twin tyres of 12.5t.
- STGO Category 3 has a gross weight limit of 150t with a maximum axle weight on twin tyres of 16.5t.

The maximum width permitted for an STGO vehicle is 6.1m and the maximum length is 30m.

Note that if a vehicle complies with the C&U limits on dimensions and axle weights, then it can be used as a standard Heavy Goods Vehicle (HGV) for transport packages with a mass up to 30t. Heavier packages would need to be transported as an STGO Category 1 vehicle.

Many of the transport packages in the 2013 DI are such that the 50t mass limit for an STGO Category 1 would be exceeded. In the generic TSC, RWM is planning to use a combination of STGO Category 1 and Category 3 vehicles recognising that the former can be used as an HGV for sufficiently light transport packages.

3.2 Rail transport

The UK rail network is illustrated in Figure 11.

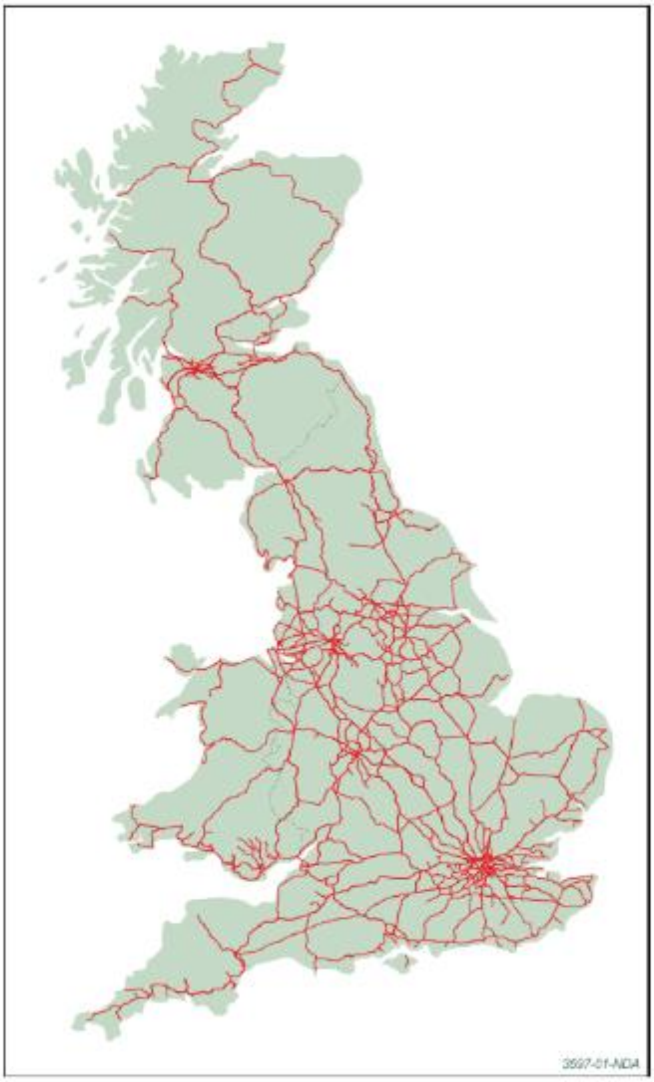

Figure 11 The UK rail network.

Two factors affect the suitability of sections of the rail network for transport of a given vehicle: route availability and loading gauge. These are considered in the subsequent sections.

3.2.1 Route Availability

The route availability of a section of track characterises the ability of the track to bear loads. Route availabilities are given as RA numbers which determine the axle weight of a vehicle. Route availabilities are listed in Table 5.

Table 5 Route availabilities of the UK rail network

Route Availability	Maximum Axle Weight (tonnes)
RA1	13.97
RA2	15.24

Route Availability	Maximum Axle Weight (tonnes)
RA3	16.51
RA4	17.78
RA5	19.05
RA6	20.32
RA7	21.59
RA8	22.86
RA9	24.13
RA10	25.40

The load on a wagon determines the route availability of the vehicle. The axle weight is derived by dividing the gross weight of the vehicle by the number of axles, the route availability is then worked out from the table by taking the nearest defining axle weight less than or equal to the vehicle axle weight. Thus, a heavy load carried on a wagon will lead to a higher route availability requirement than a lighter load carried on the same wagon.

On many rail routes, the route availability is determined by particular structures such as bridges or aqueducts. Network Rail, as the infrastructure provider, is able to authorise vehicles with a higher axle weight than permitted by the route availability to use particular sections of track. This may be by the imposition of a speed limit or the use of barrier wagons to give wider separation to the load-bearing axles on a train.

3.2.2 Loading Gauge

The loading gauge of a rail route determines the size of load that can be carried. Loading gauges are given as W values as follows:

- W6a – the basic loading gauge available over most of the UK rail network.
- W8 – allows a standard 8ft 6in (2.6m) high container to be carried on standard wagons.
- W9 – allows 9ft 6in (2.9m) "Hi-Cube" containers to be carried on low deck-height wagons.
- W10 – allows "Hi-Cube" containers to be carried on standard wagons.
- W11 – a little used loading gauge that allows 9ft 6in (2.9m) by 8ft 4in (2.55m) containers to be carried.
- W12 – the widest loading gauge allowing 8ft 6in (2.6m) wide refrigerated containers to be carried.

W12 is the recommended loading gauge for all new rail structures. Note that the loading gauge is often determined by structures such as bridges and tunnels. Accordingly, the loading gauge can be increased but only by the expensive process of re-building such structures or lowering the track.

3.3 Sea transport

Sea transport relies on the use of suitably equipped ports. A medium-sized ship would suffice for moving multiple transport packages. Pacific Nuclear Transport Limited (PNTL) has over 40 years' experience of transporting loads such as spent fuel, plutonium and vitrified HLW by sea. At present, PNTL operates a fleet of three ships that have a length overall of some 104m, a breadth of 17.15m and a draft of 6.75m. Ships such as this can be accommodated at most commercial ports in the UK.

4 Compliance with Transport Regulations

In the UK, the regulation of transport of radioactive materials is by compliance with the Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations (CDG) [17] for transport by road and rail and with the International Maritime Dangerous Goods (IMDG) Code [18] for transport by sea. Through international agreements, CDG and the IMDG Code implement the IAEA Regulations for the Safe Transport of Radioactive Material [19], which are referred to in this report as the IAEA Transport Regulations. The IAEA Transport Regulations were updated in 2018 [20]. Although this update was a relatively minor revision, the changes did introduce several additions that are relevant to DPCs. A clause was added stating:

> For *packages* intended to be used for *shipment* after storage, it shall be ensured that all *packaging* components and *radioactive contents* have been maintained during storage in a manner such that all the requirements specified in the relevant provisions of these Regulations and in the applicable certificates of *approval* have been fulfilled.

Another sentence added stated:

> The *design* of the *package* shall take into account ageing mechanisms.

where italicised terms have the meaning defined in the regulations. These 2018 additions are relevant to DPCs, however, as the this version has not yet been incorporated into UK legislation and the 2016 update to RWM's generic TSC was written against the 2012 edition, this report will refer to the 2012 edition of the regulations, acknowledging the few additions that are relevant to DPCs where applicable.

The IAEA Transport Regulations cover all issues relating to the transport of radioactive material. Thus, issues such as labelling of transport packages and the need for approved Quality Assurance programmes to be in place. However, as the title of the regulations imply, the bulk of the regulations address the safety of transport operations and it is this aspect of the regulations that will be considered in this report.

As stated in the IAEA Transport Regulations (para 104), the objective of the regulations are to "protect persons, property and the environment from the effects of radiation in the transport of radioactive material". This is done by addressing:

- The containment of radioactive contents.
- The control of external radiation.
- The prevention of criticality.
- The prevention of damage caused by heat.

An aspect of addressing these requirements is the application of a graded approach to the performance requirements, in particular control of the acceptable contents, of a transport package according to the hazard posed by the contents. The regulations define nine types of transport package. In order to meet its obligation of providing a safe and publicly acceptable solution for the geological disposal of the UK's HAW; RWM has a design that utilises Industrial Packages (IPs) and Type B packages [6].

IPs are appropriate for waste that can be classified as Low Specific Activity (LSA) material or Surface Contaminated Objects (SCO). The regulations define three categories of LSA: LSA-I, LSA-II and LSA-III. Similarly, there are three categories of SCO: SCO-I, SCO-II and SCO-III[4]. There are three types of IP: IP-1, IP-2 and IP-3 each with permissible contents defined in the regulations. RWM is proposing to use IP-2 packages, which may be used to transport SCO-II and LSA-II not under exclusive use or LSA-III under exclusive use. Exclusive use is defined in the regulations as:

> the sole use, by a single *consignor*, of a *conveyance* or of a *large freight container*, in respect of which all initial, intermediate and final loading and unloading and *shipment* are carried out in accordance with the directions of the *consignor* or *consignee*, where so required by these Regulations.

Type B packages are appropriate to the most hazardous waste.

In many cases, IP packages incorporate shielding sufficient for the waste package to meet the limits imposed on external dose rate. In such cases, the waste package qualifies as a transport package. Radioactive materials with higher specific activities are often packaged in waste packages that do not provide sufficient shielding to permit transport in their own right. Such packages would be transported in a separate transport container that would provide the required performance to meet the limits imposed by the IAEA Transport Regulations. Such transport containers are expected to be re-usable. The combination of the transport container and the waste package within it needs to meet the requirements of a Type B transport package.

The performance requirements specified in the IAEA Transport Regulations result in safety assurance being vested in the transport package design. Thus, there is minimal reliance on human intervention of operation controls.

The IAEA Transport Regulations consider routine (incident free), normal (including minor mishaps) and accident conditions of transport. The regulations require that transport packages must be able to withstand physical challenges that increase in severity according to the hazard of the package contents.

RWM recognises that it will be many years before a GDF will be available. In order to permit waste producers to produce waste packages, RWM has a disposability assessment process [21] whereby a waste packaging proposal can be assessed and, if appropriate, be endorsed through issue of a Letter of Compliance (LoC). Issue of an LoC is a formal indication by RWM that the packaging proposal will result in waste packages that meet the requirements of UK legislation and comply with the generic DSSC and are expected to remain compliant as these requirements evolve. The disposability assessment process requires that a TSA should be undertaken. Fundamental to a Transport Safety Assessment (TSA) is a demonstration that the packaging proposal will result in transport packages that are compliant with the requirements of the IAEA Transport Regulations. This compliance is assessed by consideration of a number of issues most of which apply to both IP-2 and Type B

[4] SCO-III was added in the 2018 revision, however the regulations describe the category as "a large solid object which, because of its size, cannot be transported in a type of package described in these Regulations…".

packages but some of which are specific to each package type. These are considered in the following sub-sections.

4.1 Issues applying to IP-2 and Type B Transport Packages

4.1.1 External dose rates

The IAEA Transport Regulations define two sets of limits for the external dose rate from a transport package depending on whether or not the packages are transported under exclusive use.

For transport not under exclusive use, the regulations lead to limits of:

- 0.1 mSv/h [5] at 1m from the surface of the transport package.
- 2 mSv/h on the external surface of the transport package.

For transport under exclusive use, the limits are:

- 0.1 mSv/h at 2m from the surface of the vehicle.
- 2 mSv/h on the external surface of the vehicle.
- 10 mSv/h on the external surface of the transport package with additional controls if the dose rate exceeds 2 mSv/h.

Owing to the size and shape of packages suitable for transporting wastes, the dose rate limits furthest from the package are the most restrictive. As can be seen, the limits for transport not under exclusive use are more onerous than the limits for transport under exclusive use.

The dose rate at 1m from the surface of the transport package is one of the outputs of DIQuest. Thus, if this dose rate is less than 0.1 mSv/hr at the time of transport, the package will comply with the limits in the regulations for transport not under exclusive use. The package will also comply with the less onerous limits for transport under exclusive use. Should the external dose rate exceed the limit for transport not under exclusive use, it will be necessary to derive the dose rate at 2m from the surface of the vehicle, which is not a DIQuest output, in order to check compliance with limits for transport under exclusive use.

4.1.2 Containment under normal conditions of transport

Under normal conditions of transport, the IAEA Transport Regulations place a limit on the release of radioactive material from a Type B package of 10^{-6} A_2/h [6].

The Design Authority for a Type B transport package produces a Design Safety Report (DSR) for the package. This will be assessed by a Competent Authority and, if appropriate, approved by issue of a Certificate of Approval (CoA). For IP-2 packages, approval of the DSR is by the Design Authority subject to audit by a Competent Authority. In such cases, a document similar to a CoA will be issued. A

[5] mSv/h = millisievert per hour; a Standard International unit to measure health effect of ionising radiation on the human body.

[6] A_2; an activity value (expressed in TBq) defined in the IAEA Transport Regulations and used to define the activity limits for the requirements expressed in those regulations.

component of the DSR and subsequent CoA is the contents activity limits documentation. This specifies, for each radionuclide, the maximum activity permitted in order to meet several of the criteria specified in the IAEA Transport Regulations.

Compliance with the containment limit under normal conditions of transport is assessed by comparison with the package contents limits documentation.

An example of how contents limits are derived is given by the SWTC family of reusable transport containers being developed by RWM. SWTCs are designed to carry a number of waste package designs with the combined package being transported as a Type B transport package.

In order to be released from the SWTC, radionuclides first have to be released from the waste package into the SWTC cavity and thence through the lid seal of the SWTC to the environment. Radionuclides can be released into the SWTC cavity by:

- Release of activity from the surface of the waste package.
- Release of gaseous activity from the waste package.
- Release of particulates from the waste package.

The IAEA Transport Regulations specify limits on surface contamination (see Section 4.1.4). If the waste package has the maximum permitted surface contamination the amount of activity on the surface of the package is trivial and will not challenge the containment limit even if all of the activity were released in an instant. For the SWTC, for the release of gaseous radionuclides, a calculation is undertaken to derive the activity present in the SWTC cavity that will give a release of 10^{-6} A_2/h through the lid seal at the maximum normal operating pressure of the SWTC. For each radionuclide, the fraction that the release of activity from the waste package is of the relevant activity limit is calculated. These fractions are summed over all the gaseous radionuclides and, if the resulting value is less than one, the transport package complies with the containment limit for transport under normal conditions of transport for non-fissile packages. A series of tests for fissile packages to demonstrate that particulates will not be released under normal conditions of transport are also required.

4.1.3 Heat output

The IAEA Transport Regulations do not impose limits on the heat output from waste packages. However, the regulations do specify limits on the surface temperature of transport packages. If a package is transported not under exclusive use, the surface temperature must not exceed 50°C; under exclusive use it must not exceed 85°C.

The easiest way of demonstrating compliance with the surface temperature limit is to use the results of thermal modelling undertaken to determine what waste package heat output is consistent with the surface temperature limits. As an example, thermal modelling has shown that the more onerous limit for the surface temperature of the SWTC not exceeding 50°C will be met if the heat output of the SWTC contents (a waste package or a collection of waste packages) is limited to 400W. If a waste package is compliant with this limit, it will certainly be compliant with the less onerous limit for transport under exclusive use.

Waste package heat outputs at the time of transport are available from DIQuest. Values are given for each radionuclide with the package heat output being the sum over all radionuclides. This permits compliance with the heat output limit to be assessed as well as identification of radionuclides that are significant contributors to the heat output.

4.1.4 Surface contamination

The IAEA Transport Regulations impose limits on the surface contamination of the waste package when averaged over 300cm^2 or any part of the surface of the package. These limits are:

- 4 Bq/cm^2 [7]for beta, gamma and low-toxicity alpha emitters.
- 0.4 Bq/cm^2 for all other alpha emitters.

Low-toxicity alpha emitters are defined in the regulations as

> Natural uranium, depleted uranium, natural thorium, U235, U238, Th232, Th228 and Th230 when contained in ores or physical and chemical concentrates; or alpha emitters with a half-life of less than 10 days.

Compliance with the limits is best checked by direct monitoring of the transport package.

4.1.5 Criticality

A key principle underpinning the IAEA Transport Regulations is that criticality during transport shall be prevented.

The regulations define fissile material as any material containing fissile nuclides, which are defined as U233, U235, Pu239 and Pu241. A graded set of limits and conditions are set out in the regulations for addressing criticality safety.

The most restrictive limits and conditions state when the contents of a transport package may be excluded from consideration as fissile material. These are set out in Section 4.1.5.1.

A slightly less restrictive set of limits and conditions are referred to as fissile exceptions. If a transport package satisfies a fissile exception there is no requirement for further consideration of criticality safety. Fissile exceptions are discussed in Section 4.1.5.2.

The least restrictive set of limits and conditions set out conditions under which fissile material can be transported without approval by a Competent Authority. These rules impose conditions on the quantity of fissile nuclides and the masses of neutron moderators such that criticality safety can be assured through accumulation control via definition of a Criticality Safety Index (CSI). These CSI limits are set out in Section 4.1.5.3.

Transport packages that do not meet any of the limits and conditions set out above will need to be transported in 'fissile' packages and Competent Authority approval will be required for the package to contain fissile material. In such cases, it is necessary to demonstrate that criticality during transport is prevented by specifying a Safe Fissile Mass (SFM) and showing that the package contents comply with the SFM. An SFM can be derived from generic Criticality Safety Assessments (CSAs) for a particular package design or type of fissile material. SFMs given in generic CSAs are often derived on the basis of assumed package contents. For instance, it may be assumed that the fissile material is uniformly

[7] Bq/cm^2 = becquerel per square centimetre; a unit of activity per unit area.

distributed through the wasteform or limits on the quantities of reflectors or moderators may be assumed. If the package contents cannot be demonstrated to comply with the assumptions underpinning the generic CSAs, it will be necessary to derive an SFM in a package-specific CSA.

It is noted that a number of requirements for Type A fissile and IP fissile packages are more onerous for an equivalent non-fissile package, and are equivalent to the requirements for a Type B package i.e. applicants much show compliance with the IAEA mechanical and thermal tests under accident conditions of transport.

4.1.5.1 Fissile exclusions

The most restrictive limits and conditions state when the contents of a transport package may be excluded from consideration as fissile material. The conditions are:

- If the material is unirradiated natural or depleted uranium.
- If the material is natural or depleted uranium that has been irradiated in thermal reactors only.
- If the mass of fissile nuclides is less than 0.25g.
- Any combination of the above three conditions.

4.1.5.2 Fissile exceptions

A slightly less restrictive set of limits and conditions are known as fissile exceptions. If a transport package satisfies a fissile exception there is no requirement for further consideration of criticality safety. The fissile exceptions are:

- Uranium with a maximum of 1% U235 by mass with a total plutonium and U233 content not exceeding 1% of the mass of U235 provided that the fissile nuclides are distributed essentially homogeneously throughout the material. If U235 is present in metallic, oxide or carbide forms, it shall not form a lattice arrangement.
- Liquid solutions of uranyl nitrate containing U235 to a maximum of 2% by mass with a total plutonium and U233 content not exceeding 0.002% of the mass of uranium and with a minimum nitrogen to uranium atomic ratio of 2.
- Uranium with a maximum of 5% U235 by mass provided:
    1. There is no more than 3.5g U235 per package.
    2. The total plutonium and U233 content does not exceed 1% of the mass of U235 per package.
    3. The package is transported in a consignment with no more than 45g of fissile material.
- Fissile nuclides with a total mass not greater than 2g per package provided the package is transported in a consignment with no more than 15g of fissile nuclides.
- Fissile nuclides with a total mass not greater than 45g provided the package is transported under exclusive use on a conveyance with no more than 45g fissile nuclides.

- Fissile material that can be shown to remain subcritical without reliance on accumulation control and will remain subcritical during normal and accident conditions of transport and water ingress leading to maximal neutron multiplication. This fissile exception requires Competent Authority approval.

4.1.5.3 CSI limits

The least restrictive set of limits and conditions set out conditions under which fissile material can be transported without approval by a Competent Authority. These rules impose conditions on the quantity of fissile nuclides and the masses of neutron moderators such that criticality safety can be assured through accumulation control via definition of a CSI. Three cases are defined, in each case the fissile material can be in any form:

- Provided:
    1. The smallest dimension of the package is not less than 10cm.
    2. The CSI of any package does not exceed 10 where the CSI is calculated by the formula:
       CSI = 50 x 5 x {mass(U235)/Z + mass(other fissile nuclides)/280}
       where the value of Z is to be taken from Table 6.

Table 6 Values of Z to be used in calculation of CSIs

Uranium Enrichment	Z
Up to 1.5%	2,200
Up to 5%	850
Up to 10%	660
Up to 20%	580
Up to 100%	450
If a package contains uranium with varying ^{235}U enrichments, the value of Z corresponding to the highest enrichment is to be used.	

- Provided:
    1. The smallest dimension of the package is not less than 30cm.
    2. Under normal conditions of transport, the package maintains the smallest dimension of 30cm, prevents the entry of a 10cm cube and retains the fissile material.
    3. The CSI of any package does not exceed 10 where the CSI is calculated using the formula:
       CSI = 50 x 2 x {mass(U235)/Z + mass(other fissile nuclides)/280}
       where the value of Z is taken from Table 6.

- Provided:
    1. The smallest dimension of the package is not less than 10cm.
    2. Under normal conditions of transport, the package maintains the smallest dimension of 10cm, prevents the entry of a 10cm cube and retains the fissile material.
    3. The maximum mass of fissile nuclides in any package does not exceed 15g.
    4. The CSI of any package does not exceed 10 where the CSI is calculated using the formula:
       CSI = 50 x 2 x {mass(U235)/450 + mass(other fissile nuclides)/280}.

In all the formulae given above for evaluating a CSI, the masses are the mass in grams per package and plutonium may be in any isotopic composition provided that the mass of Pu241 is less than that of Pu240.

In all three cases above, the condition on neutron moderators is:

- The total mass of beryllium, hydrogenous material enriched in deuterium, graphite and other allotropic forms of carbon in an individual package shall not be greater than the mass of fissile materials in the package except where their total concentration does not exceed 1g in any 1,000g of material. Beryllium incorporated in copper alloys up to 4% in weight of the alloy does not need to be considered.

4.1.6 Other dangerous properties

The IAEA Transport Regulations require that as well as the radioactive and fissile contents of a package, any other dangerous properties such as explosiveness, flammability, pyrophoricity, chemical toxicity and corrosiveness shall be taken into account in compliance with the transport regulations for dangerous goods in the countries through or into which the materials will be transported.

In the case of transport of radioactive wastes from their storage locations to a GDF, the transport will be entirely within the UK. As mentioned at the beginning of Section 4, the transport of dangerous goods by road and rail in the UK is regulated via the CDG regulations. In turn, these regulations implement the requirements of the European Agreement for the International Carriage of Dangerous Goods by Road [22], commonly referred to as ADR, and the Regulations concerning the International Carriage of Dangerous Goods by Rail [23], commonly referred to as RID. Thus, the contents of a transport package need to be assessed against these regulations.

The CDG regulations state that when a material has both radioactive and chemotoxic hazards, the hazard posed by the radioactive material takes precedence and the chemotoxic hazard presents a 'subsidiary risk' provided the requirements of the IAEA Transport Regulations are met.

Risks of explosivity and flammability can be assessed on the basis of release rates of flammable gases (expected to be hydrogen and methane) from waste forms. Such release rates are likely to be derived from modelling.

Risks from corrosive or pyrophoric materials, if any are identified, should be addressed by assessment of the conditioning of the wasteform.

4.2 Issues applying only to Type B Transport Packages

4.2.1 Containment under accident conditions of transport

Under accident conditions of transport, the IAEA Transport Regulations place limits on the release of activity from a transport package in the week following an accident of 10 A_2 for Kr85 and 1 A_2 for all other radionuclides.

In considering containment under accident conditions of transport, release of activity from a waste package is determined by application of release fractions that state what fraction of the radioactive contents of the package will be released in an accident. Typically, there will be a single release fraction applicable to all radionuclides for release in impact accidents; for fire accidents different release fractions are likely to be given according to the volatility of the radionuclides. The combined release from both impact and fire accidents need to be taken into consideration. For waste packages transported in reusable transport containers, any release of activity from the waste package will be into the transport container cavity. The transport container will provide a degree of protection against release of activity to the environment that can be included in the assessment of the transport package performance.

As was discussed above in Section 4.1.2, a Type B transport package will be approved for use through issue of a CoA, provided that the design is compliant with the series of tests in the IAEA Transport Regulations.

4.2.2 Pressurisation under normal conditions of transport

The IAEA Transport Regulations require the maximum normal operating pressure of a Type B package to be less than 700kPa [8] (gauge).

Compliance with this limit is assessed by consideration of bulk gas release rates from the wasteform. Such release rates are likely to be derived from modelling.

4.3 Issues applying only to IP-2 Transport Packages

4.3.1 Classification as LSA material or SCO

The classifications of LSA material or SCO[9] are set out in the IAEA Transport Regulations as follows:

- LSA-I
    1. Uranium and thorium ores and concentrates of such ores and other ores containing naturally occurring radionuclides.
    2. Natural uranium, depleted uranium, natural thorium or their compounds or mixtures that are unirradiated and in solid or liquid form

[8] kPa = kilopascals; a unit of pressure

[9] As noted earlier in the document, a third SCO category was introduced in the 2018 update of the regulations, however as this version has not yet been incorporated into UK legislation it will not be considered here.

3. Radioactive material for which the A_2 value is unlimited. Any fissile material must be excepted.
4. Other radioactive material in which the activity is distributed throughout, and the estimated average specific activity does not exceed 30 times the activity concentration. Any fissile material must be excepted.

- LSA-II
  1. Water with a tritium concentration of up to 0.8 TBq/l.
  2. Other material in which the activity is distributed throughout, and the estimated average specific activity does not exceed 10^{-4} A_2/g for solids and gases and 10^{-5} A_2/g for liquids.
- LSA-III
  Solids, excluding powders, in which:
  1. The radioactive material is distributed throughout a solid or a collection of solid objects or is essentially uniformly distributed in a solid compact binding agent (e.g. concrete).
  2. The radioactive material is relatively insoluble or is intrinsically contained in a relatively soluble matrix so that, even under loss of packaging, the loss of radioactive material per package by leaching when placed in water for 7 days would not exceed $0.1A_2$ [10].
  3. The estimated average specific activity of the solid, excluding any shielding materials, does not exceed 2×10^{-3} A_2/g
- SCO-I
  A solid object on which:
  1. The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 0.4 Bq/cm^2 for all other alpha emitters.
  2. The fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 4×10^4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 4,000 Bq/cm^2 for all other alpha emitters.
  3. The non-fixed contamination plus the fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 4×10^4 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 4,000 Bq/cm^2 for all other alpha emitters.
- SCO-II
  A solid object on which either the fixed or non-fixed contamination on the surface exceeds the

[10] The 2018 edition of the IAEA Transport Regulations does not include the requirement to meet a limit on leaching of activity.

limits specified for SCO-I and on which:

1. The non-fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 400 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 40 Bq/cm^2 for all other alpha emitters.
2. The fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 8×10^5 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 8×10^4 Bq/cm^2 for all other alpha emitters.
3. The non-fixed contamination plus the fixed contamination on the accessible surface averaged over 300 cm^2 (or the whole surface if less than 300 cm^2) does not exceed 8×10^5 Bq/cm^2 for beta and gamma emitters and low toxicity alpha emitters or 8×10^4 Bq/cm^2 for all other alpha emitters.

Package activities and specific activities are available from DIQuest, which will help check compliance with the LSA limits.

4.3.2 Unshielded dose rate

For material packaged in an IP, the IAEA Transport Regulations place a limit of 10 mSv/h at 3m from the unshielded material in the package.

DIQuest provides a range of standard external dose rates from transport packages and waste packages, including the dose rate at 3m from the surface of a waste package. However, this cannot be used directly for assessment of compliance with the limit imposed by the transport regulations, as this dose rate is calculated including the shielding provided by the waste container. Experienced DIQuest users are able to make estimates for the unshielded waste by creating analogous waste packages with minimal shielding from the waste container.

4.3.3 Conveyance limits for LSA material and SCO

The IAEA Transport Regulations impose limits on the conveyance of LSA and SCO. These are set out in Table 7.

Table 7 Conveyance limits for LSA material and SCO in IP transport packages

Nature of Material	Conveyance limit for transport other than by inland waterway	Conveyance limit for transport by inland waterway
LSA-I	No limit	No limit
LSA-II and LSA-III non-combustible solids	No limit	100 A_2

Nature of Material	Conveyance limit for transport other than by inland waterway	Conveyance limit for transport by inland waterway
LSA-II and LSA-III combustible solids and all liquids and gases	100 A_2	10 A_2
SCO	100 A_2	10 A_2

As noted in Section 3, it is unlikely that a package will be transported to a GDF using inland waterways. Thus, compliance with the more restrictive limits given in the third column of the table are unlikely to be necessary.

The total A_2 content of a transport package, needed to check compliance, is available from DIQuest.

4.4 Waste Acceptance Criteria at a Geological Disposal Facility

When radioactive waste is disposed of in an operational GDF it will be required to be compliant with waste acceptance criteria (WAC) that are specifically defined for the facility. The WAC will be determined by the safety standards to be achieved by the GDF, including requirements specified in the authorisation/permit for disposal and conditions attached to the nuclear site licence, and also take account of standardisation, design constraints, legal, operational and economic factors. The WAC are expected to be produced by the facility operator, overseen by the relevant regulatory authorities, and would be based on the safety cases produced for the operational and post-closure periods of the facility and would reflect the requirements for transport, as appropriate.

In the UK, plans for the geological disposal of higher activity radioactive waste are still at an early stage, so the information necessary to develop WAC is unavailable. However, in order that wastes can be converted into passively safe and disposable forms as soon as reasonably practicable, RWM produce generic packaging specifications. These specifications define the standard features and performance requirements for waste packages which will be compatible with the anticipated systems and safety cases for transport to and disposal in a GDF. The contents of the generic waste package specifications and the waste package requirements are defined so that they would be bounding for the likely constraints imposed by geological disposal. As such, these generic requirements may be considered as the ‘preliminary’ WAC for a future GDF [24].

In order to ensure that the packaging specifications satisfy the needs of all users, RWM have devised a hierarchical structure, as illustrated in Figure 12, that comprises three ‘levels’ of specification, in which each successive level represents an increasing degree of specificity. Each level satisfies a specific function and is produced for a particular audience:

- Level 1: the Generic Waste Package Specification defines high-level requirements for all waste packages destined for disposal in the GDF. It is aimed at industry regulators and stakeholders who are not directly involved with the packaging of waste.

- Level 2: the Generic Specifications define the requirements for all waste packages that will be disposed of in accordance with a specified range of concepts, and which will contain wastes with similar radiological characteristics. They are produced for industry regulators and for use by waste packagers involved with the development of new or innovative approaches to the packaging of waste.
- Level 3: the Waste Package Specifications define, where applicable, quantitative requirements for waste packages containing a specific type of waste and manufactured using a standardised design of waste container. They are produced for use by waste packagers intending to use such a waste container for the packaging of waste.

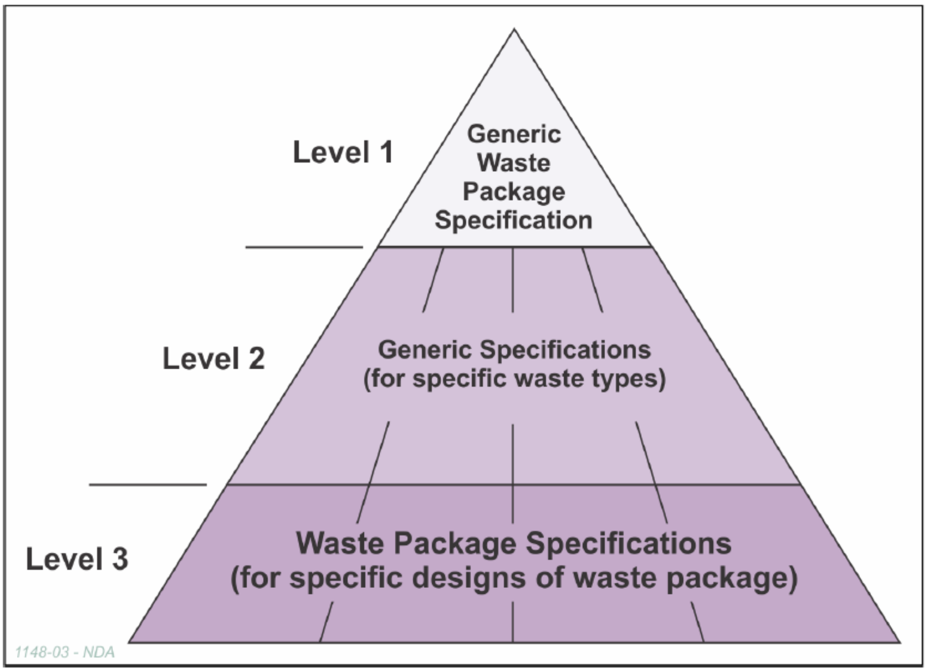

Figure 12 RWM packaging specification hierarchy [24]

RWM has carried out work to investigate the form of the WAC at other radioactive waste disposal facilities, both in the UK and abroad, notably [21]:

- the Waste Isolation Pilot Plant, the GDF for trans-uranic wastes constructed in an evaporite rock in the USA;
- the Konrad GDF for low heat generating wastes, constructed in a higher strength rock geology in Germany; and
- the LLWR, a surface disposal facility for LLW located in Cumbria.

The WAC for these three facilities are similar in form in that they each define:

- basic features (e.g. dimensions and shape) of a limited number of standardised waste containers;
- general requirements on all waste packages that are to be disposed of at the facility;
- specific requirements on the physical and chemical properties of the waste and the wasteforms produced by its conditioning;

- prohibited materials and those for which limits are placed on their quantities;
- the radionuclides that have significance to the safe operation of the facility, or which have been identified by regulatory authorities as part of the disposal authorisation; and
- limits on the inventories of those radionuclides, together with specific controls on the quantities of fissile material.

The form of the WAC for a UK GDF has not yet been defined but it is reasonable to assume that it will be similar to what is outlined above, and that it will be influenced by the fact that, at the time when a GDF is available to accept waste, much of the waste in the UKRWI will have been packaged.

5 Relevant Guidance and ALARP Arguments

The development and use of DPCs could offer considerable benefits to waste producers, the NDA and RWM as these organisations deal with decommissioning the country's nuclear sites and consider options for spent fuel. Introducing a container that is capable of safely performing more than one of the three main functions of a radioactive waste container (interim storage, transport, and final long-term disposal) could:

- Reduce the number of containers required;
- Reduce the space required for storage of containers;
- Increase the amount of metal to be disposed of;
- Minimise the number of waste handling operations;
- Minimise the number of handling/transfer systems required;
- Reduce the number of shielded stores required; and
- Reduce the cost of a GDF solution.

Despite these significant drivers for adopting DPCs, there are many technical challenges that must be considered for building a safety case that enables their use in the UK. This section of the report will attempt to briefly address the technical considerations that are specific to DPCs. It investigates the guidance and research on DPCs that have been published by a range of UK and overseas organisations, as well as considering the approach towards DPCs taken by several countries and organisations.

There is considerable overlap with terminology regarding multi-purpose containers (MPCs) in much of the guidance reviewed for this report. Where relevant, these two terms are used together to refer to the overall family of containers used for a combination of any of the following functions: storage, transport and final disposal of radioactive waste. Several documents refer only to MPC-based systems as the guidance is deemed to be potentially relevant to, and often interchangeable with, an equivalent DPC-based system.

5.1 IAEA Studies and Guidance

The IAEA is the world's central intergovernmental forum for scientific and technical co-operation in the nuclear field and works for the safe, secure and peaceful uses of nuclear science and technology [25]. The IAEA promotes and supports the establishment of comprehensive regulatory frameworks that consist of relevant legislation, regulations and guidance to ensure the safety of nuclear installations throughout their lifetime [26]. The following sub-sections refer to the most pertinent guidance and discussion surrounding DPCs published by the IAEA. For each source, a brief description is provided followed by a précis of the most relevant sections. This section aims to highlight sources of information on DPCs to the reader and pick out short sections that provide guidance on their adoption in national programmes and regulatory frameworks. It is not intended to be a detailed examination into how DPCs should be implemented in the UK.

5.1.1 Methodology for a Safety Case of a Dual-Purpose Cask for Storage and Transport of Spent Fuel, IAEA (Draft) [27]

In 2011, the IAEA initiated a study to develop guidance for the structure and content of an integrated safety case for DPCs for the transport and storage of spent fuel. A draft Technical Document [27] was written to summarise the three-year activity undertaken by a joint working group comprised of the Waste Safety Standards Committee (WASSC) and Transport Safety Standards Committee (TRANSSC). The report addresses the technical aspects of demonstrating the safety of the DPC design during storage, and compliance with the transport safety requirements extant at the time of transport at the end of the storage period.

The report is split into two parts. Part 1 provides a generic consideration of the structure and contents of a dual-purpose cask safety case (DPCSC) and includes information on administrative matters, specification of contents, DPC specifications, DPC performance criteria, compliance with regulatory requirements, operation, maintenance, and management systems. Part 2 provides generic and specific considerations for technical assessments of the safety case. Each of the following paragraphs in this sub-section discuss some of the more specific points that should be considered for DPCs.

5.1.1.1 Tracking the history of the DPC and its safety case

Owing to the many potentially complex variables associated with a DPC’s lifecycle and the several stages that are potentially challenging to a container’s properties (e.g. storage above ground in an industrial facility near the sea, or transporting on a railway line), a DPCSC must be treated as a “rolling process”. From its inception as a design up to its decommissioning, the safety case must be updated periodically, or when new findings need to be incorporated. Each update should follow a detailed change control procedure, indicating the version history and a clear identification of the stage of the DPC lifecycle.

5.1.1.2 Operational scenarios

There are multiple operational scenarios for a DPC system, which depend upon which combination of storage/transport/disposal functions the container is required to perform, and on the design of the container itself (i.e. whether the container is designed to operate unchanged through all operations or whether a range of overpacks are required for each stage). Each scenario will have a different set of operational steps that will be required to be listed in the DPCSC. These include:

- DPC package preparation (for transport and storage);
- On-site transport (before storage and/or after storage);
- Off-site transport (before storage and/or after storage);
- Handling at storage facility (before and after storage);
- Storage (on-site or off-site); and

- DPC package unloading (at the destination of transport after storage).

The report provides guidance for each of the six operational steps, detailing how each step might be carried out and under which regulations. The report also includes an example flowchart of operational steps for a DPC that is included below as Figure 13.

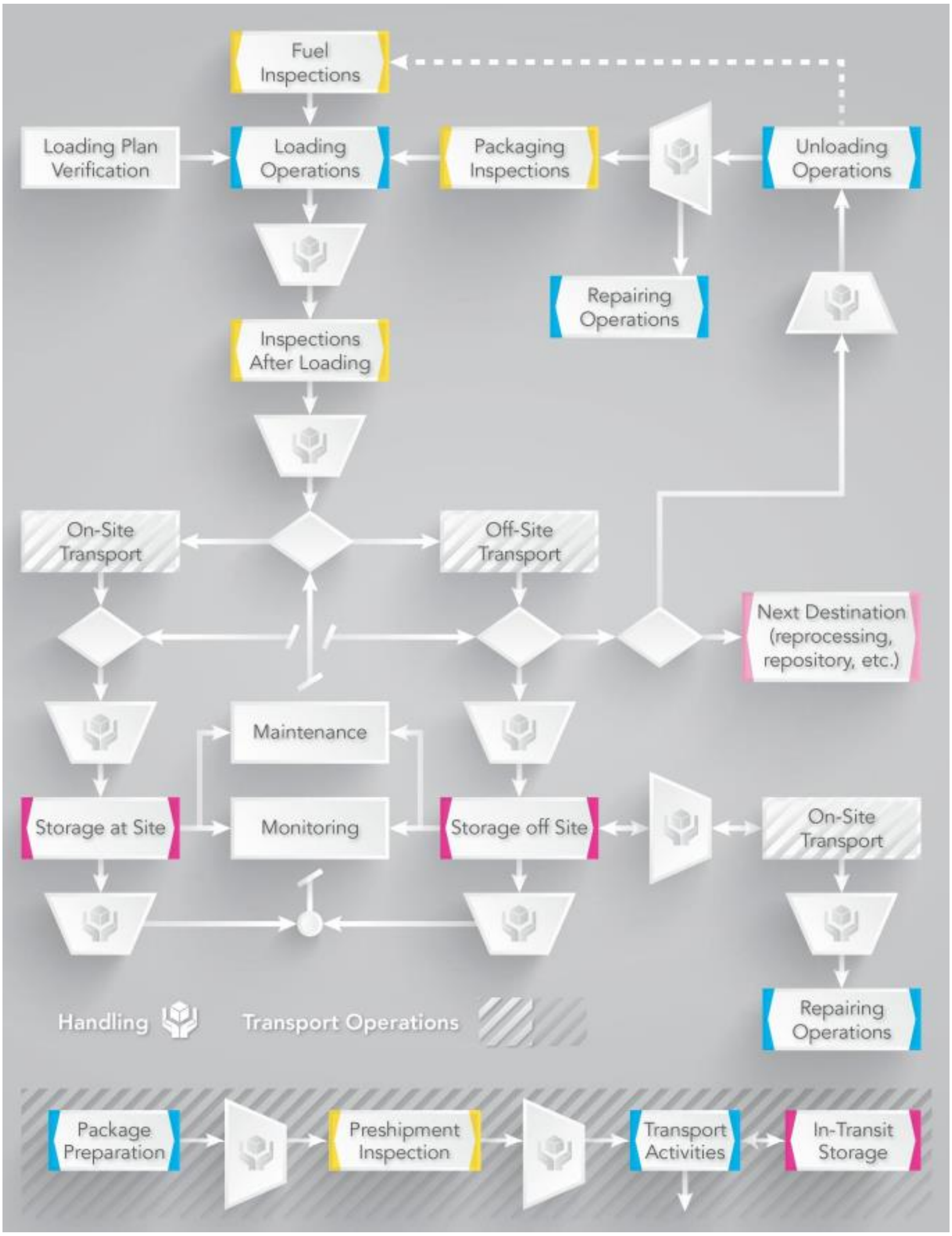

Figure 13 Transport/storage operational steps [27]

Figure 14 shows directional arrows on the operational steps diagram to demonstrate potential routes in the flowchart for two typical operational scenarios, namely on-site and off-site storage.

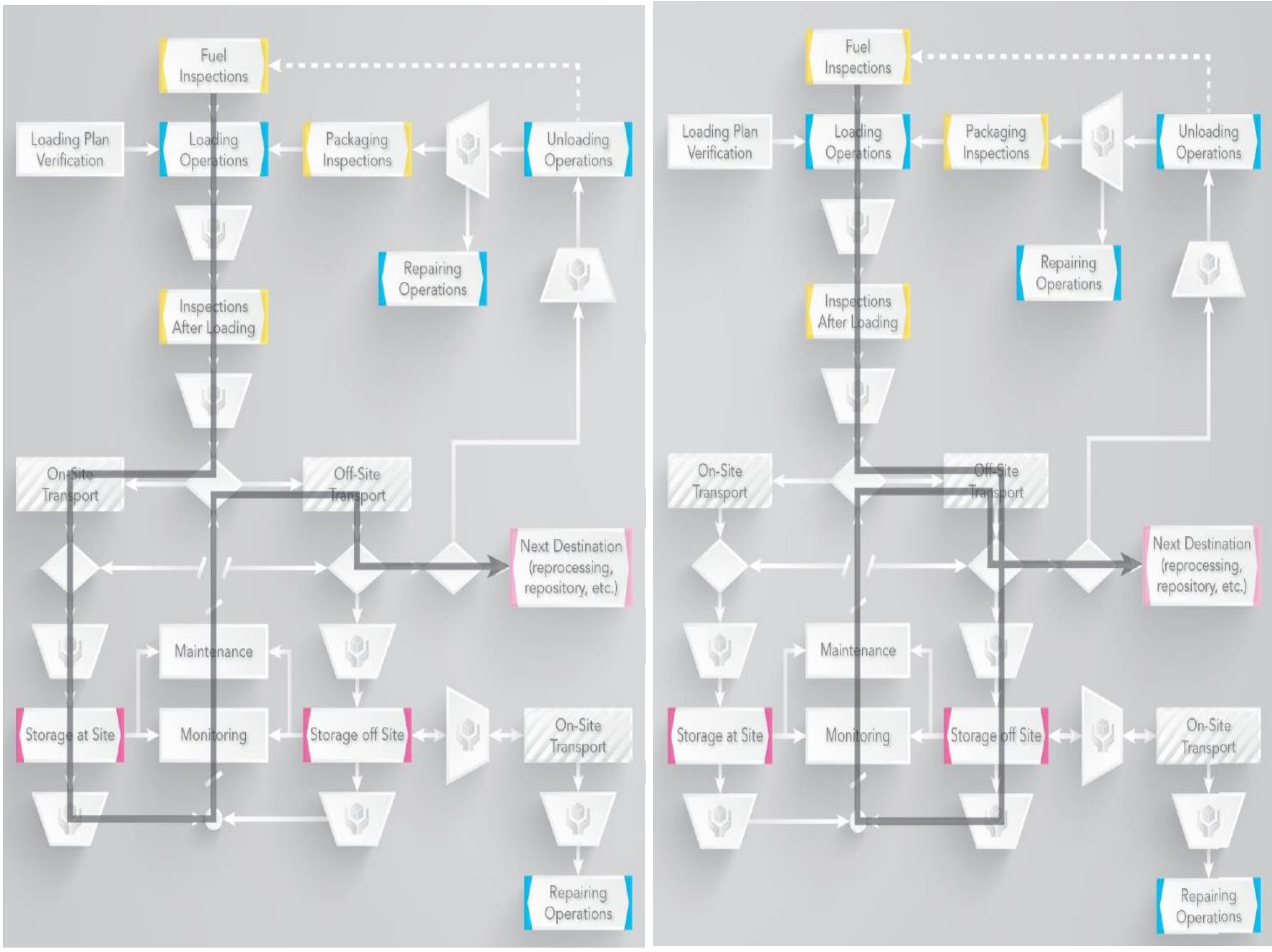

Figure 14 On-site (left) and off-site (right) storage operational steps [27]

5.1.1.3 Design considerations

The safety assessment and approval or licensing procedures need to consider the differences between the three potential DPC applications (i.e. storage, transport, disposal). The elements of the storage regime, the storage environment, the monitoring/inspection, and the records, are all required to demonstrate compliance with the transport safety case and should be clearly stated in the safety case in compliance with the transport regulations. This will enable the designers of the storage facility and those operating it to clearly understand what has to be implemented in the storage regime and provide the necessary records for future transport that this criterion has been achieved.

The design of a DPC should be defined in a 'design specification' that is justified in a technical assessment. The specification must include the acceptance criteria for transport and storage, as illustrated in Figure 15.

The transport package design acceptance criteria are derived from the international and national transport regulations, whereas the acceptance criteria for storage are derived from international standards and national regulations. The acceptance criteria must also consider the requirements that are specific to the storage facility design.

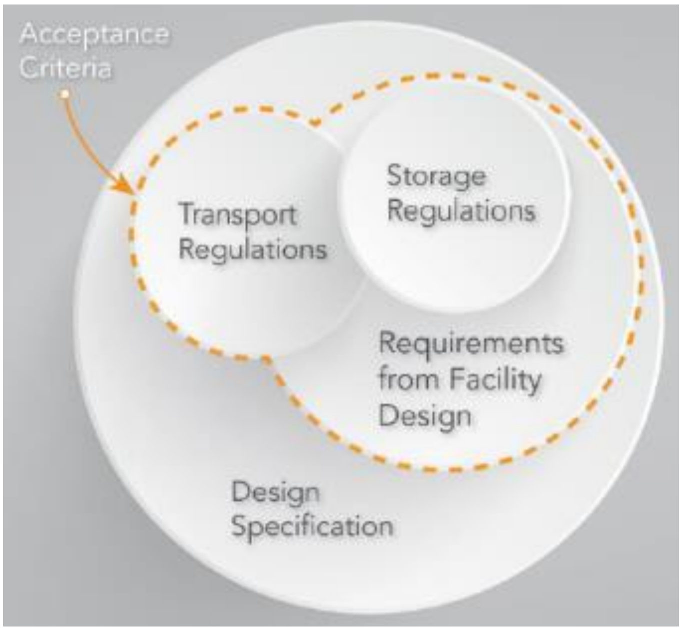

Figure 15 Relationship between design specification and acceptance criteria [27]

5.1.1.4 Components and ageing mechanisms

Although component ageing is an important aspect to be considered for all containers associated with radioactive waste, the dual-stage application of DPCs requires extra care for this aspect of design. For example, transport of a container holding spent fuel following a long period of storage should be undertaken very carefully, as the forces involved on such a journey could exacerbate an ageing issue and cause a failure.

The report considers a range of degradation mechanims for ageing of DPC components, and gives an example ranking the importance for storage and transport provided in terms of Low/Medium/High. The DPC components considered are:

- Cladding;
- Assembly hardware;
- Fuel baskets;
- Neutron poisons;
- Neutron shielding materials;
- Container; and
- Inert fill gas.

Although the above list is taken from an example table in [27], it does serve to highlight how certain stressors may not be an important issue for storage but may be for transport, or vice versa. The process can be treated as a gap analysis to identify data and modelling needs to develop the desired technical basis for storing and transporting the DPC.

5.1.2 Recommendations to TRANSSC/WASSC [28]

An objective of the work discussed in 5.1.1 [27] was to provide recommendations to the TRANSSC/WASSC for changes to be made to existing IAEA requirements and guidance relevant to the licensing and use of transport and storage casks for spent fuel. These recommendations [28] were divided to be relevant for the two committees and the most appropriate points are included below under the committee headings.

5.1.2.1 TRANSSC

1. Conformity with transport regulations that may change in future is an issue for DPCs waiting for future transportation. The Working Group recommends considering the introduction of a definition of DPC packages in the IAEA transport regulations.
2. There should be a requirement or guidance in the IAEA transport regulations to consider ageing of packages that are intended to be stored for a long time before the transport.
3. Any change of the IAEA transport regulations shall consider that in the section "Transitional Arrangements" in SSR-6 DPCs need to be considered in an appropriate manner so that they can be transported after storage. This applies to DPCs already fabricated and being used for storage of spent fuel.
4. The key issue is how to maintain the DPC Safety Case (DPCSC) for transport during storage – recognizing that storage may be for an extended period of time – so that the DPC can be used for transport regardless of the period of storage. This requires periodic review of the DPCSC and periodic inspections of the DPC. In the review, the gap analysis should be made to identify any impact of changes of transport regulations to the DPCSC and to existing DPCs. Compensating arrangements, if necessary, should be proposed at that time. The gap analysis should consider changes in regulations and change in knowledge since the previous approval period. Therefore, it is recommended to TRANSSC to develop an appropriate guidance material on this matter in SSG-26.
5. The transport regulations (SSR-6) should be reviewed with respect to the timespan between loading of the package and the completion of the shipment after storage to be consistent with the operation of a DPC, which will be transported more than a few decades after loading; e.g. it should be clarified that interpretation of para. 229 of SSR-6 does not imply that the maximum allowable timespan for a transport postulated is less than one year.

5.1.2.2 WASSC

1. Current SSG-15 "Storage of Spent Nuclear Fuel" describes an "ageing management program" only generally. It would be more informative for Member States if it could include a guideline for preparing an "ageing management program". Therefore, it is recommended to WASSC to

include the description in 1.12.2 and 1.12.3 of "Guidance for preparation of a safety case for a DPC containing spent fuel" into SSG-15 as an ANNEX.

2. It is beneficial to develop generic test conditions for storage and on-site transport in order to develop a DPCSC. For example, IAEA (or "the regulators") would need to develop a generic accident drop-test condition applicable on-site equivalent to the transport package drop test requirements in SSR-6. However, it may be difficult to establish rigorous test conditions (e.g. specifying the drop-test target) that are universal, owing to differences in national approaches. Therefore, the Working Group recommends to WASSC to support the development of a methodology to assess generic test conditions for storage and on-site transport in assisting the Member States to establish their national requirements.

Although the above points are recommendations to the aforementioned waste and transport committees, in the context of the report delivered to ONR, they serve as points of guidance to consider for UK-specific guidance; i.e. that any future national regulatory framework ensures these aspects and considerations are addressed.

5.1.3 Session Discussions Summaries: International Workshop on the Development and Application of a Safety Case for Dual Purpose Casks for Spent Nuclear Fuel, IAEA [29]

A workshop was conducted to help develop the work following the methodology described in [27]. A set of summaries was published from this workshop that summarised the discussions in each of the sessions. Below are a few points raised from some of the most relevant sessions. These points are not intended to provide specific guidance; however, they do indicate the topics considered at the workshop in 2014 and highlight the questions raised by attendees that may have been addressed since. The workshop also considered the contents and possible omissions of [27]. Recommendations were discussed for future IAEA activities regarding DPCs.

5.1.3.1 Legal and regulatory framework

- A question was raised regarding whether data on results of periodic inspections were available, and a point was made that experience should be accumulated. The availability of these data should be further investigated.
- Consideration should be given in how to accommodate new technological advances or regulatory changes when renewing transport licences.
- What are the inspection requirements for licence renewal?
- How does one harmonise storage and transport licensing?
- Test programs in Japan have carried out drop tests with aged seals.
- Uncertainty in ageing: how to add a sufficient safety margin, and at what point is a cask harmfully "aged"?
- An investigation into the required infrastructure to accommodate possible actions that may need to be taken to enable transport after long-term storage.

5.1.3.2 Design of storage facilities and operational experience of DPCs

- Criteria for inspecting casks; what are the phenomena requiring inspection?
- Records retention methodology and the media the methodology is stored on.
- Ageing management plans (AMPs) and international/national exchange of operational experience.

5.1.3.3 Designing casks for dual-purpose operations

- Design issues.
- Transport requirements vs storage requirements.
- Designing DPCs to enhance monitoring and inspection.
- Fuel design to consider storage and transportation.
- Regulatory differences between country of manufacture and country of cask user (this could be a particular issue to consider for the UK).
- Three-purpose containers to use for disposal as well as transport and interim storage.

5.1.3.4 Comments on the *Methodology for a Safety Case of a Dual-Purpose Cask for Storage and Transport of Spent Fuel* Technical Document

- Gives good structure.
- Regulator can also use as a reference to standard review plan.
- AMP application could be more detailed.
- Missing contents in the draft:
    - Lessons learned, design changes towards inspection for ageing management;
    - Impact of basket deformation on retrievability of spent fuel;
    - Thermal effects of casks in an array, impact on pads;
    - Damaged fuel;
    - Generic storage accident conditions;
    - DPC for HLW.
- Recommendations for future IAEA activities:
    - Experience compendium;
    - Considerations in transport regulations;
    - Operator’s experience;
    - Keep activities like this workshop;
    - Universal acceptance criteria for storage / generic safety case.

5.2 UK Studies and Guidance

In the UK, several SLCs have purchased DPCs/MPCs for interim storage, transport and potentially final disposal of radioactive waste and materials, following similar decisions abroad. This has led to several studies exploring how the UK might implement a DPC system and RWM considering this container type in their generic design and specification documents for a future GDF. The UK has yet to license these containers for multiple uses because of the need for containers to meet the separate regulatory requirements pertaining to on site storage, transport and final disposal.

This section highlights some of the work undertaken in the UK in investigating the use of DPCs and MPCs and summarises the conclusions from this work.

5.2.1 *The feasibility of using multi-purpose containers for the geological disposal of spent fuel and high-level waste*, Galson Sciences Ltd et al. [30]

This report investigated several models for how the UK might develop a MPC for the safe containment of radioactive waste during storage, transport and disposal for spent fuel and HLW. The optioneering study suggested that this could be achieved in either of two ways:

- **Canister-based approach**: The spent fuel is placed in a sealed vessel that is contained in different overpacks for storage, transport and disposal. Each overpack would be designed to meet the specific safety requirements of the relevant waste management phase.
- **Cask-based approach**: The spent fuel is placed in a single container (or cask) that meets the safety requirements of all phases of waste management (i.e. storage, transport and disposal).

Operational models for both the above options were described. The potential issues relating to the feasibility of using such MPCs for spent fuel (including that from new build reactors) and HLW disposal were identified in terms of constraints on the transport of large consignments of radioactive materials in the UK. The impacts on the operational and post-closure phases of a GDF were also considered.

The study concluded that a canister-based MPC system with different overpacks designed to meet specific functional and design requirements for storage, transport and disposal would be preferable to a cask-based system. It was identified that the canister should be such that:

- The condition of the spent fuel or HLW can be monitored during the long-term dry storage period;
- When placed in a transport overpack with impact limiters, it can be transported on the UK rail network;
- When placed in an appropriate disposal overpack, it meets the thermal constraints and containment requirements under evolving conditions for the range of possible disposal options; and
- It remains sub-critical under normal and accident conditions for transport and GDF operations.

Key areas where further knowledge is required to understand the constraints on using MPCs were identified as:

- Understanding methods and regulatory requirements for monitoring and inspection of wastes in MPCs during and after long-term storage and after transport to a GDF.
- Understanding the condition of spent fuel and container materials after long term storage, including the potential effects of water carry-over from pond storage (e.g. effects of internal corrosion, radiolysis and gas pressurisation).
- Understanding threshold levels at which the handling of MPCs becomes more complicated, for example, in terms of the equipment and designs needed to handle MPCs above certain sizes, weights or temperatures in a GDF.
- Understanding thermal constraints by considering the effects of high temperatures on, for example, corrosion, gas generation and engineered and/or natural barrier performance.
- Understanding criticality constraints (e.g. geometries, poisons) on large MPCs containing spent fuel or plutonium wasteforms.
- Identifying the constraints on using a canister-based MPC concept in the form of small units that can be consolidated for storage but can be transported in transport overpacks as single units and disposed of in single or multiple units in disposal overpacks.

5.2.2 *Concept Development: Disposal Concepts for Multi-Purpose Containers*, RWM [31]

As part of the research and development programme in support of the geological disposal of the UK's HAW inventory, a study was conducted on behalf of RWM to identify potentially feasible disposal concepts for a UK MPC system that has recently been developed by RWM's engineering team, in RWM's three generic geological environments. Currently available MPC systems represent a storage but not a disposal solution for the UK; this study investigates whether a transport and disposal solution can be found for these containers to make up a UK-specific MPC system.

The UK MPC system design being considered by RWM would hold 12 pressurised water reactor (PWR) spent fuel assemblies and would be constructed from stainless-steel with wall thickness 10mm. The lid would be welded, and the internal atmosphere would be inerted. It would have a design life of 100 years. A key feature of the proposed MPC design is the large internal voidage. A large amount of internal void space is a typical characteristic of an MPC and allows for heat transfer by advection of hot gas during the storage period, although the current design does not rely on this process. The disposal container for the MPC would be fabricated from carbon steel and has a minimum thickness of 140 mm. It would have a lifting feature in the lid that would be similar to that found in the standard containers. The disposal container would have a design life of 10,000 years.

The MPC waste package is described in Section 2 of [31]. The MPC would be contained in a disposal container throughout GDF operations; the MPC inside its disposal container forms the disposal package. During transport to the GDF and to the underground unloading platform, the disposal package would be contained in a transport container with impact limiters; the disposal package inside its transport container forms the transport package.

The report details eleven disposal concepts for a UK MPC system. The majority of detail in the report is not relevant to this study, however the following technical points are relevant to the ALARP considerations of a DPC system:

- The thermal output of the waste package must be below about 1500 – 2000W before a thermal limit of 100°C can be satisfied for a backfilled facility in a higher-strength or lower-strength sedimentary host rock, even if only a single waste package is considered. This constraint arises from the range of thermal conductivities of the candidate buffer materials and host rocks; it is a fundamental constraint based on the laws of physics that cannot be altered[11]. Although not explicitly stated in published documentation, it is this constraint that has led most Waste Management Organisations (e.g. SKB, Posiva, Nagra, ANDRA, Ondraf-Niras) to consider disposal packages that contain the equivalent of four or fewer PWR spent fuel assemblies for use in disposal facilities where it is desired to dispose of the spent fuel within about 100 years of discharge from the reactor.
- The thermal work confirmed that the high thermal output of the 12-assembly MPC waste package would present significant challenges for disposal in higher strength or lower strength sedimentary host rocks, where the thermal limit is likely to be about 100°C. It would be necessary to wait until the waste package thermal output had decreased to 1500 – 2000W before the buffer/backfill could be emplaced, which would require cooling times of the order of 150 – 200 years after discharge from the reactor for the 12-assembly MPC waste package for PWR spent fuel.
- Estimated footprints to dispose of the full inventory are similar to those required for cases that use standard (four-assembly) packages and emplace the buffer/backfill 75 – 100 years earlier than for the MPC waste packages. The use of fewer, larger waste packages does not bring a significant decrease in footprint because the same amount of heat needs to be dissipated.

Disposal of the 12-assembly MPC waste packages appears to be technically feasible for the set of disposal concepts explored in this study. There do not appear to be any fundamental viability issues, but the long cooling times that would be required to meet the currently assumed thermal limits for disposal concepts implemented in higher-strength and lower-strength sedimentary host rocks may challenge the economic viability of the geological disposal of MPC-12 waste packages.

5.2.3 *Safety aspects specific to storage of spent nuclear fuel TAG*, ONR [32]

The ONR has the responsibility for regulating nuclear safety in accordance with the Energy Act 2013, which provides the framework of responsibilities and the powers of the organisation. Other legislation that underpins the legal framework for the nuclear industry includes (but is not limited to): the Health and Safety at Work Act 1974; the Nuclear Installations Act 1965 (as amended); the Ionising Radiation Regulations 2017; along with other relevant statutory provisions of the Health and Safety at Work Act. A series of Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs) have been published that provide general guidance to advise and inform ONR inspectors in the exercise of their regulatory judgment.

The TAG on *Safety aspects specific to storage of spent nuclear fuel* covers the guidance on storage of spent fuel and deals with both the short-term storage of high decay heat fuel recently removed from the reactor and long-term storage of low decay heat fuel awaiting onward processing or disposal. The

[11] However, the post-closure temperature rise depends on how quickly the thermal power decays as well as on the power at a particular time. Placing a limit on the power at emplacement alone is not sufficient; more detailed calculations are needed.

guidance does not cover operations prior to storage; onward processing or disposal after the period of storage; movement on-site or transport off-site; or construction, commissioning, and decommissioning of storage handling facilities. The topics of the guidance include control of spent fuel; the storage period of the fuel; passive safety measures in place; criticality controls; allowances for future changes; information recording and reporting; and safety case and safety assessments.

A detailed overview of this document will not be included here, as it is an ONR publication. However, it would be remiss not to include this brief summary, as it is relevant UK guidance and should be acknowledged in this section.

5.2.4 Sizewell B Dry Fuel Store

Électricité de France Energy Nuclear Generation Limited (EDF) are licensed to operate Sizewell B Nuclear Power Station in Suffolk, UK. In December 2016, the ONR granted EDF consent for a Dry Fuel Store to be used to store spent nuclear fuel from the station's PWR [33]. In March 2017, EDF placed the first cask inside the storage facility, representing the first operational dry fuel storage system in the UK [34].

The cask model licensed for use in the Sizewell B dry fuel store is the HI-STORM MIC (Holtec International Storage Module Mega Impact Capable), designed by Holtec International in collaboration with EDF. The leak-tight cask features a double wall of steel and concrete and is designed for a 100-year life. The overpack is equipped with an impact limiter, designed to protect the loaded canister during its lowering into the cask's cavity. The exterior shell of each canister is eddy current tested to serve as a reference for future ageing management surveillances. The shielding in the cask provides orders of magnitude greater radiation blocking than the standard cask design, reducing the dose to workers in the surrounding environment [34].

5.2.4.1 ONR involvement with Dry Fuel Store commissioning

EDF requested that ONR consent to the commencement of active commissioning of the Sizewell B Dry Fuel Store process. This requirement was established through ONR issuing Licence Instrument (LI) 531 against Licence Condition 21 (4) Commissioning, under Sizewell B's Nuclear Site Licence, Licence Number 63 in December 2012. The ONR assessment and inspection work as part of the consideration for consent is detailed in the Project Assessment Report [33] and included the following activities:

- Monitoring EDF's development of the Dry Fuel Store Safety Case and the approach to commissioning equipment and operations to confirm design and safety functional requirements were met and the process was reliable;
- Confirming Dry Fuel Store project governance was robust by ensuring the effectiveness of processes delivered through its Design Decision Panel (DDP), Testing and Commissioning Panel (T&CP) and Project Safety Review Group (PSRG). This considered: whether individuals were suitably qualified and experienced persons to carry out their duties; that appropriate rigour was applied in the monitoring and sentencing of testing and commissioning results; that suitable project oversight was in place to confirm the project's preparedness to commence active commissioning;

- Confirming that documentation for inactive testing and commissioning had been completed and that technical queries raised from non-conformities had been addressed;
- Ensuring appropriate training was delivered to operators, operating instructions were produced to the appropriate standard and safety case limits and conditions were clearly identified;
- Confirming appropriate organisational control and supervision was in place for active commissioning of the dry fuel store process to take place.

5.2.4.2 Sizewell B Dry Fuel Store process

The ONR Project Assessment Report [33] provides a brief description of the dry fuel storage process, which begins with the movement of empty MPCs into the Station's drained cooling pond preparation bay. The MPC contained within its shielded transport package (HI-TRAC) does not have its lid fitted. The preparation bay is then flooded with cooling water to submerge the MPC and HI-TRAC. This is then moved under water through the pond fill bay transfer gate into the pond fill bay where 24 undamaged spent fuel assemblies are loaded into the MPC. The lid of the MPC is then fitted and secured with toggle bolts and moved back to the pond preparation bay. The preparation bay is drained and the MPC and HI-TRAC decontaminated and external cooling fitted. A small volume of pond water is removed from inside the MPC and checked for caesium levels to confirm fuel clad is undamaged. If caesium levels are acceptable the lid of the MPC is then welded to the shell of the vessel. After inspection of the lid weld any pond water inside the MPC is ejected using pressurised helium followed by drying of the fuels using heated helium gas. During the drying phase, the presence of krypton gas is monitored, again to confirm fuel clad remains intact. The MPC is then pressurised with dry helium gas and sealed to allow passive cooling within the MPC. This process is based on natural thermal convection, with contained fuel elements heating the helium gas which rises pushing cooler gas back down in the MPC. The heat energy in the helium is dissipated by conduction through the wall of the MPC, and then radiated from its external surface.

The loaded and sealed MPC is then moved within its shielded overpack from the pond building to a purpose-built storage facility where it is transferred into a shielded HI-STORM over-package.

5.2.4.3 Consequences of the commissioning of Sizewell B Dry Fuel Store

As a result of the licencing and commissioning of the Sizewell B Dry Fuel Store, it is planned that this system will be implemented for Hinkley Point C. The system is also identified in the AP1000 generic design assessment (GDA) and is likely to influence all future PWR (and potentially boiling water reactor (BWR)) spent nuclear fuel interim storage processes in the UK.

5.2.5 How the Management of Higher Activity Wastes are Influenced by Transport Requirements and a Case for Change: Type W [35]

The international conference on the Transport, Storage and Disposal of Radioactive Materials in 2018 saw the presentation by Sellafield of a proposition to develop a new type of container, named Type

W package [12]. Operators at Sellafield Ltd currently plan to use IP-2 packages for wastes at the lower activity end of HAW, whist Type B packages (disposal unit + transport container) will be used for waste of higher activity.

Radioactive waste requires large volume packages due to the requirements of disposal at the GDF and also due to the size of some of the items. This normally results in the use of an IP-2 package. However, for some packages the material classification requirements may be met for an IP-2 package but due to the nuclides that are present the unshielded dose rate limit cannot be met. This waste then requires that a waste package is packaged in a Type B package. A Type B package generally does not offer the cavity volume of an IP package, therefore requiring size reduction of the contents and also brings a huge step change in approval requirements and associated cost.

Sellafield therefore propose that there is the opportunity to initiate a change to the transport regulations that would define a new transport package type that would allow certain high dose wastes exceeding IP-2 unshielded dose limits to be transported in appropriately shielded disposal packages without the need for a transport container.

The paper that accompanied the presentation at the conference includes proposed draft regulatory text for such a package and lists a series of benefits for preparing, packing and handling wastes that could result from its implementation. Any changes to the transport regulations for this new type of package would likely to be relevant to DPCs/MPCs, as any points that address the transport of higher activity waste using UK infrastructure would begin to clarify some of the unknowns associated with approval of storage and transport of radioactive waste packages. At this stage it is unclear whether some DPCs/MPCs would or could be classified in this new category of package, however it is likely that DPCs/MPCs will be discussed if changes to the regulations begin to take place.

5.3 International Studies and Guidance

Several countries around the world have already adopted DPC and MPC systems for HLW and spent fuel and there are many learning points for the UK in the regulatory and technical approaches these countries have taken. This section focuses on some the guidance, research and operational examples developed in the USA, Germany and Switzerland.

5.3.1 USA: *White Paper: MPC for DoE-owned spent nuclear fuel*, Westinghouse Idaho Nuclear Company, Inc. [36]

This White Paper represents an early stage assessment by the US Department of Energy (DoE) for the use of MPCs for spent nuclear fuel. Therefore, the study provides an insight into the approach the organisation took to issues associated with implementing a multi-stage disposal solution that could share similarities to the UK situation.

Although the report was published in 1994, the document serves as a record of an approach into examining the advantages, disadvantages and other considerations of using a DPC/MPC as part of the strategy for interim storage, transportation and disposal of commercial spent fuel.

[12] It should be noted that this is the name suggested by Sellafield Ltd. This new type of package has also been referred to as an IP-4 package type.

The document details criteria and ALARP-related considerations for a range of DPC/MPC design specifications, including:

- Fuel durability, dimensions, and composition;
- Potential for designation as a hazardous waste;
- Temperature limits of the spent fuel;
- Criticality safety;
- Special nuclear materials (SNM) protection – safeguards and accountability;
- Repository waste package design requirements;
- Fuel acceptance specifications; and
- Segregation of spent fuel types.

5.3.2 USA: *Materials aging issues and aging management or extended storage and transportation of spent nuclear fuel*, USNRC [37]

The United States Nuclear Regulatory Commission (USNRC) undertook a study to evaluate the potential degradation of the systems, structures, and components of dry cask storage systems for spent fuel storage. The evaluation was performed to determine whether degraded conditions could be anticipated to develop in the materials of the SSCs over an extended storage and transportation period (> 120 years), and whether the anticipated degradation would challenge functions important to safety during storage and during post-storage transportation, including fuel retrieval for final disposition.

The evaluation showed the following major gaps in data, information and technology:

- A wide variety of alloys are, and have been, used for fuel cladding in the US commercial reactor fleet, and the response of these alloys to extended storage and transport (EST) depends on chemistry, thermal mechanical process history, irradiation history and other fabrication/service variables. The information required to address the response of a specific cladding to a specific age-related degradation process is limited and, in many cases, simply not available.
- The conditions required for corrosion-induced degradation processes such as stress corrosion cracking are, and will continue to be, present on the external surfaces of dry storage canisters, especially those fabricated from austenitic stainless steels.
- Delayed hydride cracking in the Zircaloy-clad fuel may occur and hydrogen embrittlement is anticipated as the fuel cladding temperatures fall during EST. The occurrence of these two phenomena would create cracks in a fuel cladding that has minimal fracture toughness.
- Bolted closures with metallic seal materials will experience sealing force reduction and may lose sealing capabilities during EST. The information needed to establish replacement schedules is lacking.
- The heat output, attendant radiation, and reactivity of spent MOX fuels significantly exceed those of UO_2 fuels; cask thermal and radiation design must address these conditions to avoid excessive temperatures for long times that would impact fuel and canister degradation.

- There is a potential for delayed hydride cracking of fuel cladding and stress corrosion cracking of the canister occurring in the same dry storage cask. The impact of this occurrence on dry storage confinement, retrievability, handling, and post-storage transport has not been well developed.
- The concrete pads, and also casks, will deteriorate and a path forward to address the impact of such deterioration has not been defined even though there are some independent spent fuel storage installations (ISFSI's) that are built on the only practical site at the reactor.
- The lifetime of the ISFSIs, in some cases, will exceed the lifetimes of the corporation using the facility. The information and regulations required for successful transfer of data from a dying or defunct corporation to the new tenant is minimal at best[13].
- The aluminium-based baskets and neutron absorbers could degrade during EST because of creep and/or corrosion processes. An evaluation is needed for canister internals made from materials that are subject to creep deformation at anticipated storage temperatures (e.g., aluminium). Data are needed to demonstrate that such material degradation does not compromise the retrievability of the fuel assemblies or change the configuration of the absorber and limit its effectiveness to maintain subcriticality during re-submersion, if required.
- For both normal and accident conditions for high burn-up (HBU) fuel in EST, the technical basis needs to be defined for any assumptions on:
    - the number of failed fuel rods; and
    - radiological source terms.

The document also includes a table of all credible degradation mechanisms for EST, and the specific parts of the fuel and systems, structures and components that could be affected.

5.3.3 USA: *Extended Storage Collaboration Program*, EPRI

The Electric Power Research Institute (EPRI) conducts research, development, and demonstration projects for the benefit of the public in the USA and internationally. It is expected that most American nuclear power plants will need to implement on-site dry storage facilities by 2025 or shortly thereafter The EPRI is coordinating the Extended Storage Collaboration Program (ESCP)[14] to investigate the ageing effects and mitigation options for the extended storage and transportation of spent nuclear fuel and HLW [38].

Utilities employ dry storage systems to temporarily store used nuclear fuel on site. Given the lack of a final spent fuel repository or interim consolidated storage options, it will be necessary to use these dry cask storage systems for longer than their original approved licensed period of operation.

To confirm continued safe storage into the period of extended operations, ageing management programs are employed to maintain the safety functions of safety-related structures, systems, and

[13] This represents the USNRC view on how corporations operate in the USA and is not necessarily representative of the UK situation. It could be interpreted that only minimal requirements have been specified for ensuring the successful transfer of data from a dying or defunct corporation to a new tenant.
[14] A group of organisations representing the nuclear industry, federal government, national laboratories, and suppliers of used dry fuel storage systems.

components. These ageing management programs focus on inspections, assessments, and/or repair/mitigation activities to manage potential ageing effects. Therefore, non-destructive evaluation (NDE) is seen as an important aspect to confirm continued safe operation of dry storage systems.

As a result of the need to manage ageing during the period of extended storage operations, EPRI organised the ESCP, which is composed of several subcommittees; each tasked with managing different technical areas. For example, the NDE Sub-committee was organised to help coordinate efforts to develop inspection systems that could be deployed [39].

Initial findings from a Gap Analysis as part of the Program highlighted several technical gaps in the safety case for extended dry storage of spent fuel and subsequent transportation:

- Long-term degradation of high burnup used fuel cladding via creep and hydride reorientation
- Corrosion of the exterior of stainless-steel welded canisters containing used fuel in an inert atmosphere (for example, helium backfilled inside the welded canisters)
- Degradation of concrete used for shielding and structural purposes

The research that the ESCP sub-committees have conducted and published could provide a source of information for future UK DPC systems, including potential cask inspection and test procedures.

5.3.4 Germany: *GNS′ experience in the long-term storage at dry interim storage facilities in Ahaus and Gorleben*, GNS [40]

This presentation was part of the workshop discussed in section 5.1.3 and demonstrates the German storage facilities and systems for using dual- and MPC for dry storage of spent fuel and HLW. An overview is provided of the facilities themselves, with layout, number of containers and basic operation procedures. A list of stored, licensed casks is stated, with further information on when and what the licensing covers for each storage site. A brief history of the licensing procedures is recorded, which indicates how new licences have been granted to different types of waste, cask and cranes etc. over time. The relevant supplementary regulations that the facilities adhere to and report on are listed, with several of these further explored with examples of events and issues that the facilities have had to deal with and solutions they have found. The cask approval and in-service inspection topics provide an insight into the certification process and how periodic reviews are undertaken.

5.3.5 Germany: *Safety aspects of dual and multi-purpose casks for radioactive materials*, BAM Federal Institute for Materials Research and Testing [41]

The 10th International Conference on Radioactive Materials Transport and Storage (RAMTRANS) in 2015 saw a presentation on the safety aspects of the German system for managing spent fuel and HLW using dual- and multi-purpose casks. The slides give an overview of the main design features of the several DPCs and MPCs that are licensed in Germany. Several testing procedures for ageing phenomena are also demonstrated, including horizontal and vertical drop-testing, and corrosion testing of metal lid seals with caesium. A list of essential design considerations for managing ageing effects in dual-purpose transport packages includes:

- Design that considers ageing resistance of components and materials (materials ageing assessment, corrosion resistant components, e.g. by effective inner and outer coatings and medium penetration barriers, high quality in manufacture/documentation etc.);
- Operational conditions that prevent degradation propagation and ingress of corrosive agents as much as possible (drying, evacuation, inert gas atmosphere etc.);
- Periodic package design approval certificate renewal (gap analysis of the safety case, management system adaption etc.); and
- Inspection program for tests before transport (appropriate selection of measures considering storage experiences etc.).

5.3.6 Switzerland: *Dry Storage Ageing Project for DPCs in Switzerland*, Swiss Federal Nuclear Safety Inspectorate ENSI [42]

All Swiss nuclear power plants have opted for DPCs as the final predisposal solution, including those with extended wet storage capacities [42]. This paper details some of the Swiss research efforts into ageing issues of dry storage DPCs for spent fuel and the limitations that are placed on transportation of DPCs. The paper was presented as part of the proceedings for the 2016 International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM) conference.

In the Swiss context, DPCs are used for interim storage and transport. The lifecycle of a DPC (and of spent fuel in general) involves the following steps:

- Reactor operation;
- Cooling in the spent fuel pool;
- Loading of spent fuel in the DPC;
- Closing and drying of the cavity;
- Transport of DPC to the interim storage facility;
- Dry storage of more than 40 years;
- Transport to the waste conditioning facility next to the final repository;
- Retrieving the spent fuel and deploying it into containers designed for the final disposal; and
- Transfer and final disposal in the deep geological repository.

The acceptance criteria for storing DPCs in a facility are usually based on a design lifetime of 40 years' storage, however with the final repository in Switzerland suffering delays to its programme this storage time will likely be longer.

Transport approval periods are usually limited to a maximum of five years. This forces new design features to be considered for incorporation into the design that could improve safety, operation or fabrication. The use of a DPC is usually limited to one transport from the power station to the storage facility and a second transport to the final repository after a long storage period of over 40 years.

For package components, the study identified the following for further actions;

- Metallic seals due to long-term temperature effects;

- Elastomeric seals due to difficult traceability of material properties;
- Aluminium alloys due to weakening and copper due to creeping under long-term temperature influence;
- Special stainless-steel alloys due to embrittlement caused by forced cooling; and
- Lid bolts due to the loss of pre-load because of relaxation effects.

For spent fuel, the study identified the following for further actions;

- Evaluation of acceptable hoop stress[15];
- Identification of temperature limits for the full life-cycle of spent fuel;
- Investigation of the overall influence of the drying process;
- Accuracy of temperatures identified by thermal analysis;
- Identification of structural loads introduced in spent fuel assemblies during accident scenarios; and
- Verification of cladding properties used in mechanical analysis.

For final disposal, the study identified the following for further actions;

- Ensuring cladding integrity;
- Check of cladding integrity before opening the DPC;
- Retrieval and handling of defective or damaged fuel; and
- Package design approvals (transport and storage) for damaged fuel.

5.4 ALARP Considerations

The ONR is responsible for ensuring that nuclear licensees and duty-holders meet their legal requirements to reduce risks so far as is reasonably practicable (SFAIRP). The concept of SFAIRP is usually expressed in terms of reducing risks to "As Low As Reasonably Practicable" (ALARP) [43] and is a requirement to take all measures to reduce risk where doing so is reasonable. ALARP is usually demonstrated by ensuring the argument in question is fully compliant with the relevant good practice and standards, however, for less developed/established techniques and processes, it may be demonstrated by implementing measures to the point where costs of any additional measures (in terms of money, time or trouble) would be grossly disproportionate to the further risk reduction that would be achieved [43]. From the perspective of DPCs, the ALARP considerations in this section will focus on some of the potential arguments for and against their implementation into the UK's long-term geological disposal solution for spent nuclear fuel from a nuclear safety perspective.

At this point it is pertinent to reiterate the main reasons for pursuing a DPC system, as stated in the IAEA guidance [27]:

[15] It was not clear from the report whether this refers to hoop stress in the fuel cladding, fuel assembly wrapper, or DPC.

- Reduce the number of containers required;
- Reduce the space required for storage of containers;
- Minimise the number of waste handling operations;
- Minimise the number of handling/transfer systems required;
- Reduce the number of shielded stores required; and
- Reduce the cost of a GDF solution.

5.4.1 Handling and repackaging

One of the most robust arguments for implementing a DPC/MPC approach with respect to storage, transport and/or disposal of radioactive materials is that such a system would remove many of the nuclear safety risks associated with repackaging the waste. A repackaging process would include activities such as deliberately breaking containment barriers and require multiple handling procedures to extract the waste package from the DPC/MPC before emplacement into the next cask in the process. These processes increase the risk of potential radiological release and increase the likelihood of encountering handling system errors, so avoiding repackaging could be considered an ALARP approach to storing *and* transporting radioactive materials.

Handling operations would be kept to relatively simple procedures that could notionally be conducted remotely, with no dose consequences. Removing a repackaging process from the waste handling procedure would also remove the need for a repackaging facility to be constructed, which will likely save considerable cost. However, extra provisions would instead be required for additional monitoring and testing to ensure the DPC/MPC still meets its operational specification.

A further consideration for repackaging is that a DPC/MPC packaging system would reduce the reuse of materials, as every overpack would be designed for the requirements of storage, transport and/or disposal, rather than reusing transport and disposal overpacks before emplacement in the GDF. It is likely that this would result in a more costly solution.

5.4.2 Containment barriers

DPC/MPC systems are inherently designed to have additional containment barriers built into them to meet the strict regulatory controls required for both storage and transport. From a storage perspective, the containment barriers must consider the heat dissipation and gas production during an interim storage phase by including features like venting, temperature measurement and gas sampling capabilities. However, if the system is also designed for disposal then the casks must also consider the long-term containment barriers required for geological storage, the design of the GDF, and the geology that it will be stored in.

By including dual/multi-use functionality into the design of the packages, DPCs/MPCs have these extra containment barriers at all stages of their operation which could enhance the nuclear safety performance at each stage, rather than rely on one specific overpack for each of the storage, transport and disposal stages. However, from an ALARP perspective, it is worth considering that such a system is *capable* of performing multiple roles, for example interim storage *and* transport, but in fact does not represent the *optimal* solution in isolation versus the specialist overpacks.

5.4.3 Container capacity and failed fuel

A factor to consider regarding DPCs/MPCs is their capacity: the number of spent fuel assemblies they can accommodate. While increasing the number of assemblies inside each cask increases the efficiency and decreases the cost, there is also a limit to the amount of decay heat a single unit can dissipate. Depending on the time that the spent fuel has been stored inside cooling ponds after irradiation, the storage capacity and heat radiating performance of the cask may be a limiting factor. This is discussed in section 5.2.2.

Fuel failure in modern light water reactors, such as PWRs and BWRs, is rare, with failure rates typically between 10^{-6} – 10^{-5} [44]. Fuel failures cause problems and inconveniences to plant operations, and also pose additional issues post-irradiation when considering handling, cooling, interim storage and disposal. Different countries take varying approaches to failed fuel, with some utilities (France, Sweden) having no specific conditions for handling and storage, while Japanese utilities contain leaking assemblies in leak-tight containers in cooling ponds [44]. However, once the failed fuel is removed from cooling ponds and placed in dry storage containers it is clearly beneficial to avoid repackaging if possible.

A DPC/MPC system to store, transport and potentially dispose of failed fuel would significantly reduce the nuclear safety risks associated with a leaking fuel assembly. Once the failed fuel is sealed inside the multiple containment barriers, the handling, monitoring and movement of it would become much easier and the package would not require reopening.

5.4.4 Ageing management and monitoring

Although there are several arguments that present DPC/MPC systems as an ALARP option for the storage and transport of some radioactive materials, these vessels are still susceptible to the physical, chemical and biological reactions that can break down containment barriers, given the right conditions. This necessitates that appropriate monitoring and ageing management inspections are applied to ensure that the performance of the package remains constant and remains ALARP, so that the radioactive material remains behind the containment barriers. This performance measurement is necessary not just for the decades of stationary interim storage, but also for the reorientation and forces associated with transportation to a disposal location.

While the implementation of a DPC/MPC system offers some additional benefits with regard to nuclear safety, such a system also reduces the opportunity to take corrective action if the waste evolves unexpectedly during interim storage. Key to ensuring that the DPCs/MPCs remain an ALARP option for their intended purpose will be the range of sensors and diagnostic equipment that are used to measure the health of the containers. For example, the HI-STORM overpack used in the Sizewell B Dry Fuel Store, discussed in section 5.2.4, has temperature sensors fitted to the base and lid to remotely monitor the temperature. The difference between temperature measurements indicates the health of the MPC containment, whereby a small temperature difference indicates containment is intact. This is based on the MPC being pressurised with helium gas, and natural thermal convection is in operation. A large temperature difference could suggest a loss of helium gas due to poorer heat transfer taking place within the MPC. The threat to MPC containment arises from stress corrosion cracking given the presence of chlorine.

Much of the research and investigative work undertaken and published regarding DPCs/MPCs relates to ageing management to ensure the performance of containers is acceptable and the safety case for their use remains ALARP. One important aspect of ageing management is robust record keeping for all elements of the DPC system. This is a common theme through the literature in sections 5.1, 5.2, and 5.3, in which the importance of documenting decisions and keeping accurate, detailed records of each stage of a DPC's life is critical to ensuring it continues to remain a safe container. This includes: the materials, specifications and plans used to manufacture the container; all key dates that involve a physical movement, change in function or alteration/upgrade; detailed plans for where and how the records are stored, as well as who owns the data and who is responsible for its safe-keeping over the potential century-long storage periods; and also detailed knowledge and procedures in place to understand which materials are likely to degrade over time in such an environment (e.g. rubber seals, impact limiters etc).

5.4.5 Future regulations and the GDF

Once all the technical aspects of the justification for the implementation of a DPC/MPC system are considered, a judgement is required to be taken on the vulnerability of such justifications to potential future changes to regulations and requirements. Expending additional cost in the short-term for a combined storage and transport system, only for tighter regulations or ageing issues to eliminate the justifications could significantly increase costs in future decades. The ALARP argument then hinges upon what constitutes as 'reasonably practicable', which is influenced largely by practice elsewhere.

There is also the possibility of future regulations that ease the use of DPCs/MPCs, such as the proposal to introduce new transport regulations mentioned in section 5.2.5. As more nuclear facilities approach the end of their lives and decommissioning activities commence across the country, there will likely be a significant increase in waste packages over a relatively short period of time. This will require the government and regulators to address the concerns of the industry and find pragmatic solutions to areas of legislature that begin to cause issues in safely storing, moving and/or disposing radioactive waste. This will likely result in clarifying the current regulations or introducing new legislature, whether from an international standpoint or via domestic routes by the UK government.

Other future risks to be considered include the design and construction of the GDF. The design is currently in a generic format, so as not to limit any choices for potential locations in the siting process. If a generic design changes substantially once a location is selected, then this could cause issues with the stock of DPCs/MPCs that are not optimal for future disposal solutions. Also, with the start of construction potentially decades away and interim storage facilities aiming for 100-year timescales, there is also a chance that by the end of a DPCs/MPCs life the disposal concept favoured by the government of the time has changed and the GDF that the DPC/MPC has been designed for is no longer appropriate.

6 Conclusions

The UK has operated a wide range of nuclear facilities for over half a century and these activities have led to the generation of radioactive wastes that are stored at various locations around the country. Every three years an **inventory** is taken to record the characteristics and quantities of the different waste streams, along with details of how they are contained and the plans for their eventual disposal.

The DI/IGD focuses on the waste streams that are expected to require geological disposal. The data from this inventory shows that almost half of all packages expected to arise up to 2137 will be from the Sellafield site; with **spent fuel**, uranium and plutonium packages representing *more* than half the total packages. This data helps the organisations that are responsible for decommissioning the UK's nuclear legacy to plan future solutions and facilities to deal with waste. These activities are especially pertinent as the country's first- and second-generation gas-cooled reactors retire and enter the early stages of decommissioning. These activities will significantly increase the types and quantities of radioactive waste in the inventory, with graphite, ferrous-based alloys, and concrete expected to represent the majority of the inventory in terms of mass. This wide range of waste streams requires a number of different container types to appropriately package the waste. While the aforementioned materials represent the biggest problem in terms of mass in the inventory; it is the HLW, such as **spent fuel**, that represents the biggest problem in terms of radioactivity and hence handling, packaging, and disposal. This report takes a detailed look at the option to use DPCs to package spent fuel for interim storage and transportation.

Transportation of the radioactive waste packages identified in the UK's inventory to their disposal location will be a significant undertaking, especially considering how the country's nuclear facilities are scattered around the UK and many are likely to be hundreds of miles from a potential GDF. Consideration must be given to the routes and mode of transport from the interim storage locations to the GDF. For **road transport**, many of the transport packages in the 2013 DI are such that the 50t mass limit for an STGO Category 1 would be exceeded. In the generic TSC, RWM is planning to use a combination of STGO Category 1 and Category 3 vehicles, recognising that the former can be used as an HGV for sufficiently light transport packages. For **rail transport**, the route availability is determined by particular structures such as bridges or aqueducts. Network Rail, as the infrastructure provider, is able to authorise vehicles with a higher axle weight than permitted by the route availability to use particular sections of track. This may be by the imposition of a speed limit or the use of barrier wagons to give wider separation to the load-bearing axles on a train. **Sea transport** is also an option; however, it is likely to be considered only if there is a significant saving in operator dose compared to the equivalent journey by road or rail.

The **regulation of transport** of radioactive materials by road and rail in the UK is by compliance with Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations. For transport by sea, the International Maritime Dangerous Goods Code needs to be satisfied. The IAEA Transport Regulations state that the objective of the regulations are to "protect persons, property and the environment from the effects of radiation in the transport of radioactive material". This is done by addressing: the containment of radioactive contents; the control of external radiation; the prevention of criticality; and the prevention of damage caused by heat.

For many decades the UK's spent fuel has been reprocessed at Sellafield to extract and re-use the uranium and plutonium. However, reprocessing is expected to cease in 2020 and the future plan is to store spent fuel until a decision is made on whether this material is classed as waste or not. In the meantime, as a generic design for the GDF is drawn up, the plans account for disposal of spent fuel assemblies in this facility. For the purposes of this report it is assumed that spent fuel assemblies will be disposed of in this GDF. The UK plans for spent fuel generated at Sizewell B and New Build stations is to store it in cooling ponds for several years until its decay heat has dissipated enough for the fuel to be placed in interim storage inside dry fuel stores, located at the various power stations. Once a location has been selected and the GDF constructed, the spent fuel will be transported from its interim storage location to the GDF for final disposal.

A period of interim storage could last up to a century, so the decision on container type is an important one. **DPC**s are designed to carry out **storage and transportation** functions, however with such a long period of storage before transport there is significant regulatory scrutiny given to DPCs to ensure they are able to maintain their performance over this time. A similar class of containers exist that are designed to perform storage, transport *and* disposal functions; known as MPCs. The regulations and limitations are similar for MPCs as for DPCs, so there is considerable overlap in the research and guidance for these container types. Several countries around the world have begun to implement DPC and/or MPC systems to store spent fuel, however as yet no country is transporting these containers from interim storage to a fully operational disposal facility.

The **IAEA** have published a draft version of guidance for developing methodology for a safety case to use DPCs for storage and transport of spent fuel. Although the guidance is not final and requires finessing, it provides a good basis for developing an approach for a **safety case**.

The UK has undertaken the rigorous approach of licensing a DPC for interim storage of spent fuel at the **Sizewell B Dry Fuel Store**, although it is only envisaged to use the DPC for storage at this time. Using the HI-STORM model designed by Holtec International in collaboration with EDF Energy, the first DPC was placed into storage in 2018. The processes, procedures and regulatory approach developed for this model of DPC and facility are likely to be replicated for future PWR and BWR dry fuel stores in the UK. However, the recent changes in plans for New Build nuclear power stations leaves the requirements for future dry fuel stores and DPCs uncertain.

This report has detailed a series of **case studies** and **relevant guidance** that demonstrates the significant effort in research and development that has been undertaken around the world to make dual- and multi-purpose casks and containers a viable option for storing radioactive wastes, specifically spent fuel. The implementation of a DPC system will remove the need for repackaging spent fuel packages, which will be particularly beneficial from a nuclear safety point of view for failed fuel. The **handling procedures** are likely to be simplified without any requirement to repackage and could be performed remotely. The multiple **containment barriers** designed for storage *and* transport could offer advantages over some storage-only systems, however it is not clear from the research whether a DPC in fact offers a less capable solution in isolation. However, for these containers to be continually considered an **ALARP** option for storing spent fuel into the future, a strict regime of **monitoring and ageing management** is required to ensure the DPC performance is kept constant. Key to this will be the range of sensors and diagnostic equipment that is used to measure the health of the containers through their period of interim storage. However, the ALARP arguments for use of a DPC system will be vulnerable to **future changes** to storage and transport regulations, as well as any changes to the design of the GDF.

7 Recommendations

The research and work reported in this document has highlighted the following areas for future work:

1. Interaction with industry is needed to understand the requirements of waste producers with respect to DPCs:
    a. Which waste streams would (or could) industry use DPCs for? Is spent fuel the only realistic stream or are there other waste streams that DPCs could be applicable for or beneficial to?
    b. When would DPCs be required in industry and how many?
    c. How many assemblies is industry expecting to dispose of per DPC?
        i. RWM commissioned some GDF post-closure thermal modelling on containers holding 48, 24, 12, 4 spent fuel assemblies (in high-strength sedimentary rock). This work found that any more than 4 fuel assemblies gave unacceptably high temperatures in the bentonite buffer around the disposal container. Is the industry aware of this constraint?
        ii. Has the storage and transport side of DPCs been disconnected from the disposal side?
2. A workshop with, or questionnaire to, relevant stakeholders to ascertain the answers to the questions above could prove useful.
3. The expected package arisings should be highlighted to the ONR TCA, with consideration given regarding the ability of the supply chain to deliver this number of packages, record the necessary documents, check and approve new designs, etc. Discussions should also involve other stakeholders that may benefit from early sight of this information.
4. Noting the requirement to demonstration compliance with the IAEA mechanical and thermal tests for Type A fissile and IP fissile packages, waste producers should consider segregation of fissile and non-fissile waste as there may be considerable cost savings associated with transport.
5. The limitations of UK infrastructure to transport large/heavy containers should be highlighted to the relevant stakeholders to ensure current infrastructure can handle intended movements, with limitations acknowledged in specifications.
6. Present findings from this report to Transport Container Standardisation Committee for further technical steer and guidance on likely regulatory expectations for DPCs in coming decades.
7. Investigate a source of data for AGR fuel pin failure rates. This was requested at the Technical Meeting at the beginning of May, but data in open literature could not be found. The data will likely exist somewhere within the industry and will complement the information on PWR failure rates discussed in this report.
8. Some technical specifications of DPCs require further investigation with respect to understanding how specific regulations (e.g. venting and the Pressure Systems Safety Regulations) are expected to be applied for the storage/transport/disposal of DPCs.

9. The UK is only just beginning to explore the operating envelope of Sizewell B Dry fuel store using DPCs (even if only for storage). With regulatory approval recently gained it will be interesting to see whether the industry shows further interest in DPC containers.
10. A change to the transport regulations to add a new package type has been considered by Sellafield. The appetite for helping to enable this could be further investigated at ONR. Would this introduction help or hinder the use of DPCs/MPCs in the UK?
11. Regarding the production of further guidance, it is likely too early for a specific TAG on DPCs, even if only signposting to various relevant documents already in existence. Without further input from waste producers and potential stakeholders, it is difficult to anticipate what will be expected of ONR for long-term storage and then transport (and potentially disposal) for DPCs/MPCs.
12. There is currently no legal framework for one entity to approve the multiple functions of a DPC/MPC (storage on site, transport in the public domain, GDF storage). This should be a discussion point at any future workshop or industry engagement.

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B Waste Package Stock and Arisings for Locations in the 2013 Derived Inventory up to 2137

Location	2m Box	$3m^3$ Box	$3m^3$ Drum	4m Box	500L drum	B/G box	C1 Cont.	C4 Cont.	SF Disp Cont	DCIC cube	DCIC cyl	WAGR Box	TOTAL
Aldermaston	0	14	0	29	8,463	0	0	0	0	0	0	0	8,506
Amersham	0	10	0	0	318	0	0	0	0	0	0	0	328
Ascot	0	2	0	0	0	0	0	0	0	0	0	0	2
Berkeley	0	23	0	217	0	0	0	0	0	696	171	0	1,107
Bradwell	0	0	0	258	0	0	0	0	0	106	137	0	501
Cardiff	0	0	0	0	119	0	0	0	0	0	0	0	119
Calder Hall	0	3,300	0	1	0	2	0	0	0	0	0	0	3,303
Capenhurst	0	0	0	0	6	0	0	0	0	0	0	0	6
Culham	75	0	0	0	119	0	0	0	0	0	0	0	194
Devonport	0	385	36	1	84	0	0	0	0	0	0	0	506
Dounreay	0	45	0	0	13	0	0	0	0	0	0	0	58
Drigg	0	13	0	0	410	0	0	0	0	0	0	0	423
Dungeness	0	468	76	590	316	0	0	0	0	35	186	0	1,671
Hartlepool	0	164	94	266	73	0	0	0	0	0	0	0	597
Harwell	0	718	0	0	1,145	0	0	0	0	0	0	192	2,055
Heysham	0	159	110	579	198	0	0	0	0	0	0	0	1,046

Location	2m Box	$3m^3$ Box	$3m^3$ Drum	4m Box	500L drum	B/G box	C1 Cont.	C4 Cont.	SF Disp Cont	DCIC cube	DCIC cyl	WAGR Box	TOTAL
Hinkley Point	0	315	160	531	140	0	0	0	0	78	491	0	1,715
Oldbury	0	180	0	368	0	0	0	0	0	62	89	0	699
Portsmouth	0	0	0	0	1	0	0	0	0	0	0	0	1
Rosyth	0	0	42	0	0	0	0	0	0	0	0	0	42
Sellafield	0	68,477	4	0	106,768	1,504	0	0	2,400	0	0	113	179,266
Sizewell	0	1,379	0	279	656	0	0	0	0	23	162	0	2,499
Trawsfynydd	0	188	44	620	1	0	0	0	0	0	0	0	853
Winfrith	0	0	0	0	0	0	0	0	0	0	0	122	122
Wylfa	0	0	0	617	0	0	0	0	0	43	0	0	660
U, Pu, SF	0	0	0	0	191,362	0	0	0	7,292	0	0	0	198,654
New Build	0	960	7,266	60	0	0	6,840	3,240	8,941	0	0	0	27,307

C Radioactive Waste Package Types

Excepted packages: Excepted packages are packages in which the allowed radioactive content is restricted to such low levels that the potential hazards are not significant and therefore no testing is required with regard to containment or shielding integrity. A common example of an excepted package is a package used to carry radiopharmaceuticals for medical purposes.

Industrial packages: Industrial packages are used to transport two types of material:

- Material having low activity per unit mass (known as Low Specific Activity or LSA material). Items classified as LSA material include hospital waste;
- Non-radioactive objects having low levels of surface contamination (known as Surface Contaminated Objects or SCO). Fuel cycle machinery or parts of nuclear reactors, whose surfaces have been contaminated by coolant or process water, are considered as SCO.

Both types of material are inherently safe, either because the contained activity is very low, or because the material is not in a form easily dispersible. Industrial Packages (IP) are sub-divided into three categories designated as IP-1, IP-2 and IP-3, which differ regarding the degree to which they are required to withstand routine and normal conditions of transport (see Table 8).

Table 8 Industrial package requirements

Criteria	IP-1	IP-2	IP-3
Design requirements	■ General requirements for all packages ■ Additional pressure and temperature requirements if transported by air	■ General requirements for all packages ■ Additional pressure and temperature requirements if transported by air	■ General requirements for all packages ■ Additional pressure and temperature requirements if transported by air ■ Type A additional requirements
Test requirements - normal transport conditions		■ Free drop (from 0.3 to 1.2 metres, depending on the mass of the package) ■ Stacking or compression	Each of the following tests must be preceded by a water spray test: ■ free drop (from 0.3 to 1.2 metres, depending on the mass of the package) ■ stacking or compression ■ penetration (6kg bar dropped from 1 metre)

Type A: Type A packages are used for the transport of relatively small, but significant, quantities of radioactive material. Type A packages are required to maintain their integrity during normal transport conditions and therefore are subjected to tests simulating these conditions (see Table 9)

Table 9 Type A package requirements

Criteria	Requirements
Design requirements	■ General requirements for all packages ■ Additional pressure and temperature requirements if transported by air ■ Type A additional requirements (seals, tie-downs, temperature, containment, reduced pressure, valves)
Test requirements - normal transport conditions	Each of the following tests must be preceded by a water spray test: ■ free drop (from 0.3 – 1.2 metres, depending on the mass of the package) ■ stacking or compression ■ penetration (6kg bar dropped from 1 metre)

Type B: Type B packages are required for the transport of highly radioactive material. These packages must withstand the same normal transport conditions as Type A packages, but because their contents exceed the Type A limits, it is necessary to specify additional resistance to release of radiation or radioactive material due to accidental damage.

Type B packages are used to transport material as different as unencapsulated radioisotopes for medical and research uses, spent nuclear fuel, and vitrified HLW. This type of package must be capable of withstanding expected accident conditions, without breach of its containment or an increase in radiation to a level which would endanger the general public and those involved in rescue or clean-up operations. The adequacy of the package to this requirement is demonstrated by stringent accident conditions testing (see Table 10).

Table 10 Type B package requirements

Criteria	Requirements
Design requirements	■ General requirements for all packages ■ Additional pressure and temperature requirements if transported by air ■ Type A additional requirements ■ Type B additional requirements (internal heat generation and maximum surface temperature)
Test requirements - normal transport conditions	Each of the following tests must be preceded by a water spray test: ■ free drop (from 0.3 to 1.2 metres, depending on the mass of the package) ■ stacking or compression ■ penetration 6kg bar dropped from 1 metre
Test requirements - accidental transport conditions	Cumulative effects of: ■ free drop from 9 metres or dynamic crush test (drop of a 500kg mass from 9 metres onto a specimen) ■ puncture test ■ thermal test (fire of 800°C intensity for 30 minutes) ■ immersion (15 metres for 8 hours) Enhanced immersion test for packages carrying a large amount of radioactive material: ■ 200 metres for 1 hour

Type C: The 1996 Edition of the IAEA Transport Regulations introduced a requirement for a more robustly designed package – the Type C Package – to transport the more highly radioactive material by air. Type C packages must satisfy all the additional requirements of Type A packages and most of the additional requirements of Type B packages.

Type C packages are submitted to a series of tests to prove their ability to withstand transport incidents and accidents (see Table 11).

Table 11 Type C package requirements

Criteria	Requirements
Design requirements	▪ General requirements for all packages ▪ Additional pressure and temperature requirements if transported by air ▪ Type A additional requirements ▪ Type B additional requirements (internal heat generation and maximum surface temperature)
Test requirements - normal transport conditions	Each of the following tests must be preceded by a water spray test: ▪ free drop (from 0.3 to 1.2 metres, depending on the mass of the package ▪ stacking or compression ▪ penetration 6kg bar dropped from 1 metre
Test requirements - accidental transport conditions	Test sequence on one specimen in the following order: ▪ free drop from 9 metres ▪ dynamic crush test (drop of a 500kg mass from 9 metres onto a specimen) ▪ puncture test ▪ enhanced thermal test (fire of 800°C intensity for 60 minutes) A separate specimen may be used for the following test: ▪ impact test (not less than 90 metres per second)

Packages for Uranium Hexafluoride: The IAEA Regulations include requirements for packages containing uranium hexafluoride (Hex) which are specific to this material. These packages must meet the following test requirements:

- Withstand a pressure test of at least 1.4MPa;
- Withstand a free drop test – the drop height depending on the mass;
- Withstand a thermal test at a temperature of 800°C for 30 minutes.

Packages for fissile material: Nuclear fuel cycle materials containing enriched uranium or plutonium are fissile, i.e. they can support a chain reaction. Such unwanted chain reactions are prevented during normal and accidental transport conditions by the design of the package, the arrangement of the fissile material in it and also the configuration of multiple packages.