Nirex Report

Specification for Waste Packages Containing Vitrified High Level Waste and Spent Nuclear Fuel
Specifcation for Waste Packages Containing Vitrified High Level Waste and Spent Nuclear Fuel
Conditions of Publication

This technical note is made available under Nirex's Transparency Policy. In line with this policy, Nirex is seeking to make information on its activities readily available, and to enable interested parties to have access to and influence on its future programmes. This document may be freely used for non-commercial purposes. However, all commercial uses, including copying and re-publication, require Nirex's permission. All copyright, database rights and other intellectual property rights reside with Nirex. Applications for permission to use this technical note commercially should be made to the Nirex Business Development Manager.

Although great care has been taken to ensure the accuracy and completeness of the information contained in this publication, Nirex can not assume any responsibility for consequences that may arise from its use by other parties.

©United Kingdom Nirex Limited 2005. All rights reserved

ISBN 1 84029 370 5

Bibliography

If you would like to see other publications available from Nirex, a complete listing can be viewed at our website www.nirex.co.uk, or please write to Corporate Communications at the address below, or email info@nirex.co.uk.

Feedback

Readers are invited to provide feedback to Nirex on the contents, clarity and presentation of this report and on the means of improving the range of Nirex reports published. Feedback should be addressed to:

Corporate Communications Administrator
United Kingdom Nirex Limited
Curie Avenue
Harwell
Didcot
Oxfordshire
OX11 0RH
UK

Or by e-mail to: info@nirex.co.uk
ABSTRACT

Nirex are developing a Reference Repository Concept for the long-term management of UK vitrified high-level radioactive waste and spent civil nuclear reactor fuel.

The Reference Repository Concept, based on the KBS-3V concept developed in Sweden and Finland, assumes that waste packages, in the form of sealed copper canisters containing the waste, are emplaced, surrounded by bentonite clay, in vertical holes drilled along a series of tunnels at a depth of approximately 500m in strong crystalline rock. The provision of robust waste packages therefore provides one of a series of engineered barriers to the release of radioactive material to the environment.

This specification lays out the standards and performance criteria that would need to be satisfied by waste packages containing vitrified high-level waste and spent nuclear fuel to ensure compatibility with the Reference Repository Concept.
PREFACE

This report is part of an ongoing programme of research conducted by United Kingdom Nirex Limited (Nirex) and its contractors. It is a component of the research into one of a number of options for the long-term management of radioactive waste in the UK.

One option for the long-term isolation of radioactive waste from the accessible environment is to place these wastes in a repository excavated in stable rock formations, deep underground (deep geological disposal).

Nirex has developed a multi-barrier concept for a deep repository for the phased disposal of solid intermediate-level and certain low-level radioactive wastes making use of both engineered and natural barriers to form a containment system. A repository would be carefully designed and engineered to provide deep, excavated vaults together with the necessary access ways. Typically, wastes would be packaged in steel or concrete containers, usually with a cement grout, and subsequently placed in the vaults. Some time later, the vaults would be backfilled with a cement-based material, the Nirex Reference Vault Backfill, completely surrounding the waste packages.

Nirex is also considering methods for the long-term management of other radioactive materials, such as vitrified high-level waste and spent nuclear fuel, that could also be the subject of deep geological disposal. To this end Nirex has developed a Reference Repository Concept, based on the KBS-3V concept developed in Sweden and Finland, which assumes that these categories of waste would be placed in sealed thick walled copper canisters which would be emplaced, surrounded by bentonite clay, in vertical holes drilled along a series of tunnels deep underground in a suitable geology.

With both concepts, engineered barriers are provided by the waste package and the chemical properties of their immediate surroundings, whereas natural barriers are provided by the geological formations that surround the repository and that lie between the repository and the accessible human environment.

This waste package specification presents the standards and specifications that would need to be satisfied by waste packages containing vitrified high-level waste and spent nuclear fuel that would ensure their compatibility with the Reference Repository Concept.
EXECUTIVE SUMMARY

Nirex was established in 1982 with an objective of assisting producers of radioactive waste to package those wastes in a form compatible with disposal in a long-term underground repository. Nirex fulfilled this objective for intermediate-level wastes and certain long-lived low level wastes by developing a long-term management concept, the Phased Geological Repository Concept, and by developing standards and specifications for the packaging of those categories of waste, based on that concept. These are defined in the Generic Waste Package Specification.

Historically, Nirex’s remit has been focussed on the long-term management of intermediate-level and low-level waste. However, in line with its current mission, Nirex has now applied the principles adopted in the development of the Generic Waste Package Specification for intermediate-level and low-level waste, to the long-term management of other radioactive wastes and materials that may, at some point in the future, be declared as waste. This document considers two specific categories of such material; vitrified high-level waste and spent fuel from UK civil nuclear reactors. However it is also intended that this document can be used as the basis for the assessment of the suitability of other materials for inclusion in the proposed long-term management concept.

This document defines an outline waste package specification that would allow vitrified high-level waste and spent nuclear fuel to be subject to a process of long-term waste management as defined by a Reference Repository Concept based on the KBS-3V concept developed in Sweden and Finland. It lays out the standards and performance criteria that should be satisfied by waste packages containing vitrified high-level waste and spent nuclear fuel and provides a basis for the further development of the Reference Repository Concept.

This document identifies and defines criteria for complete waste package as well as for the container and the contents that comprise the waste package. As such, it can be used as the basis for the assessment of specific packaging proposals against the requirements of the Reference Repository Concept.

The document also identifies the areas where more information and research is required to allow the standards and specifications to become more defined and to develop in parallel with the Reference Repository Concept.
LIST OF CONTENTS

1 INTRODUCTION 13
  1.1 Scope 14
  1.2 Structure 14

2 BACKGROUND 17
  2.1 The Reference HLW/SF Concept 17
  2.2 Preliminary Assessment and Evaluation of the Reference HLW / SF Concept 19
  2.3 Regulatory Foundations of WPS 19

3 IDENTIFICATION OF WPS CRITERIA 23

4 DEFINITION OF STANDARD WASTE PACKAGES 25
  4.1 Description of Waste Types 25
  4.2 Standard Waste Packages 28

5 SPECIFICATION OF WASTE CONTAINER 33
  5.1 Dimensions and Lifting Feature 33
  5.2 Materials of Construction 35
  5.3 Identification 35
  5.4 Integrity 36
  5.5 Mechanical Performance 38
  5.6 Thermal Performance 38

6 SPECIFICATION OF WASTE PACKAGE CONTENTS 39
  6.1 General Requirements for Waste Package Contents 40
  6.2 Specific Requirements for Waste Package Contents 40

7 SPECIFICATION OF WASTE PACKAGES 45
  7.1 Activity Content 46
  7.2 Gross Mass 46
  7.3 Surface Dose Rate 47
  7.4 Heat Output 48
  7.5 Surface Contamination 50
  7.6 Gas Generation 50
1 INTRODUCTION

Nirex was established in 1982 with an objective of assisting producers of radioactive waste to package those wastes in a form compatible with disposal in a long-term underground repository.

Nirex fulfilled this objective for intermediate-level wastes (ILW) and certain long-lived low level wastes (LLW) by developing a long-term management concept, the Phased Geological Repository Concept (PGRC) [1], and by developing standards and specifications for the packaging of those categories of waste based on this concept. These are defined in the Generic Waste Package Specification (GWPS) [2].

Historically, Nirex's remit has been focussed on the long-term management of ILW and certain long-lived LLW. However, the Nirex mission, as agreed with Defra and the DTI is now:

‘In support of Government policy, develop and advise on safe, environmentally sound and publicly acceptable options for the long-term management of radioactive materials in the UK.’

Accordingly, Nirex now intends to apply principles similar to those adopted for ILW and LLW to the long-term management of other radioactive wastes and materials that may, at some point in the future, be declared as waste. These include:

- vitrified high-level waste (HLW);
- irradiated/spent nuclear fuel (SF);
- separated plutonium;
- highly enriched uranium (HEU);
- uranium arising from nuclear fuel cycle processes\(^1\).

The primary focus of this document is on the first two categories of these materials and, in the case of SF, this is currently limited to fuel from civil nuclear reactors\(^2\). However it is also intended that the document can be used as the basis for the assessment of the suitability of other materials for inclusion in the proposed long-term management concept.

Nirex’s approach to the development of packaging standards and specifications is that, in the absence of an operational facility, such standards and specifications must be generic in that they are:

- derived from a full consideration of all future phases of a defined waste management concept; and

---

\(^1\) The enrichment of uranium and the reprocessing of irradiated SF.

\(^2\) As distinct from fuel from military (i.e. submarine) and research reactors.
• independent of the location of the site of any future repository, which could be implemented at a range of different sites within the UK, representing a range of geological environments.

This waste package specification (WPS) applies that approach to the packaging of HLW and SF which is to be subject to the long-term waste management regime as defined by a Reference Repository Concept for UK HLW/SF (here on in referred to as the Reference HLW/SF Concept [3]). It presents what is required from waste packages to enable safe and environmentally sound management and, whereas it does not define how waste should be packaged, it lays out the standards and performance criteria that should be satisfied by the resulting waste package thus providing a basis for the further development of plans for the packaging and disposal of HLW and SF in the UK.

1.1 Scope

This document applies the principles adopted in the development of the GWPS for ILW/LLW to UK HLW and SF to allow the definition of an outline WPS that would allow such materials to be subject to the process of long-term waste management as defined by the Reference HLW/SF Concept [3].

It should be noted that, as explained in Section 2.1, a number of aspects of the Reference HLW/SF Concept are provisional and that further development of it may result in changes to a number of the waste package parameters and performance requirements that make up this WPS. There will also be a number of aspects of waste package performance that are not currently fully understood and that will therefore require further work, the results of which could have consequences for the WPS. This document therefore also aims to identify those aspects of uncertainty and the work that is required to complete the understanding of the required waste package performance in order that the WPS presented herein can subsequently be fully developed.

1.2 Structure

The remainder of this document comprises:

Section 2 which provides background information on the Reference HLW/SF Concept together with relevant national and international regulations and guidance that have been used in the development of this document;

Section 3 which describes the process for the identification of WPS criteria for waste packages containing HLW and SF;

Section 4 which defines the standard waste packages for HLW and SF;

Section 5 which describes and explains the requirements for waste containers;

Section 6 which describes and explains the requirements for the contents of HLW/SF waste packages;

Section 7 which describes and explains the requirements for waste packages, over and above those dealt with in Sections 5 and 6;

Section 8 which sets down the quality management requirements for waste package development, production and storage, and for the data that need to be recorded for waste packages;
Section 9 which identifies the information that has been identified as being needed to further develop this WPS;

Appendix A which lists the criteria that are used as the basis for a WPS;

Appendix B which defines the identification system for waste packages;

Appendix C which outlines the regulatory and other restrictions that apply to the transport of HLW and SF through the public domain.

Also included is a Glossary of terms relevant to this document, a list of Abbreviations and a list of References.
2 BACKGROUND

This document builds on the work carried out by Nirex for the Radioactive Waste Policy Group (RWPG) to consider the form of the Waste Acceptance Criteria (WAC) for wastes other than ILW and for materials that could be declared as waste in the future [4]. It also responds to issues raised in the BNFL Stakeholder Dialogue Plutonium Working Group regarding a waste form qualification system for plutonium [5].

When radioactive waste is disposed of in an operational facility, waste packages are required to meet WAC which have been produced by the facility operator in conjunction with the relevant regulatory authorities. A similar procedure would be expected to apply for UK HLW/SF that form the subject of this document. WAC for HLW/SF would be based on a specific repository design and geological conditions and would take account of detailed design considerations, finalised safety cases, operational and transport factors and the terms of the disposal authorisation, site licence conditions and statutes in force at the time.

As the repository concept for HLW and SF is at a conceptual stage, the information necessary to develop firm WAC is unavailable. However, to allow the feasibility and safety of the concept to be assessed, and potentially to permit the assessment of packaging proposals for HLW and SF, Nirex is developing a WPS in order to provide industry and regulators with a coherent set of requirements against which waste packages containing HLW or SF can be assessed.

The approach adopted for producing WPS for HLW/SF requires as its starting point the definition of a long-term waste management concept for those materials. This is provided in the form of a Reference HLW/SF Concept [3]. The WPS thus produced will facilitate the manufacture of waste packages that are compatible with all of the aspects and phases of that concept.

The WPS developed in this document should therefore be considered as preliminary WAC and will be developed in parallel with the development of the concept. Ultimately, if this option is selected by the UK Government for implementation, the WPS would lead to WAC that would be used to judge the compatibility, or otherwise, of waste packages with the as built and licensed repository. This approach is consistent with that described in IAEA guidance [6].

2.1 The Reference HLW/SF Concept

In accordance with Government policy [7], the approach promulgated by nuclear industry regulators is for radioactive waste to be converted into ‘passively safe’ forms as soon as is reasonably practicable taking full account of the long-term disposability requirements [8]. For ILW/LLW managed in accordance with the PGRC this generally involves the conditioning of waste into a solid matrix within a stainless steel waste container. The engineered barrier system defined by the PGRC has not been designed to manage wastes with the considerably higher heat outputs typical of UK HLW/SF (i.e. 2-3 orders magnitude higher than that from typical ILW) [9]. For this reason, and for those resulting from considerably more robust nature of the wasteform for HLW/SF, it is clear that the repository concept for such materials will have significant differences from that proposed for UK ILW/LLW.
Internationally, a range of geological disposal concepts have been discussed and investigated for HLW and SF over a number of years. The concepts vary according to the nature and quantity of waste to be managed and the different geological and social settings. Nirex has reviewed this range of concepts and has selected a concept to demonstrate the viability of HLW and SF disposal in the UK. Concepts were screened in order to select a concept that was at an advanced stage of development, based on well-established properties of the engineered and natural containment barrier systems, allowed for ease of retrieval and was supported by extensive R&D.

The Reference HLW/SF Concept [3] selected by Nirex for this ‘viability demonstration’ is based on the Kärnbränslesäkerhet-3 (Nuclear Fuel Safety, KBS-3) concept developed by Svensk Kärnbränsleförsörjning (Swedish Nuclear Fuel and Waste Management, SKB) for spent fuel in Sweden. This concept has been extensively studied by the Swedish and Finnish national programmes for more than 20 years [10]. This selection also reflects the maturity of the Swedish and Finnish programmes, their involvement of stakeholders and their level of regulatory scrutiny and, in the case of Sweden, international peer review.

The KBS-3 concept provides physical containment of SF by packaging within a sealed copper and cast iron container where the copper provides long-term containment of radionuclides and the cast iron, mechanical strength. Under suitable geochemical conditions, the corrosion of copper is extremely slow, and the container is expected to maintain its integrity for an extremely long time. Within the Swedish concept waste packages are deposited in vertical holes, lined with bentonite clay and drilled along a series of access tunnels at a depth of approximately 500m in saturated granitic rock. Following deposition of the waste, the tunnels and associated access caverns would be backfilled with a mixture of bentonite and crushed rock.

The Reference HLW/SF Concept provides a foundation for further design and development. The repository layout and the design of the waste packages have yet to be optimised and this process may result in changes to the dimensions of the waste packages and some of the allowable parameters of the package contents such as heat loading. Such changes would have consequences for the WPS which would be expected to evolve as the Reference HLW/SF Concept is developed.

The Reference HLW/SF Concept assumes the transport of HLW (as vitrified product contained in stainless steel canisters) from Sellafield, and SF (as complete or ‘consolidated’ fuel assemblies) from Sellafield and/or power stations to the repository by road, rail or sea. Following receipt at the repository the HLW/SF would be loaded into waste containers in a dedicated packaging plant, assumed to be provided as an integral part of the repository facilities. The design of waste container is based on that developed by SKB for the disposal of SF and adapted in terms of length, diameter and specific design of cast iron insert, to allow variants to be used for UK designs of SF (i.e. AGR and PWR) and vitrified HLW.

Following packaging, the complete waste packages would be transferred underground and emplaced in vertical deposition holes, surrounded by an enveloping bentonite liner, made from sections, and capped with a concrete floor slab (Figure 1).
2.2 Preliminary Assessment and Evaluation of the Reference HLW / SF Concept

In collaboration with international partners, Nirex has performed a preliminary assessment of the Reference HLW/SF Concept [11]. This has provided the basis for assessment of concept viability, confirmation of the fundamental safety and feasibility of the reversibility of the key emplacement steps, prioritisation of R&D and the development of waste package standards and specifications. Preliminary assessments of operational safety [12] and post-closure performance [13] have been undertaken and a transport safety assessment is being undertaken.

2.3 Regulatory Foundations of WPS

In common with the GWPS, the WPS for HLW/SF is based on a specified concept, and:

- UK Government policy;
- UK legislation, regulations and guidelines;
- EU Directives;
- International guidelines and best practice.
2.3.1 UK Policy, Legislation, Regulations and Guidelines

Government Policy

Current UK Government policy on radioactive waste management is expressed in the 1995 White Paper [7] which states that radioactive wastes should be managed and disposed of in a way which protects the public, the workforce and the environment. The policy is framed within the context of international guidelines and regulations, with the Safety Fundamentals [14] and Basic Requirements [15] documents of the IAEA being specifically highlighted.

On the specific topic of the packaging of ILW, the White Paper states that, where practical and cost-effective to do so, owners and packagers of radioactive waste should ensure that wastes are characterised and segregated, and stored in accordance with the principles of passive safety. The 2004 policy statement on decommissioning [16] states that operators should process wastes in accordance with the Nirex Letter of Compliance (LoC) assessment process. Additionally, the 2005 Regulators’ Guidance to Industry [8] requires that waste packages be produced taking account of disposal requirements.

As defined in the White Paper, a passively safe wasteform is one in which the waste has been made chemically and physically stable, and is stored in a manner that minimises the need for safety mechanisms, maintenance, monitoring and human intervention, and that facilitates retrieval from storage for final disposal. The concepts of passive safety and disposability apply to all radioactive wastes and therefore form the foundations for the WPS that are the subject of this document.

Whilst the above policy and guidance is mainly directed at the conditioning and disposal of ILW many of the principles outlined are equally applicable to HLW and SF. It is therefore assumed for the purposes of this document that the regulatory position for all types of radioactive waste would be fundamentally the same.

In 2001 the Government initiated a consultation process, Managing Radioactive Waste Safely (MRWS) [17] which included consideration of 'materials not currently classified as wastes' and HLW. In the case of the former a policy of early definition of a waste management strategy for these material was encouraged. For HLW no explicit policy for its management exists, beyond a period of at least 50 years of storage to allow heat output to decline. However it was acknowledged that the issues associated with the long-term management of HLW are similar to those of ILW.

Control of Radioactive Wastes

The current requirements under the Radioactive Substances Act 1993 [18] provide the framework for controlling the creation, accumulation and disposal of radioactive wastes so as to protect the public from hazards which may arise from their disposal to the environment. Responsibility for regulation and control under this Act is exercised by the Environment Agency (EA) in England and Wales, and by the Scottish Environment Protection Agency (SEPA) in Scotland. These agencies have published guidance on requirements for authorisation of LLW and ILW disposal facilities on land [19] and, although no such guidance currently exists for HLW or SF, the principles promoted in this guidance

---

3 The LoC process is the means by which Nirex assesses ILW packaging proposals against the requirements of the PGRC to ensure that waste packages will be compatible with the requirements for long-term management.

4 Defined as plutonium, uranium and SF.
can be applied in a similar manner to wastes with higher activities. On the basis of national and international legislation and best practice, the guidance identifies a number of principles and requirements for the management of radioactive wastes. These are relevant to the repository design and operation, and to the systems that support it, including waste packaging.

Radioactive effluents from the repository site would also be regulated under the Radioactive Substances Act, and licensed by the appropriate environment agency.

**Operations**

Operations at civil nuclear sites where waste packages are manufactured and stored are regulated by the Health and Safety Executive (HSE) through HM Nuclear Installations Inspectorate (NII). It is assumed that the repository, as defined in the Reference HLW/SF Concept, would be a licensed nuclear site, for as long as operations continue there.

Operators of designated nuclear installations require a site licence under the Nuclear Installations Act 1965 [20]. This requires licensees to demonstrate that planned operations can be carried out safely, and that appropriate management arrangements are in place to satisfy the conditions of the site licence.

The underlying HSE approach to protecting workers and the general public is set out in its publication ‘Reducing Risks, Protecting People’ [21]. More specific guidance on the approach to safety in nuclear installations is given by the NII Safety Assessment Principles (SAPs) [22]. The SAPs deal with operations at existing types of nuclear plant (i.e. nuclear power plans and fuel cycle facilities), including the immediate operational aspects of waste packaging and storage. Although the SAPs do not explicitly consider the rather different types of operations at a waste repository, in particular the interaction between operational safety and safety in the very long term they are considered to be applicable to all types of nuclear plant. Nirex has derived Nuclear Design Safety Principles (NDSPs) for repository operations [23] which are based on the NII SAPs wherever they are relevant.

**Radiological Protection**

Radiological protection of the workforce and the general public is governed in the UK primarily by the Ionising Radiations Regulations 1999 [24] under the Health and Safety at Work, etc. Act 1974 [25]. These regulations were recently revised to take account of the 1996 EU Directive on Radiological Protection [26].

**Transport**

UK regulations governing the safe transport of radioactive wastes by road and rail have the effect of requiring conformance with the IAEA Transport Regulations [27]. These are an internationally developed set of model regulations that are recommended to be enacted into the national laws and regulations by member states. In the UK, this process involves EU Directives [28, 29] followed by separate regulations for transport of radioactive materials by road [30] and by rail [31].

The Reference HLW/SF Concept assumes that waste packages would be manufactured in a facility that would be an integral part of the HLW/SF repository and does not assume that complete waste packages are transported through the public domain at any stage following manufacture. In this case the IAEA Transport Regulations would not apply to HLW/SF waste packages although, as outlined in Appendix C, they would apply to the transport of HLW and SF to the repository.
2.3.2 International Legislation and Guidance

The influence of international legislation and guidance, in particular those produced by the IAEA, is key in the development of WPS. The main sources relevant to the transport, storage and disposal of radioactive waste are:

- *Regulations for the Safe Transport of Radioactive Material* (i.e. the Transport Regulations [27]); and

- *Characterisation of Radioactive Waste Forms and Packages* [32].

Also of relevance to HLW and SF, despite it primarily dealing with ILW and LLW, is:

- *Containers for Packaging of Solid Low and Intermediate Level Radioactive Wastes* [33].
3 IDENTIFICATION OF WPS CRITERIA

The WPS is the key document defining packaging standards and performance requirements, which ensures that all waste packages are compatible with the repository concept and underlying safety and performance assessments. The approach adopted in this document is to define a WPS that is explicitly and clearly based on the Reference HLW/SF Concept in a similar manner to that adopted for the production of WPS for ILW and LLW in the GWPS [2].

The process of waste management defined by the Reference HLW/SF Concept involves the receipt of vitrified HLW and entire or partially dismantled SF assemblies at the repository site and loading of this material into waste containers, within a dedicated packaging plant (which could be located on the surface or underground) to form disposable waste packages. The waste packages would then be transferred underground and emplaced in deposition holes.

The waste package would therefore be created by the combination of the waste container, comprising a copper canister and cast iron insert, with the HLW or SF contents, by a packaging process as summarised in Figure 2.

Figure 2 Packaging Process for HLW/SF

Nirex has previously carried out an initial consideration of the WAC that would be relevant to the long-term management of HLW and SF [4] and this has been used as the starting point for the identification of relevant WPS criteria. The criteria have been grouped according to whether they apply to the overall waste package or primarily to the waste package contents or the waste container. The subsequent sections of this document deal with the WPS criteria grouped on this basis, structured as follows:

- Section 4: Definition of standard waste packages;
- Section 5: Specification of waste container;
- Section 6: Specification of waste package contents;
- Section 7: Specification of waste package.
Appendix A summarises the process adopted for the identification of the WPS criteria and their allocation to the three areas of waste container criteria, contents criteria and waste package criteria.
4 DEFINITION OF STANDARD WASTE PACKAGES

A waste package consists of a waste container plus the wasteform that is contained within it. In the Reference HLW/SF Concept [3] the waste container consists of a copper outer containment barrier with an inner cast iron structure to support the waste items and provide the waste package with mechanical strength. The wasteform comprises intact or partly dismantled irradiated fuel assemblies, or vitrified HLW.

Standardisation of certain key features of waste packages is important, especially where it enables handling operations to be optimised around a limited number of variants, with consequential benefits in safety, logistics and cost throughout all the phases of waste management.

The outline design for the Reference HLW/SF Concept currently defines dimensions for the three waste package types and these have been used as the basis for this Specification. An initial assessment of the Concept has identified the need to investigate options for optimising the quantity of HLW/SF per waste package and any changes resulting from such optimisation would clearly have an impact on many of the criteria in this Specification.

4.1 Description of Waste Types

The wastes covered by this specification fall into three distinct types:

- HLW – in the form of vitrified HLW (the wasteform) contained in a stainless steel canister;
- PWR Fuel – in the form of complete fuel assemblies;
- AGR Fuel – in the form of consolidated AGR fuels bundles (the wasteform) contained in a stainless steel ‘basket’.

4.1.1 Vitrified HLW Canister

Liquid HLW is converted to a solid, more stable form by immobilising it in a borosilicate glass matrix (the process of vitrification) contained within sealed stainless steel containers known as WVP canisters (Figure 3).

---

5 The process by which fuel ‘pins’ are extracted from AGR fuel assemblies (Figure 4) and placed in stainless steel baskets is known as ‘consolidation’ – See Section 4.1.3.
BNFL have produced a Vitrified Residue Specification [34] which lists the ‘guaranteed parameters’ for WVP Canisters\(^6\).

All WVP Canisters will have nominally identical dimensions and mass although the radionuclide inventories (and resulting heat output, surface dose rate etc) vary significantly depending on the source of the HLW (i.e. from the reprocessing of Magnox or oxide fuel, the irradiation of the fuel, time since irradiation etc). Table 1 lists the basic parameters of WVP canisters.

It is currently assumed that each vitrified HLW waste package would contain two WVP Canisters.

**Table 1**  **Basic Parameters of WVP HLW Canister**

| Containment material | 309 Stainless Steel or equivalent  
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>5mm wall thickness</td>
</tr>
<tr>
<td>Dimensions</td>
<td>433mm diameter x 1347mm high</td>
</tr>
<tr>
<td>Gross Mass</td>
<td>550 kg</td>
</tr>
<tr>
<td>Identification</td>
<td>Unique Identifier on Canister</td>
</tr>
</tbody>
</table>

\(^6\) This reference quoted is for WVP Canisters produced for ‘non-UK Thorp Baseload Customers’ and differs in some respects for UK contracts. Where relevant these differences are reflected in this Specification.
4.1.2 PWR Fuel Assembly

PWR fuel assemblies comprise a square grid of zircalloy clad fuel pins containing UO₂ pellets (Figure 4).

Figure 4 Typical PWR Fuel Assembly

It is currently assumed that PWR fuel assemblies would be transported and packaged essentially intact\(^7\) and that four such assemblies would be contained in a waste package.

Table 2 lists typical data regarding PWR fuel.

Table 2 Basic Parameters of PWR Fuel Assembly

<table>
<thead>
<tr>
<th>Containment material</th>
<th>Zircalloy fuel cladding</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dimensions</td>
<td>240mm square x 4100mm high</td>
</tr>
<tr>
<td>Gross Mass</td>
<td>~700kg</td>
</tr>
<tr>
<td>Identification</td>
<td>Unique identifier assumed</td>
</tr>
</tbody>
</table>

\(^7\) PWR fuel assemblies, as delivered for packaging, will comprise all components between the top and bottom nozzles as shown in Figure 2. It is assumed that the control rod assembly and rod absorber would be removed.
4.1.3 Consolidated AGR Bundle

AGR fuel assemblies comprise a circular array of 36 stainless steel fuels pins containing UO$_2$ pellets and mounted inside a graphite sleeve (Figure 5).

Figure 5 Typical AGR Fuel Assembly

which would involve the removal of the graphite sleeve and the extraction of the fuel pins from the mounting assembly. The fuel pins would then be placed in stainless steel baskets to form ‘consolidated fuel bundles’ each containing ~150 fuel pins. It is currently assumed that eight such bundles would be contained within a waste package.

Table 3 lists typical data for consolidated AGR fuel bundles.

Table 3 Basic Parameters of Consolidated AGR Fuel Bundle

<table>
<thead>
<tr>
<th>Containment material</th>
<th>Stainless steel fuel cladding</th>
</tr>
</thead>
<tbody>
<tr>
<td>Consolidated Bundle Dimensions</td>
<td>240mm diameter x 1000mm high</td>
</tr>
<tr>
<td>Gross Mass</td>
<td>~200kg</td>
</tr>
<tr>
<td>Identification</td>
<td>Basket marked with unique identifier</td>
</tr>
</tbody>
</table>

4.2 Standard Waste Packages

Nirex has defined three standard waste containers that are suitable for the packaging of the three types of waste considered in this Specification. The range of containers is not necessarily fixed and could be added to if additional categories of waste were identified for inclusion within the Reference HLW/SF Concept.
4.2.1 Waste Package for Vitrified HLW

The vitrified HLW waste package is designed to accommodate two standard Waste Vitrification Plant (WVP) canisters as illustrated in Figure 6.

All vitrified HLW waste packages shall comply with the following standards:

- Dimensions within a defined envelope;
- A standardised lifting feature;
- Gross mass not exceeding 15t;
- Defined format and locations for identifier;
- Physical containment provided, for a defined time period, by the waste container body and welded lid.

The standard features of the vitrified HLW waste package are illustrated in Figure 6.

**Figure 6 Standard Features of Vitrified HLW Waste Package**
4.2.2 Waste Package for PWR Spent Fuel Assemblies

The PWR spent fuel waste package is designed to accommodate four intact fuel assemblies as illustrated in Figure 7.

All PWR spent fuel waste packages shall comply with the following standards:

- Dimensions within a defined envelope;
- A standardised lifting feature;
- Gross mass not exceeding 20t;
- Defined format and locations for identifier;
- Physical containment provided for a defined time period by the waste container body and welded lid.

The standard features of the PWR spent fuel waste package are illustrated in Figure 7.
4.2.3 Waste Package for Consolidated AGR Spent Fuel

The AGR spent fuel waste package is designed to accommodate eight consolidated fuel bundles, each containing fuel pins from 3 AGR fuel assemblies enclosed in a stainless steel basket, as illustrated in Figure 8.

All AGR spent fuel waste packages shall comply with the following standards:

- Dimensions within a defined envelope;
- A standardised lifting feature;
- Gross mass not exceeding 12t;
- Defined format and locations for identifier;
- Physical containment provided for a defined time period by the waste container body and welded lid.

The standard features of the AGR spent fuel waste package are illustrated in Figure 8.

Figure 8  Standard Features of Consolidated AGR Spent Fuel Waste Package
5 SPECIFICATION OF WASTE CONTAINER

The following criteria have been identified for the empty waste container which comprises of an outer copper canister (including a copper lid welded in place after waste loading) and an internal cast iron structure or insert. The copper and cast iron components each play separate and distinct roles in the overall performance of the waste package. The primary purpose of the sealed copper outer canister is to provide long-term containment of the radioactivity whilst also providing shielding to reduce external radiation levels. The cast iron insert provides support and location for the contents, mechanical strength, radiation and a heat transfer medium to reduce peak temperatures within the waste package. Such a partition of roles between the two components is useful in identifying the safety function of each and in the process of specifying performance requirements for the complete waste container.

Reference has been made to the SKB work on the bases of design for waste containers [35].

Reference [33] identifies the key parameters for waste containers for ILW and LLW but many of its principles are directly applicable to HLW and SF. These include:

- handling:
  - shape;
  - dimensions;
  - lifting arrangements;
  - impact resistance.

- container durability:
  - concept integrity requirements;
  - container material;
  - manufacturing processes;
  - container closure.

- waste package identification.

Note that in this and subsequent Sections, statements that form part of the Specification are shown highlighted in blue italic script.

5.1 Dimensions and Lifting Feature

Standardisation of waste container design is internationally recognised as good practice [33]. It simplifies handling arrangements and therefore reduces opportunities for error whilst also permitting safe and efficient operation of repository systems.

The key criteria specified are those considered to be essential for container standardisation, namely dimensions, shape, maximum gross mass and lifting features.
5.1.1 Dimensions and Shape

The dimensions of the waste container shall be compatible with the repository provisions for handling and emplacement.

To ensure compatibility with the dimensions and shape of the deposition holes (Figure 1), all HLW/SF waste packages would be cylindrical in shape, without any protrusions, and would have a common diameter of 900mm. The length of the standard waste packages, as defined in [3] and specified in Section 4, have been chosen for compatibility with the dimensions of the three waste types and are summarised in Table 4. The waste package contents, and therefore dimensions, have yet to be optimised and may change as a result of optimisation of the repository system.

Table 4 Dimensions of HLW/SF Waste Containers

<table>
<thead>
<tr>
<th></th>
<th>Vitrified HLW Waste Package</th>
<th>PWR SF Waste Package</th>
<th>AGR SF Waste Package</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overall Height</td>
<td>3200 ± 1</td>
<td>4500 ± 1</td>
<td>2500 ± 1</td>
</tr>
<tr>
<td>Outside Diameter</td>
<td>900 + 0.3/-0</td>
<td>900 + 0.3/-0</td>
<td>900 + 0.3/-0</td>
</tr>
<tr>
<td>Minimum Wall Thickness</td>
<td>50</td>
<td>50</td>
<td>50</td>
</tr>
</tbody>
</table>

All dimensions in mm.

To ensure full compatibility with all repository systems and, in particular, to minimise the presence of air gaps between the waste package and the bentonite following resaturation, tolerances would be specified for waste package dimensions. This would include tolerances on the linear dimensions (i.e. length and diameter) as well as for the ovality of the circular section, flatness and parallelism of the ends and overall straightness of the final waste package.

The tolerances on waste package length and diameter specified above match those quoted for SKB waste packages for PWR/BWR fuel [36].

5.1.2 Lifting Feature

The waste container shall have a standard lifting feature compatible with the repository handling and emplacement equipment.

All HLW/SF waste packages would have the same standard lifting arrangement in the form of a recessed feature in the waste container lid. This would allow all waste packages to be handled in the same manner using identical handling equipment.

The lifting feature should remain serviceable for the operational period\(^8\) of the repository.

---

\(^8\) Defined, for the purposes of this document, as the period until the deposition tunnels are backfilled and sealed.
5.2 Materials of Construction

5.2.1 Copper Canister

*The material for the copper canister shall be pure oxygen free copper of types ASTM UNS C10100 (Cu-OFE), EN133/63: 1994 Cu-OF1 or the equivalent.*

The grade of copper specified for canisters is assumed to be the same as that defined by SKB for the KBS-3V repository concept, the technical considerations and tests of properties that led to this choice are described in [35].

The specification of ‘pure oxygen free copper’ is driven by the requirement for corrosion resistance (see Section 5.4) and the specific grade by the needs of fabrication (i.e. forming and welding) and subsequent ultrasonic testing.

5.2.2 Cast Iron Insert

*The material for the cast iron insert shall be spheroidal graphite cast iron\(^9\) and shall fulfil the requirements of EN 1563 grade EN-GJS-400-15U.*

The grade of cast iron specified for waste container inserts is assumed to be the same as that defined by SKB for the KBS-3V repository concept, the technical considerations and tests of properties that led to this choice are described in [35].

Spheroidal Graphite (SG) cast iron, so named due to the shape of precipitated graphite in the material, has been selected primarily on the basis of its greater strength and resistance to cracking. The material also has good casting and machining properties both of which are important parts of the fabrication process.

5.3 Identification

*The waste container shall be marked with an unique alpha-numeric identifier in a format specified by Nirex.*

*The identifier shall be marked at four equally spaced positions on the vertical surface of the waste container, 100mm down from the top surface, and at the centre of the horizontal surfaces of the waste container lid and base.*

*The identifier shall be capable of being read for the duration of the operational phase of the repository.*

The application of a unique identification marking to each waste container enables the identification and tracking of waste packages throughout all phases of waste management and permits assignment of the appropriate data record.

It is anticipated that the identification system used for HLW/SF waste would be of the same system and format as that defined for ILW waste packages [37]. This system allows for the unique identification of a very large number of waste packages from many identifiable sources, with a format that includes check digits, and allows each waste package to be identified remotely and its number verified by an automatic computer check. The use of a standard character set (OCR-A [38]) with a specified character height (i.e. 6-10mm) allows for remote machine-readability and also allows for direct visual checking.

\(^9\) Sometimes referred to as ‘ductile cast iron’.
Nirex Report N/124

The identifier system, described fully in [39], conforms to that recommended by the IAEA [33], and has been proven by use in several ILW packaging plants in the UK. The system and format for waste package identifiers is summarised in Appendix B.

For remote and/or automatic identifier reading systems to operate effectively, it is important to establish standard locations for the identifiers. For ease of reading during the manufacture and emplacement of the waste package the identifier should be marked in four positions on the vertical surface of the top end of the waste package, for ease of viewing without having to reposition the package, and to provide redundancy to minimise the risk of a package becoming unidentifiable. To provide ease of identification at all stages of waste management (particularly following emplacement) the identifier should also be marked on the horizontal surface at the top of the waste package.

As well as being selected to ensure ease of reading, the positions for the identifiers are also selected so that they do not compromise the integrity requirements of the waste container and are positioned away from welded areas to avoid distortion of the characters.

The marking of the identifier should be sufficiently durable so that it is capable of being read electronically during the operational period of the repository. The method of marking should not compromise the integrity of the waste package containment, accordingly inks containing chloride should not be used. To satisfy both of these requirements, the preferred method of inscribing the identifier is by machining.

5.4 Integrity

The integrity of the waste container and its lifting feature(s) shall be such that it will:

- be retrievable for the duration of the operational phase of the repository, and;
- maintain the containment of its contents for the thermal period\(^{10}\).

Integrity is defined as the ability of a waste container to maintain the containment of its contents, as well as of the surety of physical handling features (i.e. lifting locations), which may be threatened by the onset of corrosion. The basic integrity requirement for HLW/SF waste containers needs to encompass the operational period of the repository so that the waste package enters the post-closure period in good condition.

The Reference HLW/SF Concept acknowledges that retrieval of waste packages is feasible up to the time when the deposition tunnels are backfilled. Backfilling of each deposition tunnel is assumed to occur as soon as all the deposition holes in it are filled although it is anticipated that this backfilling could be deferred if an extended period of retrievability is opted for. The act of backfilling itself is reversible and therefore would not constitute an immoveable obstruction to retrieval if future generations so wished it.

SKB estimate that it would take at least 50 years to emplace all the SF intended for their repository, and plan to proceed in steps that allow for the retrieval of deposited waste to ensure freedom of choice for the future whilst, at the same time, demonstrating their concept for deep disposal on the full scale and under actual conditions. They have also demonstrated the feasibility of the retrieval of emplaced waste packages involving the removal of the bentonite liner by water jetting [40].

---

\(^{10}\) Assumed to be a period of \(\sim 1000\) years.
The requirement for waste package integrity during the ‘thermal period’ of at least 1,000 years, to allow sufficient time for short lived radionuclides to decay, has been defined previously by Nirex for waste packages containing HLW/SF in [4].

The copper canisters currently defined by the Reference HLW/SF Concept are specified to have a wall thickness of 50mm and are expected to maintain integrity for a period well in excess of the thermal period. The post-closure assessment [13] is based on a very low rate of failure with a reference waste container failure time of 100,000 years.

Meeting a specified integrity requirement will require consideration of:

- the canister material;
- the wall thickness;
- method(s) of fabrication and final closure;
- corrosion mechanisms, including internal and external surface finish and the availability of reactants.

The corrosion performance of the grade of copper selected for canisters (Section 5.2.1) is discussed by SKB in [10]. Rates of corrosion of between $6 \times 10^{-3}$ µm/year (for corrosion due to sulphide formation) and 0.125 µm/year (for corrosion resulting from oxygenated water) have been identified. In the SKB design [35] a 50mm canister wall thickness is specified to satisfy the requirement for radiation shielding\textsuperscript{11} although from the corrosion point of view, the wall thickness of the copper layer must be at least 15mm. Applying the higher of the corrosion rates given above to this minimum wall thickness results in a minimum canister ‘life’ of ~120,000 years which accords with the SKB assumption that ‘No known corrosion process will lead to a canister life of less than 100,000 years in a deep repository’ [35].

It is acknowledged that the manufacture of large numbers (i.e. many thousands) of containers will mean that not all will have such a wall thickness at all points and that welds, particularly the lid seal may result in a lesser thickness of containment. In an assessment of the mechanical integrity of the SKB waste container, the Swedish Nuclear Power Inspectorate (SKI) acknowledge that ‘at most one canister in a thousand may leave the encapsulation plant with less than 15mm ligament anywhere in the weld.’\textsuperscript{41}

All of the techniques used during waste container fabrication, including casting, welding and the control of surface finish must be specified and controlled in such a manner as will allow the waste container to satisfy a specified integrity requirement. Control of surface finish is an important protection against the early onset of corrosion and is also important in permitting the effective decontamination of the external surfaces of waste packages to allow the requirements of Section 7.3 to be satisfied.

\textsuperscript{11} The full requirement being for 100mm ‘total metal cover’ of which 50mm is provided by the copper canister and 50mm by the cast iron insert.
5.5 Mechanical Performance

The mechanical properties of the waste container shall be such as to permit the complete waste package to satisfy the mechanical performance requirements of the Reference HLW/SF Concept.

The mechanical demands placed on waste packages during normal and accident situations are outlined in Section 7.7. The waste container, in particular its cast iron insert, will be expected to provide most of the mechanical strength of the complete waste package and of ensuring that it is capable of withstanding normal handling forces without any significant distortion, as well as providing sufficient strength to ensure that the waste package is capable of withstanding the forces that would result from impact accidents identified by the operational safety assessment.

5.6 Thermal Performance

The thermal properties of the waste container shall be such as to permit the complete waste package to satisfy the thermal performance requirements of the Reference HLW/SF Concept.

The contents of waste packages will produce significant quantities of heat which will require the waste container and insert to meet certain thermal performance requirements. Fire accidents would also place demands on the waste container and, although minor by comparison, normal variations in the temperature of the operating environment could also provide a challenge to the final waste package
6 SPECIFICATION OF WASTE PACKAGE CONTENTS

The Reference HLW/SF Concept has been developed to accommodate certain specific inputs, namely SF from UK PWR and AGR power stations, and vitrified HLW. The characteristics of these inputs have dictated certain aspects of the Concept design, including the safety and performance assessments.

This section defines the parameters for the 'acceptable contents' of waste packages on the basis of these materials. If in the future it is decided to consider the inclusion of other materials into the Concept, then they can be judged against these defined acceptable contents criteria.

The contents of the waste packages constitute the 'wasteform' and IAEA guidance that deals with the characterisation of radioactive wasteforms [32] identifies the following key criteria which are relevant to the acceptable contents of waste packages:

- Radionuclide inventory, which will also cover:
  - External dose rate;
  - Heat output;
  - Fissile material content.
- Dimensions;
- Mass;
- Surface contamination;
- Integrity;
- Gas generation.

The material will also need a means of identification and an auditable record of inventory and history.

The physical forms of the HLW and SF currently identified for potential disposal are well defined and these have been used to help define some aspects of the Reference HLW/SF Concept. In particular a common waste container design has been defined which differs only in the cases of overall length and the design of the cast iron insert, both of which are controlled by the size and shape of the anticipated contents.

Information on the materials given in this section is based on that contained within the outline design for the Reference HLW/SF Concept [3] with additional information from other sources, as indicated.

Transport of HLW and SF through the public domain will place restrictions on the nature and quantities of these materials that can be carried in a single transport package. Although outside the terms of this document, these restrictions are summarised in Appendix C.
6.1 General Requirements for Waste Package Contents

In the original consideration of WAC for HLW and SF [4] a number of specific wasteform criteria were identified. The limits placed on these criteria will largely be set by the required characteristics of the final waste package, as defined in Section 7, but it is important that limits and controls are placed on the HLW and SF input material to ensure that, following packaging of the waste, the waste packages will have the required characteristics.

6.1.1 Information required on Waste

To ensure compatibility with the requirements for waste packages described in Section 7, information should be available to describe the following properties of the waste package contents:

- Radionuclide Inventory\(^{12}\);
- Mass;
- Dimensions;
- Quantities on non-radioactive materials.

6.1.2 Properties of the Waste

In order to be compatible with the Reference HLW/SF Concept waste package contents should have acceptable:

- Thermal properties;
- Radiation Tolerance;
- Chemical Durability;
- Physical Properties.

6.2 Specific Requirements for Waste Package Contents

6.2.1 Heat Output

*The heat output of the waste package contents should be such that, at the time of emplacement, the total heat output of the waste package will satisfy the requirements of the Reference HLW/SF Concept.*

*The heat generated by the waste package contents should not cause excessive degradation of the contents themselves in such a way as would result in an early loss of their integrity.*

Both vitrified HLW and SF generate significant amounts of radiogenic heat. As discussed in Section 7.4 the generation of heat needs to be taken into account in both waste package and repository design. The impact of heat on the waste itself also needs to be taken into account as excessive temperatures could result in damage to the containment provided by

\(^{12}\) Sufficient data will be required to allow determination of heat output, fissile material content, and the surface dose rate of complete waste packages.
the waste itself (i.e. the cladding of the SF or the WVP canister). A limit would therefore be placed on the heat generation of HLW/SF received at the repository for packaging.

6.2.2 External Dose Rate

The external dose rate of the waste package contents shall be such that, when the shielding effects of the waste container are taken into account, the external dose rate of the waste package will satisfy the requirements of the Reference HLW/SF Concept.

The high radionuclide inventory of HLW/SF results in significant external dose rates from the unshielded waste package contents. The Reference HLW/SF Concept assumes that the waste container will provide sufficient radiation shielding to allow waste packages to be handled safely following manufacture and, in particular, during emplacement (Section 7.3). The design of the waste container\textsuperscript{13} will make assumptions as to the maximum external dose rate of the waste package contents. Additionally, a limit on the dose rate from of the unpackaged material would be placed by the transport system (Appendix C) and by the design of the repository receipt area and the packaging plant.

6.2.3 Fissile Material Inventory

The fissile material inventory of the waste package contents shall be such that, when the neutronic behaviour of the contents and the waste container are considered, the complete waste package will be capable of satisfying the requirements of the Reference HLW/SF Concept for criticality safety.

The fissile material\textsuperscript{14} inventory of HLW canisters is relatively small whilst that of SF is generally much higher. However, both types of material include large proportions of neutron absorbing U-238 and little or no material that could be considered as significant neutron moderators or reflectors.

An initial consideration of the criticality safety of HLW/SF waste packages [13], using average data for fissile material content, has indicated that there is no risk of criticality if the fissile material remains within the waste package and that criticality is unlikely to occur at any stage during the Reference HLW/SF Concept.

However, significant variation in both the quantities of uranium and plutonium and the proportions of fissile isotopes can be expected for SF depending on fuel burn-up. The extreme cases would be for lightly irradiated enriched fuel which would have little plutonium (although that present would contain a high proportion of fissile Pu-239) but would be relatively highly enriched in U-235 (up to a maximum ‘fresh’ fuel enrichment of ~5%) and for ‘mixed oxide’ (MOX) fuel.

6.2.4 Chemical Durability

The chemical durability of the waste package contents shall be such as to permit the complete waste package to satisfy the long-term containment performance requirements of the Reference HLW/SF Concept.

The physical and chemical nature of the anticipated waste package contents (i.e. vitrified HLW in stainless steel canisters and ceramic fuel in stainless steel/zircalloy cladding; see Section 4.1) are such that their chemical durability would expect to be high and that they

\textsuperscript{13} Note that both the copper canister and the cast iron insert play significant roles in providing radiation shielding.

\textsuperscript{14} Defined as U-233, U-235, Pu-239 and Pu-241.
will provide an important barrier to the early release of radionuclides into the internal voidage of the waste package. Loss of activity from the waste packages, for example by leaching, following waste container failure and subsequent water ingress, would be expected to be at a low rate. The main barrier for the containment of radionuclides in the Reference HLW/SF Concept is the waste container and is based on the very long term integrity of the copper canister (see Section 5.4). However, the assumptions made in the preliminary post-closure safety assessment [13] require a minimum level of for long-term chemical durability and, in particular, leachability performance of the waste package contents.

6.2.5 Materials to be excluded from Waste

The following materials shall be excluded from wastes:

- Combustible Materials;
- Pyrophoric Materials;
- Free Liquids;
- Explosive Materials;
- Toxic Materials;
- Corrosive Materials;
- Compressed Gases.

It is acknowledged that HLW and SF both contain materials from the categories listed above (e.g. uranium is a toxic material which, in finely divided form as a metal or oxide, can be pyrophoric). The materials excluded are therefore those which do not form an integral part of the material primarily intended for packaging (i.e. the HLW/SF). The exclusions would however apply to the presence of those categories of materials that had been generated as a result of any degradation of the HLW/SF prior to packaging.

6.2.6 Gas Generation

Gases generated within the waste package shall not compromise the ability of the complete waste package to meet any of the performance requirements of the Reference HLW/SF Concept.

Gases may be generated internally by the contents of waste packages as well as by components of the waste container such as the cast iron insert. Gases, including radioactive species, can also be released by radioactive wastes. If generated in sufficient quantities, these gases can give rise to a range of potential effects that may have an influence on the performance of waste packages and their influence on both the near-field and far-field (see Section 7.6). This includes the potential for the pressurisation of sealed waste packages, inflation and, in extreme cases, rupture.

The processes by which gases may be generated or released include:

- chemical processes such as corrosion;
- microbial degradation of organic materials;

The long-term integrity requirements of HLW/SF waste packages (see Section 5.4) precludes the use of vented waste containers.
• radiolysis of water and organic materials;
• radioactive decay producing gaseous species (e.g. radon);
• release of gases entrained in waste.

It is assumed that liquids and organic materials will not be present in HLW/SF waste packages and that, prior to final sealing, the waste package will be purged using a dry inert gas to remove oxygen and moisture. Accordingly, only the following gas generation/release mechanisms have been considered:

• radioactive decay of radium to form radon;
• production of helium by \( \alpha \)-emitters;
• release of entrained gases (i.e. the noble gases plus tritium and chlorine, although only the latter two are listed in the inventory for either HLW or SF);
• production of gaseous carbon compounds from C-14 present in the waste and reacting with available hydrogen and/or oxygen.

Table 5 lists the approximate volumetric gas generation rates for the first two gases and the total potential volumes for the latter two are shown below for each of the three waste types on the basis of the radionuclide inventories given in [3].

**Table 5  Gas Generation by HLW and SF**

<table>
<thead>
<tr>
<th>Gas</th>
<th>Source</th>
<th>HLW Canister</th>
<th>PWR Assembly</th>
<th>AGR Fuel Bundle</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radon ((m^3/y))</td>
<td>Radioactive decay of radium</td>
<td>4x10^{-13}</td>
<td>5x10^{-14}</td>
<td>1x10^{-14}</td>
</tr>
<tr>
<td>Helium ((m^3/y))</td>
<td>From (\alpha)-emitters</td>
<td>3x10^{-5}</td>
<td>1x10^{-3}</td>
<td>1x10^{-4}</td>
</tr>
<tr>
<td>Tritium ((m^3\text{ total}))</td>
<td>Entrained in waste</td>
<td>5x10^{-6}</td>
<td>5x10^{-4}</td>
<td>6x10^{-6}</td>
</tr>
<tr>
<td>Cl-36 ((m^3\text{ total}))</td>
<td>Entrained in waste</td>
<td>1x10^{-3}</td>
<td>1x10^{-4}</td>
<td>3x10^{-5}</td>
</tr>
<tr>
<td>CO/CO(_2)/CH(_4) ((m^3\text{ total}))</td>
<td>C-14 entrained in waste</td>
<td>1x10^{-2}</td>
<td>3x10^{-4}</td>
<td>1x10^{-4}</td>
</tr>
</tbody>
</table>
This shows that, in most cases, the rates of gas generation, or the total volume of gas capable of being released by HLW/SF is very small and would not lead to significant internal pressurisation of waste packages. The possible exception to this is the generation of helium by long-lived $\alpha$-emitters in the actinide rich SF. Over the integrity period assumed for waste packages (i.e. $>10^5$ years) this could amount to volumes of helium of the order of $100\text{m}^3$ being released into the relatively small free volume (i.e. $<1\text{m}^3$) within the waste package and internal pressures of the order of 10MPa.

It should also be borne in mind that vitrified HLW is contained within a nominally sealed stainless steel canisters and that both AGR and PWR fuel pins comprise sealed stainless steel or zircalloy tubes containing the UO$_2$ fuel. Accordingly, the loss of a relatively robust initial barrier of each waste component (i.e. HLW canister or fuel pin) would be needed before all of the gas produced by the waste package contents could be released into the free volume within the waste package and cause pressurisation.

In view of the low volumes of gas generated, and the rigidity of the waste container and the relatively large free volume within the waste package, gas generation is not considered a significant issue in the short and medium term.

The effects of gas in the longer term (i.e. post-closure) has been considered [13] with the conclusion that:

- hydrogen, generated by the anaerobic corrosion of the cast iron insert and stainless steel and Zircalloy fuel cladding, will be the only bulk gas generated in significant quantities and, if released from failed waste packages, would not pose a flammability hazard at the surface;

- radioactive gas generation and release from failed waste packages may pose an unacceptable radiological risk only for waste package failure times significantly shorter than the reference failure time of 100,000 years.
7 SPECIFICATION OF WASTE PACKAGES

This Section identifies the criteria applicable to complete waste packages and, where possible, sets limits and/or performance criteria to ensure that waste packages are compatible with the Reference HLW/SF Concept for emplacement and the subsequent stages of long-term waste management. The process for the production of WPS therefore considers the requirements for waste packages imposed by three distinct phases of their management as defined by the Concept:

- short-term - waste package manufacture, transfer underground and emplacement in deposition holes;
- medium-term - the operational (and potential retrieval) phase - prior to tunnel backfilling;
- long-term - the post-closure phase.

For reference in the ensuing sections, Table 6 summarises the basic parameters of the three waste packages as currently defined in outline design for the Reference HLW/SF Concept [3]. It should be noted that these are ‘typical’ values to be used for the purposes of initial assessment. As with all waste streams, variations will occur between individual WVP canisters or SF assemblies and maximum values may be significantly higher than those recorded below. The extent and consequences of such variation is to be explored by Nirex as part of the further work outlined in Section 9.

It should also be noted that optimisation of the repository layout and the design of waste packages may result in changes to the dimensions and the contents of the waste packages.

Table 6 Typical Parameters of HLW/SF Waste Packages

<table>
<thead>
<tr>
<th></th>
<th>Vitrified HLW Waste Package</th>
<th>PWR SF Waste Package</th>
<th>AGR SF Waste Package</th>
</tr>
</thead>
<tbody>
<tr>
<td>Height (mm)</td>
<td>3200</td>
<td>4500</td>
<td>2500</td>
</tr>
<tr>
<td>Diameter (mm)</td>
<td>900</td>
<td>900</td>
<td>900</td>
</tr>
<tr>
<td>Gross Mass (t)</td>
<td>13.2</td>
<td>17.9</td>
<td>10.1</td>
</tr>
<tr>
<td>Total Activity Content (TBq)</td>
<td>$6.3 \times 10^3$</td>
<td>$1.3 \times 10^4$</td>
<td>$3.6 \times 10^3$</td>
</tr>
<tr>
<td>Heat Output (W)</td>
<td>800</td>
<td>1016</td>
<td>432</td>
</tr>
<tr>
<td>Mass of Uranium (kg)</td>
<td>1.6</td>
<td>2060</td>
<td>1000</td>
</tr>
<tr>
<td>Fissile Content$^{16}$ (kg)</td>
<td>0.02</td>
<td>32</td>
<td>9</td>
</tr>
</tbody>
</table>

Waste packages will be manufactured from waste containers meeting the requirements set out in Section 5 and containing contents as specified in Section 6.

The criteria defined in this section are general in nature and will remain so until the parameters of the standard waste packages have been optimised and the key elements of the repository design specified.

### 7.1 Activity Content

The total activity of the waste package shall be restricted to meet the specification requirements for heat output, surface dose rate, criticality safety, impact and fire accident performance and post-closure performance.

Limit on the activity contents of waste packages tend to be derived limits placed on the effects of the total activity content and/or the activity of specific radionuclides to satisfy the requirements of:

- surface dose rate;
- heat output;
- criticality safety;
- impact and fire performance;
- post-closure risk assessment.

The first four of these potential limiting factors are addressed in the relevant sections below. A probabilistic calculation of risk has been carried out using the model developed with SKB [13]. The peak value of the mean annual individual radiological risk was found to be $10^{-11}$, which is substantially below the radiological risk that defines the target applicable to ILW and LLW of $10^{-6}$ per year [19]. This indicates that the currently estimated average activity contents of HLW and SF waste packages are well within any limits that the post-closure risk might impose.

### 7.2 Gross Mass

The gross mass of the waste package shall be compatible with the repository provisions for handling and emplacement.

Mass limits for waste packages will ultimately be set by the mass rating of the repository handling equipment. As currently defined, waste package gross masses range from 10.1t (AGR) to 17.9t (PWR) although these figure may change as the Reference HLW/SF Concept is optimised.

The physical configuration of some waste package contents (in particular PWR fuel assemblies) is such as may lead to and uneven mass distribution within the waste package. This may have consequences for the stability of the waste package during handling and for the design and operation of the deposition machine. Excessive unevenness of mass distribution within waste packages should therefore be avoided.
7.3 Surface Dose Rate

The surface dose-rate of the waste package shall be such as to:

- allow safe handling at all stages of the Reference HLW/SF Concept;
- minimise radiolytic generation of oxygen at the external surface of the waste package.

The shielding provided by the waste container will be insufficient to reduce external radiation levels to allow contact handling of the waste package. The Reference HLW/SF Concept assumes that waste packages will be transferred underground, and to a point above the deposition hole, in a shielded cask comprising of two cylindrical sections joined by a bolted connection. During emplacement the two sections of the flask will be separated and the waste package turned through 90° to allow vertical emplacement.

The design of the emplacement equipment and, in particular, the shielding provided by the cask will be such as to ensure that operator dose is as low as reasonably practicable (ALARP) and that the dose design limit of 20mSv/y (equal to the value defined by the Ionising Radiations Regulations [24] as the maximum annual committed effective dose equivalent for employees on Nuclear Licensed Sites and by the NII SAPs [22] as the Basic Safety Limit) in the Nirex Radiological Protection Policy Manual (RPPM) [42] is not exceeded. This will ultimately place a limit on the surface dose rate of waste packages but this cannot be quantified until the degree of shielding provided by the cask is known.

SKB [43] have adopted a similar approach by specifying a design limit of 5mSv/y for the most exposed personnel (this being 10% of the ICRP effective dose limit of 50mSv/y). This has been used to derive a design goal dose rate of 0.05mSv/h at the surface of the cask (assuming that the most exposed person is involved in a repairing a faulty deposition machine is permitted to receive a dose equal to 20% of the annual limit in an operation lasting 20 hours). Using the same approach, a surface dose limit of 0.2mSv/h would apply to casks in the Reference HLW/SF Concept.

In 1995 SKB [43] determined that waste package surface dose rates would be up to 300mSv/h for typical PWR fuel although SKB anticipate that some fuel types will exceed this. On the basis of the original value SKB have designed a deposition machine cask with 155mm of steel shielding (for $\gamma$-radiation) and 165mm of polythene (for neutrons).

At some point following the emplacement of waste packages in deposition holes the surrounding bentonite liner will become saturated with ground water. An excessive surface does-rate could lead to the generation of oxygen by radiolysis of the water and this could result in accelerated corrosion of the copper canister. The possibility for radiation induced changes in the performance of the bentonite also exists. SKB have concluded that, for the maximum surface dose-rate expected from waste packages containing Swedish SF (i.e. 500mGy/h [44]), radiolytic generation of oxygen will be negligible and effects on the bentonite will be only marginal [40]. The potential for higher dose-rates from UK SF/HLW and differences in the design of the waste container will, however, need to be assessed.
7.4 Heat Output

The total heat output from the waste package shall not exceed that which would result in accelerated deterioration of the contents.

The temperature of the external surface of the waste package shall not exceed 100°C at any time following deposition.

The maximum heat output from the standard waste package, following deposition, shall not exceed:

<table>
<thead>
<tr>
<th>Waste Package</th>
<th>Maximum Heat Output following deposition (W)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vitrified HLW Waste Package</td>
<td>1000</td>
</tr>
<tr>
<td>PWR SF Waste Package</td>
<td>1160</td>
</tr>
<tr>
<td>AGR SF Waste Package</td>
<td>860</td>
</tr>
</tbody>
</table>

It is expected that the great majority of heat generated by waste packages containing HLW/SF will be radiogenic. A small contribution may derive from chemical processes (such as corrosion) and microbiological processes occurring within waste packages but in view of the controlled nature of the waste package contents these contributions are likely to be negligible, particularly in the early stages when radiogenic heat generation is a maximum. Following deposition the waste package will be subject to external sources of heat (i.e. from the host rock) but these are also likely to be small, although the background temperature of the host rock will be a factor in setting waste package heat output limits.

Following manufacture and prior to the completion of the deposition operation the waste package is expected to be stored for a short period of time, loaded into a shielded cask and transported to the deposition location. During this time the temperature of the waste package should not be such that the contents would deteriorate. This will require a limit to be placed on the heat output of the contents which may or may not be more restrictive than that set by post emplacement considerations.

An upper limit of 100°C has been placed on the surface temperature of waste packages in the KBS-3V concept [45]. This limit has been placed to avoid the need to include the effects of the boiling of water in the thermal modelling of the emplaced waste package and, specifically:

- to avoid excessive drying of the bentonite;
- to limit the corrosion rate (the corrosion rate of copper approximately doubles for each 10°C rise in temperature);
- to avoid the enrichment of potentially corrosion accelerating salts at the canister surface.
SKB have used a ‘threshold temperature’ of 80°C in their thermal modelling of waste packages [46]; as a means of allowing for the potential of a 10°C temperature difference across a possible air gap between the bentonite and the waste package surface, and a further 10°C to allow for uncertainties in the thermal properties (conductivity, heat capacity etc) of the bentonite and surrounding rock. However, SKB note that further work is required to develop the treatment of the gap between the buffer and rock and the handling of inhomogeneous thermal rock properties [10]. More work is required to take into account uncertainties for the Reference HLW/SF Concept and the setting of a ‘threshold temperature’ for this concept. This work will also take into account the work currently being undertaken by SKB in this area.

A wide range of heat output is expected for UK SF and vitrified HLW and significant variation on the typical values shown in Table 6 can be expected between individual waste packages. For example, the heat output from SF is approximately proportional to burn-up which results in a variations of a factor of ~2 for PWR fuel ~3 for AGR fuel [47], the values in Table 6 being towards the upper end of these ranges. In the case of vitrified HLW canisters it has been estimated that the heat outputs from WVP Canisters containing HLW from different sources varies from 1900W to 3200W per canister at the time of manufacture and between 400W and 600W after 50 years [34].

SKB have developed a model to calculate the expected thermal transient in a KBS-3 repository [48]. The model has been used to calculate the expected maximum surface temperatures for UK HLW/SF waste packages in the Reference HLW/SF Concept with the external waste package dimensions as defined in [3].

Table 7 shows the predicted maximum temperatures resulting from the three waste package types generating heat at the typical rates listed in Table 6, together with the heat outputs that would result in maximum waste package surface temperatures of 80°C, 90°C and 100°C.

<table>
<thead>
<tr>
<th>Heat Output</th>
<th>Vitrified HLW Waste Package</th>
<th>PWR SF Waste Package</th>
<th>AGR SF Waste Package</th>
</tr>
</thead>
<tbody>
<tr>
<td>86°C</td>
<td>1000W</td>
<td>1160W</td>
<td>860W</td>
</tr>
<tr>
<td>90°C</td>
<td>1020W</td>
<td>1020W</td>
<td>750W</td>
</tr>
<tr>
<td>62°C</td>
<td>850W</td>
<td>1020W</td>
<td>750W</td>
</tr>
</tbody>
</table>

Table 7 Predicted Thermal Performance of HLW/SF Waste Packages after Deposition
7.5 Surface Contamination

The non-fixed contamination on the external surfaces of the waste package shall be kept as low as reasonably practicable, and, when averaged over an area of 300 cm$^2$ of any part of the surface of a waste package, should not exceed the following:

- 4.0 Bq cm$^{-2}$ - beta, gamma and low toxicity$^{17}$ alpha emitters;
- 0.4 Bq cm$^{-2}$ - all other alpha emitters.

Minimisation of the surface contamination is considered good practice. It is also important that waste packages are produced and maintained with the minimum of surface contamination, in order to avoid the spread of contamination to various parts of the repository and to the environment in general. Such a regime will help ensure the minimisation of doses to workers and members of the general public.

Limits are specified to control surface contamination to realistic and achievable levels, to reduce any requirement for the decontamination of waste package handling areas and help ensure that doses to workers and members of the general public are ALARP.

The limits specified are intended to control surface contamination to realistic and achievable levels. They will reduce any potential requirement for the decontamination of waste package handling areas, as well as the requirement to decontaminate the internal surfaces of reusable transport containers during turn-round maintenance.

The specified surface contamination limits are in accordance with the limits specified for a controlled contamination area in Nirex category C1 [49], which complies with the requirement to minimise the extent of any contamination controlled areas [50]. They are also in accordance with IAEA recommendations [33] for radioactive waste packages.

7.6 Gas Generation

Gas generated by the waste package shall not compromise its ability to meet any of the requirements of the Reference HLW/SF Concept.

Gases generated by the waste package contents can give rise to a range of potential effects that may have an influence on the Reference HLW/SF Concept. These could include:

- pressurisation leading to distortion and/or damage to the waste container;
- releases of radioactive/toxic/flammable gases from packages;
- pressurisation and damage to the bentonite liner;
- changes to the chemical characteristics of the bentonite liner.

As shown in Section 6.2.5 the expected total volumes and rates of generation of gases, released from HLW and SF by a number of different mechanisms, are expected to be very small and are unlikely to result in any of the deleterious effects listed above.

---

$^{17}$ Defined as natural uranium; depleted uranium; natural thorium; uranium-235 or uranium-238; thorium-232; thorium-228 and thorium-230 when contained in ores or physical and chemical concentrates; or alpha emitters with a half-life of less than 10 days.
7.7 Mechanical Performance

*The waste package shall be capable of withstanding all credible mechanical challenges that the Reference HLW/SF Concept may impose, without any change that would render it incompatible with any of the requirements of this Specification.*

Mechanical demands will be placed on the waste package under normal and accident conditions. Under normal conditions the waste package is required to perform in a manner compatible with the Reference HLW/SF Concept. In the event of the waste package being exposed to mechanical accidents (i.e. impacts) the release of contents must be such that on- and off-site doses are within the targets specified by the relevant regulations.

7.7.1 Normal Mechanical Challenges

*The waste package shall be capable of withstanding a pressure of 14MPa applied normally to its external surfaces whilst maintaining the defined dimensional envelope and without loss of integrity for the duration of the operational period of the repository.*

During normal operations the most significant forces on the waste package will result from hydrostatic pressure and the pressure exerted by the swelling of the bentonite liner following re-saturation. In the KBS-3V Concept developed by SKB the values used for both of these pressures is 7MPa (assuming a maximum repository depth of 700m) giving a total of 14MPa [35]. The waste package must be capable of withstanding these forces without loss of integrity over the timescale defined in Section 5.4.

7.7.2 Impact Performance

*The waste package should be designed such that, in the event of an impact accident, the release of radioactive material is low and predictable, exhibits progressive behaviour with increasing impact severity and does not exhibit significant ‘cliff-edge’ performance characteristics within the anticipated range of impact conditions.*

The waste package shall be capable of withstanding normal handling, including minor impacts etc, and remain suitable for safe handling during all subsequent phases waste management as defined by the Reference HLW/SF Concept.

*The waste package shall be capable of being dropped, in any attitude, from a height of 5 metres onto an unyielding surface, whilst retaining its radioactive contents.*

*The waste package shall be capable of being dropped, in any credible attitude, from a height of 10 metres onto an unyielding surface, with a loss of contents as particles <100µm that should not result in an on-site dose of greater than 1Sv nor an unprotected off-site does of greater than 100mSv.*

Mechanical challenges such as impact accidents represent potential mechanisms by which the radioactive contents of waste packages can be released into working areas and beyond the repository site boundaries in an uncontrolled manner. They also could lead to damage to waste packages that makes them unsuitable for further safe handling and/or emplacement and lead to a requirement for return to the packaging plant for reworking.

---

18 ‘Cliff edge’ performance characteristics imply that a small adverse change in conditions would produce a major deterioration in package performance.
Three impact challenges are defined:

- A ‘minor impact’, following which the waste package remains suitable for onward waste management;
- A ‘significant impact’, which results in no loss of containment, but following which the waste package may require remedial work to restore its condition;
- A ‘Design Basis Accident (DBA)’ impact, which potentially results in a loss of containment and contents and dose to workers and the general public.

**Minor Impacts**

Waste packages should be robust to minor impacts suffered in the course of normal handling operations following manufacture and until emplacement is completed. Following such impacts the waste package should continue to be compatible with normal lifting and handling arrangements.

**Significant Impacts**

The inherently robust nature of HLW/SF waste packages and their contents means that it is not unreasonable to expect them to be able to withstand the effects of impacts from significant heights (i.e. a few metres) without any loss of containment and release of contents. A drop height that constitutes a ‘significant’ impact would be identified by the operational safety assessment and could, for example, correspond to the effective height that a waste package could fall if it toppled from a vertical to a horizontal position at some stage following manufacture and prior to emplacement. In the case of the tallest waste package (i.e. the PWR waste package) this height is 4.5m and therefore a bounding height of 5m would be seen as bounding for all waste packages. Following such an impact, damage to the waste package, such as damage to the lifting feature that would render it unusable, may make it unsuitable for continuing safe long-term waste management and may require reworking or repackaging.

Modelling work has been carried out by Posiva\textsuperscript{19} [51] to investigate a range of repository impact accidents on KBS-3 type waste packages containing Boiling Water Reactor (BWR) fuel. This work showed that, for a waste package with a gross mass of 24t (i.e. $\sim \frac{1}{3}$rd greater than the PWR waste package), a square drop, lid down, onto an unyielding surface from a height of 0.5m, led to plastic strains that would render the waste package ‘useless for disposal’ although no breach of the copper canister would result.

The Posiva work also included a consideration of the effects of the impact resulting from the toppling of a BWR waste packages (height = 4.8m). This showed that ‘large plastic deformations’ of the copper canister would result but without any breach of containment, when such a waste package topples on to a flat unyielding surface. However, if the top of the waste package were to impact a rigid beam on the floor, breach of the canister and the potential for the release of contents would be expected.

\textsuperscript{19} Posiva Oy is the organisation responsible for the characterisation of sites for the final disposal of SF in Finland, and for the construction and operation of the Finnish SF repository.
**Design Basic Accident Impacts**

Following their manufacture waste packages will be subjected to a series of operations leading to their emplacement in deposition holes and the possibility for mechanical damage exists at several stages, both before and after emplacement. As part of the operation safety assessment, a Design Basis Accident (DBA) analysis will be carried out to identify the opportunities for package damage due to impacts, and to assess the radiological consequences, both to workers on-site and members of the public off-site.

In the case of ILW waste packages the Generic Operational Safety Assessment (GOSA) [52] identified a number of mechanical challenges to waste packages that were considered DBAs and assigned the impact resulting from a 25m drop as being their equivalent in terms of potential damage to waste packages. This value was derived from historical ILW repository design and has been retained to provide a margin of conservatism for the maximum actual drop height whilst also allowing for other types of impact (e.g. roof collapse, or dropping of waste packages and/or transport containers on to other waste packages). It also offers the assurance that impact performance would not be considerably worsened by only a small increase in accident conditions beyond the specified limit (so-called ‘cliff edge’ effects).

In the case of the Reference HLW/SF Concept the opportunities for waste packages to be dropped from heights of this magnitude (i.e. 25m) are more limited, as is the possibility of waste package damage due to other types of impact. A consideration of the scenarios for impact accidents in the operational safety assessment [12] has identified the accidental dropping of a waste package into a deposition hole during emplacement as the most extreme credible impact accident. As shown in Figure 1, the maximum depth of a deposition hole (assuming that the bentonite liner had not been emplaced) is 7.55m. Including an allowance for the height above repository floor level from which a waste package could accidentally fall leads to a bounding height of ~8m for a DBA impact accident which is conservatively increased to 10m, for the purposes of this Specification, to be representative of all DBA mechanical challenges to which HLW/SF waste packages could be subjected.

In accordance with NII SAPs [22] and the Nirex's NDSPs [23], limits are placed on the dose consequences of such an accident. Specifically, SAP P25 requires that:

‘….following any design basis fault sequence:

- none of the physical barriers to the escape of radioactivity is breached or, if they are, then at least one barrier remains intact;

- there is no release of radioactivity except in the most severe cases and, even then, no person outside the site will receive a dose of 100mSv or more; and

- no person on the site will receive an excessive dose from the release of radioactive material or by direct radiation including that from criticality incidents.’

---

20 It is important to note that the SAPs are currently in the process of review and that this will result in DBA dose limits being linked to the risk of such accidents. This may result in the dose limits quoted here being reduced.
The final operational safety assessment for a HLW/SF repository will present a detailed analysis of the radiological consequences of all DBAs that would be liable to result in mechanical damage to waste packages, against the dose requirements of SAP P25. Implicit in this analysis will be the assumption of specified minimum standards of waste package impact performance. In order for the analysis to be valid, it is therefore vital that waste package impact performance specifications exist and can be shown to be met by all packages.

7.8 Thermal Performance

The waste package shall be capable of withstanding all credible thermal challenges that the Reference HLW/SF Concept may impose, without any change that would render it incompatible with any of the requirements of this Specification.

Thermal demands will be placed on the waste package under normal and accident conditions. Under normal conditions the waste package is required to perform in a manner compatible with the Reference HLW/SF Concept. In the event of the waste package being exposed to accidental thermal (i.e. fire) conditions the release of contents must be such that on- and off-site dose are within the targets specified by the relevant regulations.

7.8.1 Normal Thermal Challenges

The waste package must be capable of withstanding the effects of the variations in temperature that will occur during normal repository operations without any significant distortion that could result in the waste package not being able to be emplaced in the bentonite liner or that would result in significant gaps between the waste package and the liner following emplacement.

7.8.2 Fire Performance

The waste package should be designed such that, in the event of a fire accident, the release of radioactive material is low and predictable, exhibits progressive behaviour with increasing impact severity and does not exhibit significant ‘cliff-edge’ performance characteristics within the anticipated range of fire conditions.

The waste package should be capable of withstanding a fully engulfing, 1000°C hydrocarbon pool fire of 1 hour duration, with a release of contents that should not result in an on-site dose of greater than 1Sv nor an unprotected off-site dose of greater than 100mSv.

As in the case of impact accidents, fire accidents represent a potential mechanism by which the radioactive contents of waste packages can be released into working areas and beyond the repository site boundaries, in an uncontrolled manner.

In deriving criteria for fire performance it is necessary to assign appropriate fire conditions (i.e. flame temperature and fire duration) and targets for what is considered to be an acceptable activity release following such an accident. Flame temperature is affected by a number of variables, but the dominant one is the type of fuel (i.e. the flammable and/or combustible material) involved in the fire. Duration, on the other hand, is dependent on factors such as the quantity of fuel involved, relative quantities of flammable and

---

21 For ILW safety assessments, ‘excessive dose’ has been interpreted as 1Sv resulting from the effects of suspendable particles (i.e. <100µm aerodynamic diameter) only, and taking no account of external radiation from the released material.
combustible materials, their surface areas, the size of the pool resulting from a liquid fuel accident, the fire-fighting measures and their effectiveness, etc.

Nirex has commissioned work to consider the nature of fires in underground repositories [53] and, in the case of the PGRC, this led to the definition of a DBA fire as having a maximum flame temperature of 1000°C and a duration of 1 hour. The validity of assuming the same fire challenge for the Reference HLW/SF Concept is still to be reviewed but, in the absence of information to indicate that this will be less severe than for the PGRC, the operational safety assessment [12] assumes the same criteria for the DBA fire for HLW/SF waste packages.

As DBAs, the same limits on the dose consequences that applied to DBA impacts also apply to repository fires; not more than 100mSv to a member of the public off-site, and not more than 1Sv to a person on-site. Hence waste packages must be capable of withstanding the specified fire without a release of activity that would breach these limits.

Posiva have carried out modelling work to investigate the effects of fires on KBS-3 type waste packages [54]. This has indicated that, with flame temperatures in the range 1000-1200°C, a fire duration of 2 to 3 hours would be necessary to cause waste package contents temperatures in excess of 570°C (this being considered the minimum temperature at which damage to fuel cladding could occur). No mention is made in the work of any breach of the waste package.

7.9 Criticality Safety

The waste package shall be designed to preclude nuclear criticality, singly or when in arrays with other waste packages.

The presence of fissile materials, neutron moderators and reflectors in the waste package shall be controlled to ensure that it does not present a criticality safety hazard at any stage of the Reference HLW/SF Concept.

A preliminary criticality assessment has been carried out [13] and concludes that, for intact waste packages containing the average inventory defined in [3]:

- HLW waste packages contain relatively small quantities of fissile material and could be considered ‘benign’ from the viewpoint of criticality safety during all phases of the Reference HLW/SF Concept;
- Individual PWR waste packages could be demonstrated to be sub-critical by using calculations similar to those carried out by SKB;
- AGR waste packages contain insufficient fissile material (i.e. ~1/3rd of the minimum critical mass for ~1% w/o U-235) to constitute a criticality hazard.

The long term behaviour of fissile material (i.e. following the loss of integrity of the waste package) was also considered and it was felt that the possibility for the redistribution/accumulation of fissile material from a number of waste packages to form a critical assembly would be low but would need to be considered by further analysis.

In view of the expected significant variation in both the quantities and relative proportions of uranium (i.e. enrichment of U-235) and plutonium isotopes in SF it is believed that the preliminary assessment offers no more than an indication that typical waste packages will not constitute a criticality safety concern. More detailed work is required to define specific limits on the quantities of fissile material that will be allowable in the different types of waste
Nirex Report N/124

packages containing SF. This work would also be required to determine limits on fissile materials for waste packages individually and in arrays, and when damaged as a result of an impact accident. It should also include a consideration of the evolution of waste packages and the redistribution of fissile material to ensure continued sub-criticality well into the post-closure phase of the repository.

The results of the work should form the basis of a formalised method of ensuring the long-term sub-criticality of waste packages, such as that used in the production of Criticality Control Assurance Documentation (CCAD) for ILW/LLW waste packages containing fissile material.

7.10 Nuclear Security Physical Protection

The nature and quantities of nuclear material contained within the waste package shall be limited such that it does not require protection to a higher category than that defined for the repository as a whole.

The Nuclear Industries’ Security Regulations (NISR) 2003 lays down the approvals required for the physical protection of nuclear materials in transit between licensed sites, against the risk of theft or sabotage. They are administered and enforced by the Office for Civil Nuclear Security (OCNS) acting on behalf of the Secretary of State for Trade and Industry.

The category of the repository will be defined in the LoC Security Plan that will be produced by Nirex, for approval by OCNS, when a decision has been made as to which types of nuclear material will be packaged and held there.

HLW is defined by the NISR as ‘irradiated nuclear material’ and, as such, no mass limits are placed on the quantities that can be carried in waste packages when protected accordingly to Category III standards. The same is true of waste packages containing SF with an unprotected dose-rate of >1Gy/hr at 1 metre. Waste packages containing SF without such ‘self-protection’ (i.e. lightly irradiated fuel) would require protection to a higher standards; to Category I if they contained more than 2kg Pu per waste package or Category II if they contained more than 0.5kg Pu.

7.11 Safeguards

The safeguards status of any fissile or source materials (i.e. isotopes of uranium, plutonium and thorium) contained within the waste package shall be ascertained and procedures developed to ensure that they are appropriately recorded.

Wastes containing isotopes of uranium, plutonium or thorium derived from the UK civil nuclear programme are likely to be subject to international safeguards and as such, reported to the relevant safeguards authorities. It is the responsibility of the custodian or intended recipient of the waste to ascertain its safeguards status and to agree with national and international safeguards authorities the process for recording and reporting. Specific safeguards approaches will be agreed for waste packages based on the proliferation risk and the storage or disposal regime but, in principle, where materials are subject to safeguards, it is likely that they will continue to be subject to controls during all stages of repository operations up to and beyond emplacement.

22 ‘Nuclear material’, as defined by OCNS, includes plutonium and enriched uranium.
The implication of the application of safeguards measures will have to be considered in greater detail, particularly if retrievability of the waste containers is an integral part of the Reference HLW/SF Concept.

7.12 Quality Management

*Quality management shall be applied to all aspects of the packaging process that affect the quality of the waste package product.*

It is important that confidence exists in the quality of all components involved in the production and subsequent treatment of the waste packages produced at the repository. This will extend to all components of the waste package (i.e. the waste container and the waste package contents) and any relevant processes involved in their manufacture and subsequent history.

This will require the establishment, implementation and maintenance of a formal and effective Quality Management System (QMS), with the objective of assuring product quality and data records for the packaged waste. The QMS should apply to all activities that can affect the product quality of the packaged wastes and should comply with ISO9000.

A full description of the quality management requirements for waste packages can be found in Section 8.1.

7.13 Data Requirements

*Information shall be recorded on all relevant details of the waste, waste container and of the manufacture and subsequent treatment of the waste package during all stages of the Reference HLW/SF Concept.*

Information must be recorded for each waste package to enable conformance with the necessary performance criteria to be demonstrated for future phases of waste management as defined by the Reference HLW/SF Concept.

This information will produce a data record that will fully describe each waste package including:

- the physical, chemical and radionuclide inventory of the contents of each waste package together will all relevant details of original manufacture and subsequent history (i.e. irradiation, storage conditions, post-irradiation treatment and processing including any instances of non-standard treatment etc);
- the manufacturing processes and subsequent history of the waste container (i.e. the copper canister and lid and cast iron insert);
- the manufacturing processes and subsequent history of the waste package.

This will require the establishment of a system for acquiring, recording and subsequently managing this information for each waste package.

A full description of the data requirements for waste packages can be found in Section 8.2.
8 QUALITY MANAGEMENT AND DATA REQUIREMENTS

It is necessary to be confident that each waste package will be acceptable for emplacement and subsequent long-term management. The quality management and data requirements given in this Section provide a major component of this assurance.

8.1 Quality Management for Waste Packaging

As described in Section 3, it is envisaged that the waste packaging plant will be an integral part of the HLW/SF repository and the quality management and data requirements that follow primarily apply to that plant. Quality management and data requirements will also apply to the consigners of wastes to the repository and these will be specified and controlled by way of the repository WAC that will be developed in the future.

Processes shall be established and implemented for the packaging of radioactive wastes which encompass the whole lifetime of all components of the waste package contents and the waste container, and of the complete waste package, to ensure that packaged waste has the properties ascribed to it. These arrangements should be reviewed periodically and adequate records maintained. Persons and organisations responsible for verifying these processes should have appropriate authority and independence.

8.1.1 Quality Management System

A formal and effective Quality Management System (QMS) shall be established, implemented and maintained with the objective of assuring the quality of both the waste package product and the associated data records.

As a minimum, the QMS shall comply with BS EN ISO9001 [55].

The QMS should apply to all activities, interactions and aspects that can affect the quality of the waste package product.

8.1.2 Waste Product Specification

A Waste Product Specification (WPrS) shall be established and maintained for each waste package type. This shall fully define:

- the waste package contents, including original manufacture and subsequent history (irradiation, processing, storage etc);
- the waste container, including manufacturing process(es) and subsequent history;
- the manufacturing processes for the final waste package;
- all relevant supporting R&D.
8.1.3 Demonstration of Effectiveness and Verification

*It shall be demonstrated, by providing objective evidence, that:*

- the QMS employed applies to all stages of the process from initial design through to final packaging and emplacement;
- the waste is being packaged in compliance with the QMS and the WPrS; and
- the implementation of the QMS and compliance with the WPrS are verified by independent audit or assessment.

8.1.4 Assessment

*Access for the assessment of activities that affect the quality of waste package products shall be provided to regulators or their nominated representatives, upon reasonable request.*

8.2 Data Requirements

This section identifies the information that will need to be made available by the waste consigner and/or generated during the process of waste package manufacture to allow each final waste package to be fully described and assessed as to its suitability for long-term management as defined by the Reference HLW/SF Concept.

Some of the required information will be historical, having been generated at the time of original manufacture of the material through irradiation, storage and subsequent processing of the material etc. Other information will be acquired during the process of waste package design and performance assessment, testing etc. Finally, following receipt of the material to be packaged at the repository, information will be generated during waste container production and transport, waste package manufacture and subsequent emplacement.

8.2.1 Aims

*The overall aims of the data requirements are:*

a) Data shall be recorded for each waste package, including its contents and all components of the waste container;

b) Each waste package shall be readily identifiable, and shall be linked to data recorded about that package, its contents and container, and also to the WPrS with which it complies;

c) The recorded data shall:

- facilitate tracking of the location and status of each waste package at all times;
- provide verification of conformance of a package with the relevant LoC submission and WPrS;
- enable demonstration of conformance with facility WAC;
- facilitate provision of the disposal record.
Of particular significance is a realistic and justifiable record of the nature and contents of each package which:

- covers the physical, chemical and radionuclide content;
- identifies, or enables prediction of, package properties and performance;
- allows prediction of the evolution of the package characteristics with time, and of the effect of interactions with other repository components.

8.2.2 Implementation

The data recording system needs to cover the history of the waste package contents, the waste container and the final waste package.

The data that will need to be available for each waste package shall include:

- the waste package identifier;
- the origin and history of the waste package contents from the time of original manufacture, through irradiation (where relevant), subsequent processing and storage, transport to the repository site and on-site storage;
- the origin and history of all components of the waste container, including full specifications, manufacturing details and history of all of its components;
- details of the waste package design and manufacturing processes;
- the waste package radionuclide inventory;
- the waste package non-nuclear contents inventory;
- waste package properties, including gross mass, surface dose rate, surface contamination levels, heat output etc;
- the Waste Product Specification;
- a full waste package history, including production, storage, emplacement and post-emplacement history;
- criticality safety information, including CCAD, appropriate criticality safety cases, declaration and verification of compliance with the CCAD;
- administrative information, including original waste owner contact and authorisation and agreement for transfer of waste package for disposal.
- international Safeguards information including Safeguards category, material description, mass of fissile isotopes etc.
9 INFORMATION NEEDED FOR FURTHER DEVELOPMENT

This section identifies what information is needed to allow further development of the criteria contained in this document in order that they can be developed to the same level as those defined for ILW and LLW by the GWPS.

Nirex has previously considered the scope of the R&D work that would be required to support the development of the Reference HLW/SF Concept [56] and this section should be seen as being in addition to those requirements. The key challenges relating to the Reference HLW/SF Concept are also described in Nirex Report N/122 [57].

9.1 Definition of Waste Package Contents

The physical description of vitrified HLW canisters is currently well defined but this is not the case for SF. It is currently assumed that AGR SF will be 'consolidated' at some point prior to transport to the repository, information regarding the nature of the product of the process of fuel consolidation is required. Information is also required regarding the state of PWR SF as received at the repository especially as it may affect the mass distribution of waste packages.

9.2 ‘Out of Specification’ Waste Package Contents

This Specification has only considered waste package contents in a condition which may considered ‘standard’ (i.e. vitrified HLW in canisters which accord fully with the BNFL Vitrified Residue Specification [34] and PWR/AGR fuel which is undamaged and the dimensions of which conform to the original specification). ‘Out of specification’ material (i.e. damaged AGR/PWR fuel, other types of fuel, HLW canisters not in full compliance with the BNFL specification) will also need to be subject to long-term waste management. A definition of the likely extent to which waste package contents may be ‘out of specification’ will be required.

9.3 Confirmation of Waste Package Descriptions

The outline design for the Reference HLW/SF Concept [3] currently defines dimensions for the three waste package types and these have been used as the basis for this Specification. These waste package designs are based on designs developed in some European countries or previously investigated by British Energy. The process of concept development is iterative, the initial assessment of the Reference HLW/SF Concept identified the need to investigate options for optimising the quantity of HLW/SF per waste package. Any changes to waste package dimensions and contents would have a significant impact on many of the criteria in this Specification.

9.4 Waste Package Dimensions

Some basic tolerances are defined for waste package dimensions in this specification. However, in order to ensure full compatibility with repository systems, particularly during waste package emplacement, a determination of the acceptable tolerances for all of the key dimensions (i.e. length, diameter, straightness, ovality of container, dimensions of lifting feature) for each of the waste package designs will be required.
9.5 Waste Package Mass

The masses currently specified in this document are actual masses determined for the three waste package types in the outline design for the Reference HLW/SF Concept [3] and do not represent limits on waste package mass. Such a limit will be set by the capabilities of the repository lifting equipment and the deposition machine and will be applied to all waste packages, irrespective of contents.

9.6 Waste Container Integrity Requirements

A number of waste container integrity assumptions have been made in both the Nirex [13] and the SKB [44] post-closure assessments. The specification of waste container integrity requirements are an important part of a WPS and a review of the waste package integrity requirements for the Reference HLW/SF Concept is necessary to confirm these.

9.7 Activity Content

Significant variation exists in the likely radionuclide inventories of the three waste package types. This variation has significant effect of the definition of a number of WPS criteria including surface dose rate, heat output, gas generation, criticality safety and impact and fire accident performance. Work should be undertaken to define the maximum radionuclide inventories, in particular for fissile species, for the three waste types.

9.8 Heat Output

More work is required in this area to take into account uncertainties for the Reference HLW/SF Concept and to demonstrate the robustness of the 100°C limit for waste package surface temperature.

Section 7.4 shows how the SKB thermal model can be used to set heat generation limits for waste packages. The model has been developed for the specific internal configuration of SKB waste packages (i.e. PWR and BWR fuel) but can be applied to waste packages with the external dimensions of the UK waste packages to gauge their likely response to internally generated heat. However the model will need to be modified to allow the internal configuration of UK type waste packages to be more closely matched and to allow definitive heat limits to be derived.

Table 7 shows the dependence of the heat output limit on the maximum allowable waste package surface temperature and also that the limits derived from the model may place too restrictive limits on the heat output of waste package contents with the existing configuration. Heat output is clearly an important parameter in waste package and repository design and early consideration of it will be needed.

Thermal modelling of the waste package prior to emplacement will be required to ensure that the waste package surface temperature does not exceed that which would cause problems during handling and that the temperature of the contents does not lead to their damage.
9.9 Surface Dose Rate

As discussed in Section 7.3, the maximum allowable surface dose rate of waste packages will, in part, depend on the shielding provided by the deposition cask used to transfer waste packages underground and during deposition, together with the dose consequences of the mode of operation of the transfer/deposition equipment.

In order to quantify the acceptable surface dose rate of waste packages the following information is required:

- the highest surface dose rates of HLW canisters, PWR fuels assemblies and consolidated AGR fuel bundles;
- the shielding provided by the waste canister;
- the acceptable surface dose rate of the deposition cask;
- the design of the deposition cask.

Information is also required on the potential for the radiolytic generation of oxygen from groundwater at the surface of waste packages and the consequences for waste package integrity.

9.10 Gas Generation

Section 7.6 has shown that the generation of gases by most mechanisms by waste package contents is likely to be insignificant. The possibility of the production of significant quantities of helium by SF was identified. Further work is required to identify the likely extent of helium generation and the possibility of resulting excessive waste package pressurisation.

9.11 Impact and Fire Performance

The requirements for impact and fire accident performance of waste packages can be broken down into a number of elements:

- the challenges to the waste package; the representative drop heights for the three categories of impact accidents, and the severity (i.e. temperature and duration) for the fire accident;
- the response of the waste package - loss of waste package integrity and release of activity and/or the ability of the waste package to be handled after the accident and/or the suitability of the waste package for onward waste management;
- the consequences of waste package response - on- and/or off-site dose consequences of activity release and/or the recovery of damaged waste packages.

Three type of mechanical challenge have been considered in Section 7.7.2:

- minor impacts - which result in no significant damage to the waste package such that it can continue to be managed without rework;
significant impacts - which result in damage to the waste package, that may require rework, but which result in no loss of containment and release of contents;

DBA impacts - which result in the breach of the waste package containment and release of contents.

Currently, bounding impact drop heights of 5m and 10m respectively have been adopted for the latter two challenges, as explained in Section 7.7.2. To allow criteria for all three challenges to be fully defined for these type of impact, the following areas need to be addressed:

- identification of the types of mechanical challenge to waste packages (work has already started to address this issue [12]);
- determination of the response of waste packages to such challenges to identify the boundary between 'significant' and DBA impacts and define representative drop heights for the two accident types;
- determination of waste package activity release fractions\(^{23}\) following DBA impact accidents;
- determination of the exposure routes of released activity and the resulting dose to both on- and off-site persons.

The robust nature of HLW/SF waste packages and their contents would be expected to result in relatively small release fractions for even the most severe challenges. However, the high specific activity of HLW/SF could mean that activity associated with relatively small releases of material could be significant.

Modelling is required to demonstrate the continued integrity of waste packages following a 'significant' impact and to ascertain the nature and consequences of any breach of the waste package containment following a DBA impact accident and the magnitude of activity release from the package contents.

The approach to determining the on- and off-site dose consequences of any release from waste packages following impact and/or fire accidents will be similar to that adopted for the PGRC as, once released from a waste package, activity would behave in a similar manner whatever the source.

The approach to evaluating the potential for fire accidents and their consequences will be similar to that for impact accidents. In the case of fire the operational safety assessment identifies the nature of fires that could occur including the presence and quantities of fuels and other combustible materials, to allow the severity (i.e. maximum temperature and duration) of the DBA fire accident to be defined. It is currently assumed \([12]\) that the 1000°C/1 hour fire deemed representative for fires in an ILW repository would be equally representative for the Reference HLW/SF Concept. Confirmation of this fire severity and modelling of the response of each of waste package type to such a fire is therefore required.

\(^{23}\) Defined as the quantity of activity as a fraction of the total waste package inventory.
9.12 Other Mechanical Challenges to Waste Packages

It is known that waste packages will experience mechanical challenges other than those resulting from impacts. These included external loading of the waste package resulting from hydrostatic effects, including glacial ice, and from tectonic changes etc resulting in rock movement. The ability of waste packages to resist such loadings would be part of a complete WPS.

9.13 Criticality Safety

As noted in Section 7.9, significant variation in the fissile material inventory of waste packages is likely and this may have an impact on the assurance of the benign nature of waste packages from the criticality viewpoint. The initial work to consider the operational and post closure criticality safety of waste packages [13] has shown that waste packages containing 'typical' fissile material inventories will be acceptable. However the variation in the quantity and nature (i.e. U-235 enrichment and Pu isotopic mixture) is likely to be large in AGR and PWR fuel (it is assumed that HLW canisters contain too little fissile material to constitute a criticality safety concern under any circumstance) and the work done to date will not apply to the full range of fuel anticipated.

Modelling work is therefore required to define generic fissile material limits for AGR and PWR waste packages.
10 SUMMARY

This document identifies and defines the standards and specifications for waste packages containing HLW and SF that would be subject to the long-term waste management process as defined by the Reference HLW/SF Concept. Criteria are identified and defined for the complete waste package together with those with particular relevance to the waste canister and the acceptable contents of the waste packages. As such, this document can be used as the basis for the assessment of specific packaging proposals, for the categories of waste considered, against the requirements of the Reference HLW/SF Concept.

The document also identifies the areas where more information and research is required to allow the standards and specifications to become more defined and to develop in parallel with the Reference HLW/SF Concept.
APPENDIX A  Identification of WPS Criteria

Nirex has previously carried out an initial consideration of the WAC that would be relevant to the long-term management of HLW and SF [4]. Although not based on a specific long-term management concept, this work identified a total of 14 waste package and wasteform criteria for packages containing HLW or SF:

<table>
<thead>
<tr>
<th>Radionuclide Inventory</th>
<th>Thermal Effects</th>
<th>Criticality Safety</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiation Effects</td>
<td>Mechanical Properties</td>
<td>Combustibility</td>
</tr>
<tr>
<td>Gas Generation</td>
<td>Free Liquids</td>
<td>Explosive Materials</td>
</tr>
<tr>
<td>Compressed Gases</td>
<td>Toxic/Corrosive Materials</td>
<td>Chemical Durability</td>
</tr>
<tr>
<td>Physical Properties</td>
<td>Identification</td>
<td></td>
</tr>
</tbody>
</table>

In the development of the GWPS for ILW [2] waste package criteria were identified following an extensive review of the PRGC and its supporting documentation, of the relevant national and international legislation, regulations and guidance and a consideration of the highly diverse physical and chemical nature of the UK’s ILW. This process resulted in the identification of 18 waste package and wasteform criteria:

<table>
<thead>
<tr>
<th>Activity Content</th>
<th>Surface Dose Rate</th>
<th>Heat Output</th>
</tr>
</thead>
<tbody>
<tr>
<td>Surface Contamination</td>
<td>Dimensions</td>
<td>Lifting Feature</td>
</tr>
<tr>
<td>Gross Mass</td>
<td>Gas Generation</td>
<td>Venting</td>
</tr>
<tr>
<td>Integrity</td>
<td>Criticality Safety</td>
<td>Wastefom Properties</td>
</tr>
<tr>
<td>Impact Performance</td>
<td>Fire Performance</td>
<td>Stackability</td>
</tr>
<tr>
<td>Identification</td>
<td>Physical Protection</td>
<td>Safeguards</td>
</tr>
</tbody>
</table>
These criteria were deemed to encompass all the requirements for waste packages containing ILW when compared with the requirements of the PGRC and were used in combination with the criteria identified for HLW and SF listed above to derive a list of criteria for waste packages containing HLW and SF against the requirements of the Reference HLW/SF Concept.

These criteria were then considered to determine whether they were predominantly requirements of the waste package as a whole or requirements of either of the waste package inputs (i.e. waste container or contents). Table A1 lists the outcome of this process together with references to the relevant sections in this document.
Table A1  Waste Package Criteria

<table>
<thead>
<tr>
<th>Criterion</th>
<th>Component</th>
<th>Section(s)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Activity Content</td>
<td>Waste package + Contents</td>
<td>7.1</td>
</tr>
<tr>
<td>Surface Dose Rate</td>
<td>Waste package + Contents</td>
<td>7.3</td>
</tr>
<tr>
<td>Heat Output</td>
<td>Waste package + Contents</td>
<td>7.4</td>
</tr>
<tr>
<td>Surface Contamination</td>
<td>Waste package</td>
<td>7.5</td>
</tr>
<tr>
<td>Dimensions</td>
<td>Waste container</td>
<td>5.1.1</td>
</tr>
<tr>
<td>Lifting Feature</td>
<td>Waste container</td>
<td>5.1.2</td>
</tr>
<tr>
<td>Gross Mass</td>
<td>Waste package</td>
<td>7.2</td>
</tr>
<tr>
<td>Leachability</td>
<td>Contents</td>
<td>6.2.4</td>
</tr>
<tr>
<td>Gas Generation</td>
<td>Waste package + Contents</td>
<td>6.2.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.6</td>
</tr>
<tr>
<td>Integrity</td>
<td>Waste container</td>
<td>5.4</td>
</tr>
<tr>
<td>Criticality Safety</td>
<td>Waste package + Contents</td>
<td>6.2.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.9</td>
</tr>
<tr>
<td>Wasteform Properties</td>
<td>Contents</td>
<td>6</td>
</tr>
<tr>
<td>Impact Performance</td>
<td>Waste package + Waste container</td>
<td>5.5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.7</td>
</tr>
<tr>
<td>Fire Performance</td>
<td>Waste package + Waste container</td>
<td>5.6</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.8</td>
</tr>
<tr>
<td>Retrievability</td>
<td>Waste package + Waste container</td>
<td>5.4</td>
</tr>
<tr>
<td>Identification</td>
<td>Waste container</td>
<td>5.3</td>
</tr>
<tr>
<td>Physical Protection</td>
<td>Waste package + Contents</td>
<td>7.10</td>
</tr>
<tr>
<td>Safeguards</td>
<td>Waste package + Contents</td>
<td>7.11</td>
</tr>
</tbody>
</table>
APPENDIX B  Waste Package Identifier System and Format

The identifier shall consist of ten alpha-numeric characters arranged in a horizontal sequence from left to right with no intermediate spaces or other markings as shown in Figure B1.

Figure B1  Format of Waste Package Identifier

The alphabetic characters shall be restricted to those required in the hexadecimal number system, namely A, B, C, D, E, F.

The characters shall be of the OCR-A form, as specified by BS5464: Part 1, [38].

The complete system of identifiers shall be defined in terms of three data fields contained within the ten alpha-numeric character sequence. The three data fields shall be numbered 1, 2, 3 from left to right and shall have the following meaning and format:

Data Field 1: 2 hexadecimal characters - HH
Data Field 2: 6 decimal characters - DDDDDD
Data Field 3: 2 decimal check characters - CC

Arranged as shown in Figure B1.

Data Field 1

Two sequential hexadecimal characters with no intermediate space. Each character to be one of the following:

0 1 2 3 4 5 6 7 8 9 A B C D E F

This data field identifies a waste producing organisation or waste packaging site.
Nirex Report N/124

Data Field 2
A sequence of six decimal digits with no intermediate spaces. Each character to be one of the following:
0 1 2 3 4 5 6 7 8 9
This data field identifies a package number from 000001 to 999999.

Data Field 3
A sequence of two decimal digits with no intermediate space. Each character to be one of the following:
0 1 2 3 4 5 6 7 8 9
This data field is a check number that is derived mathematically from Data Fields 1 and 2. The check numbers shall be derived by the following algorithm.

\[ CC = 97 - R \text{ where } R = \left\{ \left( HHDDDDDD \right) \times 100 \right\} \mod 97 \]
where HHDDDDDD is a real number.

In each case H is to be converted to its decimal equivalent and is to consist of 2 digits, using zero to ‘pad out’ where necessary.

When a computer program is used to generate the check digits, it shall be independently checked, by the use of validated software, to ensure that these digits are correct.
APPENDIX C  Regulatory and other Restrictions on the Transport of HLW and SF

As discussed in Section 3 it is assumed that all waste packages will be manufactured in a facility that will be an integral part of the HLW/SF repository. Accordingly the Reference HLW/SF Concept does not anticipate that complete waste packages will be transported through the public domain at any stage following manufacture. However, vitrified HLW and SF assemblies will require to be transported to the repository packaging plant and it is anticipated that shielded transport containers meeting certain specified transport requirements will be utilised. Whilst not part of the specification for the completed waste package, as defined in this document, it is nevertheless relevant to summarise the criteria that will apply to this transport operation.

C1  Regulatory Restrictions

It is assumed that the material will be transported through the public domain from the site of arising or storage to the repository. The material will be carried in shielded transport containers and the resulting transport packages will be subject to the IAEA Transport Regulations [27]. These may place limits on the nature and quantities of material that can be transported in a single transport package and on the required performance of the transport container.

The ensuing section deals with the limits imposed on the contents of transport packages by the IAEA Transport Regulations, assuming that packages will be transported as Type B transport packages and under the conditions of ‘exclusive use’ as defined by the Regulations24. Direct references to the IAEA Transport Regulations are made by way of reference to the paragraph number.

C1.1  Activity Content

A limit of $10^{5}A_{2}$25 on the contents of a Type B transport package which has not been designed to withstand an ‘enhanced water immersion test’ (Paragraph 730). Current Nirex ILW transport container designs, the Reusable Shielded Transport Container (RSTC) and the Standard Waste Transport Container (SWTC), have not been qualified to this standard although both designs are believed to be sufficiently robust if a need for such qualification was required. Other nuclear operators within the UK have transported spent fuel with activity levels well in excess of $10^{5}A_{2}$ for many years. There is therefore unlikely to be any explicit limit on the activity of HLW/SF over and above those placed implicitly on the total contents of a transport container by virtue of limits on external dose rate, heat output, criticality safety and fire and impact performance requirements.

24 ‘Exclusive use’ is defined by the IAEA Transport Regulations (Paragraph 221) as meaning ‘the sole use, by a single consignor, of a conveyance or large freight container, in respect of which all initial, intermediate and final loading and unloading is carried out in accordance with the consignor or consignee’.

25 $A_{2}$ is a measure of activity linked to possible exposure pathways and defined in the IAEA Transport Regulations.
C1.2 Dose Rate

The dose rate at 2m from the surface of a transport package shall not exceed 0.1mSv h$^{-1}$ and the dose rate on its external surface shall not exceed 10mSv h$^{-1}$ (Paragraph 572).

The dose rate from the transport package contents will be such that these external limits are not exceeded when the contents are shielded by the walls of the transport container.

C1.3 Heat Output

The maximum surface temperature on any ‘readily accessible’ external surface of the transport shall not exceed 85°C (Paragraph 662). This will place limits of the heat generation of the contents of a transport container.

Transport packages with surface heat flux greater than 15Wm$^{-2}$ will require specified stowage provisions during transport and storage in transit (Paragraph 565).

C1.4 Gas Generation

The total gas generated by the waste package should not exceed a value that would over-pressurise the transport container (i.e. not exceed the maximum normal operating pressure (MNOP) - Paragraph 661).

Pressurisation of the transport container cavity could compromise its ability to restrict the loss of radioactive contents (which would include the release of radioactive gases) from the transport container, during normal conditions of transport to $10^{-6}$ A$_2$ per hour (Paragraph 656 (a)).

Gas generation by HLW and SF will be small and both are contained within nominally sealed containers so the release of gas into the transport container cavity will be correspondingly small.

C1.5 Criticality Safety

Packages containing relatively small quantities of fissile material$^{26}$ can be excepted from the requirements for packages containing fissile material (Paragraphs 671 to 682). Individual HLW canisters would be excepted from if they contain less than 15g of fissile material (Para 672 (a) (i)) as would groups of such canisters in a transport container provided that they did not together exceed the consignment mass limit (Para 672 (a)) and could be shown to contain less than 5g of fissile material in any 10 litre volume (Para 672 (a) (iii)).

Both types of SF will contain too much fissile material to be excepted and would require modelling to show compliance with Paragraphs 673-682, proving sub-criticality for individual transport packages and arrays of transport packages under normal and accident conditions of transport.

---

$^{26}$ Defined as U-233, U-235, Pu-239 and Pu-241.
C1.5 Impact Performance

A ‘Mechanical Test’ is defined for Type B transport containers (Paragraph 727). This comprises:

- a free drop from 9m on to a flat unyielding surface; and
- a drop from 1m on to a specified ‘punch’; and
- a crush test involving a 500kg mass being dropped from 9m on to the transport package.

Following such a test, and a ‘Thermal Test’ as defined below, the design of the transport package should be such that it will (Paragraph 656):

‘….restrict the accumulated loss of radioactive contents in a period of one week to........not more than A$_2$…’

For unshielded ILW packages the Mechanical Test is used as the basis for defining a ‘transport impact accident’ for waste packages inside a transport container as a 10m drop on to a flat unyielding surface.

The allowable releases from the ‘bare’ contents of a transport container under accident conditions depends on:

- the design of the transport container containment system and particularly the leakage rate through it following an accident;
- the internal free volume of the transport container, and;
- the internal pressure in the week following the accident.

If, as expected, the contents produce little or no gas during transport, the internal pressure will be very low and the allowable releases from the contents into the transport container cavity could be many hundreds of A$_2$.

C1.7 Fire Performance

The ‘Thermal Test’ for Type B transport containers (Paragraph 728) is defined as being the equivalent of being exposed to a fully engulfing hydrocarbon fuel/air fire with an average temperature of 800°C for a period of 30 minutes, followed by a period of unforced cooling. As explained above, the thermal test is required to follow the mechanical test and the activity release limit of A$_2$ applied to the week following both tests.

The transport container is expected to provide significant protection to its contents from the effects of a fire and will continue to provide significant containment during and following the fire.

C2 Other Restrictions Imposed by Transport

A number of restrictions will be placed on the transport of HLW and SF through the public domain as a result of the limitations of the UK transport infrastructure.
Nirex Report N/124

C2.1 Dimensions

Materials to be transported must be capable of fitting inside the transport container whilst making efficient use of the internal volume and without leaving excessive ‘play’ that would allow movement of the contents during transport. The external dimensions of transport containers will be limited by the mode of transport selected and the internal dimensions will be similarly limited with the additional constraint set by the need for shielding.

C2.2 Mass

In the case of items with high external dose-rates such as HLW/SF a large proportion of the total mass of the transport package is that of the transport container as a result of the size of the contents and its shielding requirements (in the case of unshielded ILW packages the mass of the waste packages is less than 20% of the total mass of the transport package).

Gross mass limits are placed on transport packages by the mode of transport selected. A limit of 65t is placed on packages transported by rail on a four axle wagon. Greater masses can be transported on wagons with larger numbers of axles (e.g. 6 or 8) although such wagons may not be able to be used on some tracks due to cornering restrictions etc.

Greater masses can, in principle, be transported by road although particularly restrictive limits are placed on loads deemed as ‘divisible’.

C2.3 Nuclear Security Physical Protection

The Nuclear Industries Security Regulations (NISR) 2003 lays down the approvals required for the physical protection of nuclear materials in transit between licensed sites, against the risk of theft or sabotage. They are administered and enforced by the OCNS acting on behalf of the Secretary of State for Trade and Industry.

Material is categorised on the basis of nuclear material content and physical form. HLW and SF are both defined by the Regulations as ‘irradiated materials’ and provided the dose rate at 1 metre exceeds 1Gy/hr they require protection to Category III standards. Lower dose rates (i.e. for lightly irradiated fuel) may result in material requiring higher levels of protection depending on the quantity of fissile material and its physical form.

C2.4 Safeguards

Packaged wastes that contain isotopes of uranium, plutonium or thorium derived from the UK civil nuclear programme may be subject to national and international controls known as Nuclear Materials Safeguards. In principle, where these materials are subject to Safeguards, it is likely that they will be subject to those controls during all stages of the long-term waste management including transport.

Safeguards are assumed to be relevant for SF but not HLW due to small quantities of fissile material.
11 GLOSSARY OF TERMS

Also see the list of Abbreviations.

Cross-references to other definitions in this Glossary are in bold.

Conditions for Acceptance

Those criteria relevant to the acceptance of waste packages for safe handling and potential disposal at a specific facility (also known as Waste Acceptance Criteria)

Contents

In the context of this document the contents of a waste package comprise vitrified HLW in WVP canisters, PWR fuel or consolidated AGR fuel.

Criticality

Criticality is a state in a radioactive substance in which a self-sustaining neutron chain reaction occurs. Criticality requires that a sufficiently large quantity of fissile material be assembled into a configuration that can sustain a chain reaction; unless both of these requirements are met, no chain reaction can take place – the system is said to be sub-critical.

Dose, Dose rate

In the context of this document ‘dose’ can be taken to mean effective dose equivalent unless stated otherwise. Similarly, ‘Dose rate’ would mean the effective dose equivalent per unit time. The SI unit of effective dose equivalent is the sievert (Sv), and typical units of dose rate are sievert/hour (Svh\(^{-1}\)) and sievert/year (Svy\(^{-1}\)).

Dose equivalent

Dose equivalent takes into account not only the energy deposited in body tissue by radioactivity (either external or internal) but also the different biological effectiveness of the various forms of radiation in causing harm to body tissues. The SI unit of dose equivalent is the sievert (Sv).

Effective dose equivalent

In addition to dose equivalent taking into account the biological effectiveness of various forms of radiation, effective dose equivalent takes into account the differing sensitivities of various body tissues. Effective dose equivalent thus aims to reflect the risk to health for the irradiated person, regardless of the widely different dose equivalents that might be received by the various organs. This is a useful concept for comparisons between the risks from various radiation exposure pathways.

Fissile material

Fissile material is a material that undergoes fission under neutron irradiation. For regulatory purposes material containing any of the following nuclides is considered to be ‘fissile’: uranium-233, uranium-235, plutonium-239, plutonium-241.
Nirex Report N/124

Hazardous materials
Materials that can endanger human health if improperly handled.

High Level Waste (HLW)
Waste in which the temperature may rise significantly as a result of their radioactivity, so that this factor has to be taken into account in designing storage or disposal facilities. HLW is primarily produced from the reprocessing of irradiated nuclear fuel.

Immobilisation
The process by which the radioactivity present in waste is conditioned into a form that confers passive safety. This reduces the potential for migration or dispersion of the radioactivity by natural processes during handling and potential disposal.

Intermediate Level Waste (ILW)
Waste with a radioactivity content that exceeds either of the upper limits for Low Level Waste, but has lower radioactivity and heat output than High Level Waste.

Letter of Compliance\(^{27}\) (LoC) process
The process by which Nirex assesses proposals for waste packaging for compatibility with the relevant waste management concept and compliance with the relevant Nirex standards and specifications.

Low Level Waste (LLW)
Waste with a radioactivity content that is greater than acceptable for disposal with household refuse (Very Low Level Waste) but does not exceed 4 GBq/t of alpha radioactivity or 12 GBq/t of beta/gamma radioactivity.

Passive safety
Passive safety is a state in which radioactive waste is chemically and physically stable, and is stored in a manner that minimises the need for safety mechanisms, maintenance, monitoring and human intervention.

Quality Management System
A Quality Management System is the overall system by which an organisation determines, implements and ensures quality.

Retrieval, Retrievability
Facilities provided by the engineered features of the waste packages and repository system, for future generations to retrieve emplaced waste packages.

\(^{27}\) Formerly known as a Letter of Comfort
Shielding

Shielding is the protective use of materials to reduce the dose rate outside of the shielding material. The amount of shielding required to ensure that the dose rate is ALARP will therefore depend on the type of radiation, the activity of the source, and on the dose rate that is acceptable outside the shielding material.

Toxic substance

A substance that, if it is inhaled or ingested or if it penetrates the skin, may involve serious acute or chronic health risks.

Transport Regulations

The IAEA Regulations for the Safe Transport of Radioactive Material [27]; and/or those regulations as transposed into an EU Directive, and in turn into regulations that apply within the UK.

Verification

A review of work activities by a competent individual who is outside the direct line management of a facility. In the context of this document, Independent Verification is used to confirm that activities that may affect product quality have been properly carried out and recorded.

Waste Acceptance Criteria

Those criteria relevant to the acceptance of waste packages for safe, transport handling and potential disposal at a specific facility (also known as Conditions for Acceptance).

Waste container

The outer containment of the waste package (i.e. the copper canister) together with any internal furniture (i.e. the cast iron insert).

Wasteform

The component of the waste package contents that derives from the original waste:

- the vitrified HLW but not the canister in which it is contained;
- all original components of SF, but not any materials used in subsequent treatment (i.e. the baskets used to contain consolidated fuel).

Waste package

The waste container and its contents, as manufactured and prepared for all future stages of long-term waste management.

Waste Product Specification

A document prepared by the waste packager which describes the quality (properties and composition) and performance characteristics of each distinct type of waste package produced in a waste packaging plant.
12 ABBREVIATIONS

AGR  Advanced Gas-cooled Reactor
ALARP  As Low As Reasonably Practicable
BWR  Boiling Water Reactor
CCAD  Criticality Compliance Assurance Documentation
CEC  Commission of the European Communities
CfA  Conditions for Acceptance
DBA  Design Basis Accident
EA  Environment Agency
EC  European Community
EU  European Union
GOSA  Generic Operational Safety Assessment
GWPS  Generic Waste Package Specification
HEPA  High Efficiency Particulate in Air (filter)
HEU  Highly Enriched Uranium
HLW  High Level Waste
HSE  Health and Safety Executive
HWR  Hazardous Waste Regulations
IAEA  International Atomic Energy Agency
ILW  Intermediate Level Waste
ISO  International Organisation for Standardisation
KBS  Kärnbränslesäkerhet - Nuclear Fuel Safety
LLW  Low Level Waste
LoC  Letter of Compliance
NDSP  Nuclear Design Safety Principles
NII  Nuclear Installations Inspectorate
**Nirex Report N/124**

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>NISR</td>
<td>Nuclear Industries Security Regulations</td>
</tr>
<tr>
<td>OCNS</td>
<td>Office for Civil Nuclear Security</td>
</tr>
<tr>
<td>PGRC</td>
<td>Phased Geological Repository Concept</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurised Water Reactor</td>
</tr>
<tr>
<td>QMS</td>
<td>Quality Management Systems</td>
</tr>
<tr>
<td>RPPM</td>
<td>Radiological Protection Policy Manual</td>
</tr>
<tr>
<td>SAPs</td>
<td>NII Safety Assessment Principles</td>
</tr>
<tr>
<td>SEPA</td>
<td>Scottish Environment Protection Agency</td>
</tr>
<tr>
<td>SF</td>
<td>Spent (Nuclear) Fuel</td>
</tr>
<tr>
<td>SFM</td>
<td>Safe Fissile Mass</td>
</tr>
<tr>
<td>SKB</td>
<td>Svensk Kärnbränsleförsörjning - Swedish Nuclear Fuel and Waste Management Company</td>
</tr>
<tr>
<td>WAC</td>
<td>Waste Acceptance Criteria</td>
</tr>
<tr>
<td>WPS</td>
<td>Waste Package Specification</td>
</tr>
<tr>
<td>WPrS</td>
<td>Waste Product Specification</td>
</tr>
<tr>
<td>WVP</td>
<td>Waste Vitrification Plant</td>
</tr>
</tbody>
</table>
13 REFERENCES

20. Nuclear Installations Act 1965 (as amended) (c.57).
<table>
<thead>
<tr>
<th>No.</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>25</td>
<td>Health and Safety at Work Act, 1974 (c. 37).</td>
</tr>
</tbody>
</table>

43 Correspondence Pettersson(SKB)/King (Nirex), 7th October 2005 (Available on request).


