Nuclear Decommissioning Authority

WASTE PACKAGE SPECIFICATION AND GUIDANCE DOCUMENTATION

WPS/715: 3 cubic metre Box Waste Package Specification: Explanatory Material and Design Guidelines for corner-lifting variant

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WASTE PACKAGE SPECIFICATION AND GUIDANCE DOCUMENTATION

3 CUBIC METRE BOX WASTE PACKAGE SPECIFICATION: EXPLANATORY MATERIAL AND DESIGN GUIDELINES FOR CORNER-LIFTING VARIANT

This document forms part of a suite of documents prepared and issued by the Radioactive Waste Management Directorate (RWMD) of the Nuclear Decommissioning Authority (NDA).

The Waste Package Specification and Guidance Documentation (WPSGD) provide specifications and guidance for waste packages, containing Intermediate Level Waste and certain Low Level Wastes, which meet the transport and disposability requirements of geological disposal in the UK. They are based on, and are compatible with, the Generic Waste Package Specification (GWPS).

The WPSGD are intended to provide a 'user-level' interpretation of the GWPS to assist Site License Companies (SLCs) in the early development of plans and strategies for the management of radioactive wastes. To aid in the interpretation of the criteria defined by the WPSGD, and in their application to proposals for the packaging of wastes, SLCs are advised to contact RWMD at an early stage.

The WPSGD will be subject to periodic enhancement and revision. SLCs are therefore advised to contact RWMD to confirm that they are in possession of the latest version of any documentation used.

WPSGD DOCUMENT NUMBER WPS/715 - VERSION HISTORY		
VERSION	**DATE**	**COMMENTS**
WPS/715/01	March 2007	Aligns with Issue 2 of GWPS (Nirex Report N/104), as published March 2007
WPS/715/02	October 2007	Updated to acknowledge NDA assumption of Nirex responsibilities
WPS/715/03	March 2008	Changes to NII SAPs and modelling of DBAs

This document has been compiled on the basis of information obtained by Nirex and latterly by the NDA. The document was verified in accordance with arrangements established by the NDA that meet the requirements of ISO 9001. The document has been fully verified and approved for publication by the NDA.

CONTENTS

1 INTRODUCTION

The Radioactive Waste Management Directorate (RWMD) of the Nuclear Decommissioning Authority (NDA) has been established with the remit to implement the geological disposal option for the UK's higher activity radioactive wastes. The NDA is currently working with Government and stakeholders through the *Managing Radioactive Waste Safely* (MRWS) consultation process to plan the development of a Geological Disposal Facility (GDF).

As the ultimate receiver of wastes, RWMD, acting as GDF implementer and future operator, has established waste packaging standards and defined package specifications to enable the industry to condition radioactive wastes in a form that will be compatible with future transport and disposal. In this respect RWMD is taking forward waste packaging standards and specifications which were originally developed by United Kingdom Nirex Ltd, which ceased trading on 1st April 2007 and whose work has been integrated into the NDA.

The primary document which defines the packaging standards and specifications for Intermediate Level Waste (ILW), and certain Low Level Wastes (LLW) not suitable for disposal in other LLW facilities is the Generic Waste Package Specification (GWPS) [1]. The GWPS is supported by the Waste Package Specification and Guidance Documentation (WPSGD) which comprises a suite of documentation primarily aimed at waste packagers, its intention being to present the generic packaging standards and specifications at the user level. The WPSGD also includes explanatory material and guidance that users will find helpful when it comes to application of the specification to practical packaging projects. For further information on the extent and the role of the WPSGD, reference should be made to the *Introduction to the Waste Package Specification and Guidance Documentation, WPS/100*[1].

RWMD has defined a range of standard containers that can be used to produce waste packages, and has issued a specification for each waste package. To assist waste packagers in applying these specifications to their waste packaging proposals, each waste package specification is accompanied by a document containing explanatory material and design guidelines. This document contains the explanatory material and design guidelines that accompany *Specification for corner-lifting variant of the 3 cubic metre Box Waste Package, WPS/315*.

2 BACKGROUND

2.1 The Concept of Geological Disposal

A key aspect in the production of standards and specifications for packaged waste is the definition of a disposal system which encompasses all stages of the long-term management of waste from retrieval through to final disposal.

In line with the MRWS consultation process, RWMD are continuing to develop concepts for the geological disposal for higher activity wastes which include ILW, and certain LLW

[1] Specific references to individual documents within the WPSGD are made in this document in *italic script,* followed by the relevant WPS number.

not suitable for disposal in other LLW facilities[2]. It is envisaged that the geological disposal of such wastes would comprise a number of distinct stages including:

- the retrieval and conditioning of the waste to create disposable waste packages, usually at the site of waste arising;
- a period of interim surface storage, also at the site of arising;
- transport of the waste packages to a GDF;
- transfer of waste packages underground and emplacement in disposal vaults;
- a period of monitored storage underground, during which retrieval by relatively simple means would be feasible;
- back-filling of the disposal vaults, followed by eventual sealing and closure.

The timing and duration of each stage would depend on a number of criteria, including the geographical location and host geology of a GDF as well as the disposal concept selected for implementation.

The Phased Geological Repository Concept (PGRC) [2], has been developed as one manifestation of geological disposal and has been adopted as the reference concept for the purposes of establishing packaging standards. The PGRC is supported by a suite of safety, security and environmental assessments intended to demonstrate that this concept will provide safety to workers and the public and provide the necessary level of environmental protection.

The safety philosophy adopted in the PGRC is one of containment of radionuclides by multiple barriers, of which that provided by the waste package is a key component. Included in these barriers are those provided by the waste package, which itself can be considered as two independent but complimentary barriers, the waste container and the wasteform, each of which plays an important role in the containment of radionuclides.

As the MRWS consultation process continues it is anticipated that the siting process, based on expressions of interest from volunteer communities, may lead to the identification of sites for investigation as to suitability to host a GDF. The disposal concept design and safety case will be developed to suit the specific characteristics of the site and packaging standards will be updated to reflect the new circumstances as appropriate.

2.2 The Generic Waste Package Specification

A major area of the RWMD's work is the provision of advice to the packagers of radioactive waste in the UK, by way of the definition of packaging standards and the assessment of individual waste packaging proposals against those standards.

The primary document that defines packaging standards for ILW is the GWPS. Derived from the PGRC and its associated generic documentation, which comprise the system specifications and safety assessments that define the PGRC, the GWPS provides the basis for assessing the suitability of waste packages containing ILW for disposal in a GDF.

The packaging standards defined by the GWPS are generic in two respects in that they are:

- derived from a full consideration of all future stage of long-term waste management; and

[2] The generic description 'ILW' is used in the remainder of this document to describe both these categories of waste.

- independent of the location of the site of a GDF, which could be implemented at a range of different sites within the UK, representing a range of geological environments.

The format of the GWPS is to define:

- general requirements that are applicable to all waste packages;
- a range of standard waste containers;
- specific requirements for the standard waste package design that are created using the standard waste containers;
- requirements for the conditioned wasteforms that are placed into containers;
- requirements for quality management and for the creation and maintenance of records about each individual waste package.

The GWPS therefore defines the performance requirements for the two barriers to the release of radionuclides provided by the waste package, the waste container and the wasteform, against which the overall performance of waste packages can be assessed.

2.3 The Assessment of Packaging Proposals

Since the mid-1980s, waste producers in the UK have made significant investment in waste retrieval and packaging plant as a means of ensuring that such wastes are rendered passively safe and suitable for disposal. Historically Nirex was responsible for the assessment and endorsement of the suitability of packaging processes for this latter need, originally by way of the ‘Letter of Comfort’ assessment process. Over the ensuing two decades the Letter of Comfort process has developed and matured to a point that the assessments undertaken were established on a more structured footing with detailed advice being issued to waste producers highlighting further information needs, or need for further development and/or research before a Letter of Comfort could be issued. The assessment process was also modified to integrate better with the implementation of packaging plant projects, with staged interactions occurring at a number of stages before active operation of a packaging plant commenced. The status of the assessment process was strengthened in January 2004, when support was provided by UK nuclear regulators, and it was recognised within improved regulatory arrangements for nuclear licensed sites [3]. This was accompanied by significant changes to the assessment process which was renamed the ‘Letter of Compliance’ assessment process, a full description of which can be found in *Guide to the Letter of Compliance Assessment Process, WPS/650*.

In April 2007 Nirex was dissolved and its responsibilities assumed by RWMD. This included the role of assessing and endorsing nuclear site operators’ waste packaging proposals through the LoC assessment process.

In undertaking LoC assessments RWMD determines whether wastes, when packaged, will have characteristics compliant with plans for transport to, and operations at a GDF, and ultimately whether the wastes could be accommodated within a GDF long-term post-closure safety case. The main output of a LoC assessment is an Assessment Report which may be accompanied by the issue of a LoC endorsing the packaging proposal. In line with the recently updated regulatory guidance [4] such endorsement is now seen by the regulators as an important component of the operator’s Radioactive Waste Management Case.

Specification for the corner-lifting variant of the 3 cubic metre Box Waste Package, WPS/315 is intended to provide waste packagers with a reference point against which waste packaging proposals can be progressed to the point at which a submission for assessment by way of the LoC process can be made. Waste packagers will find

Guidance on the Preparation of Letter of Compliance Submissions, WPS/908, of assistance in this matter.

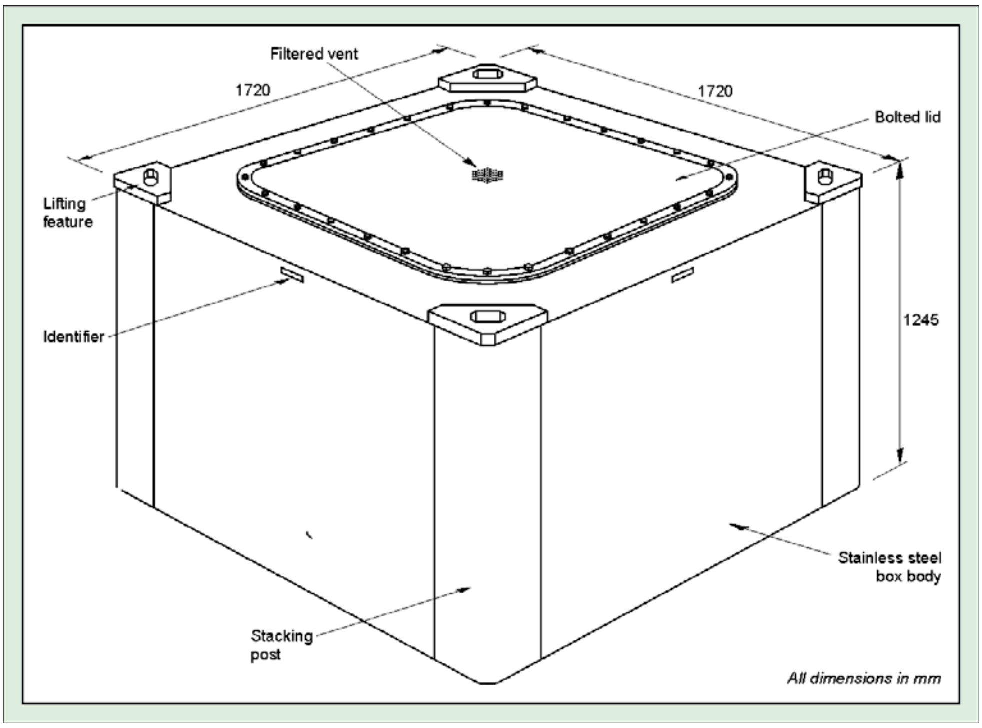
3 THE 3 CUBIC METRE BOX WASTE PACKAGE

The 3 cubic metre Box waste package is one of a limited range of standard waste packages defined by the GWPS. It is used for the conditioning of ILW which includes larger items, typically legacy wastes stored at various nuclear facilities throughout the UK.

Within the standard dimensional envelope of the 3 cubic metre Box waste package, two distinct variants have been developed to accommodate the process requirements for treating particular waste types. The two variants, which are mainly distinguished by their lifting features which are located at the mid points of the sides of the waste package (the ‘side-lifting’ variant - see *Specification for the side-lifting variant of the 3 cubic metre Box Waste Package, WPS/310*) or at the corners of the waste package (the ‘corner-lifting’ variant - Figure 1). Fuller descriptions of both variants can be found in the GWPS [1].

The 3 cubic metre Box waste package is an ‘unshielded waste package’ in that it is typically manufactured from relatively thin stainless steel and, because of either external radiation levels or requirements for the containment of radionuclides, remote handling is usually required. For similar reasons, 3 cubic metre Box waste packages are usually transported in a reusable shielded transport containers, the combination of waste packages and transport container being classed as a Type B transport package under the IAEA Transport Regulations [5].

Figure 1 The corner-lifting variant of the 3 cubic metre Box waste package



4 PACKAGING CRITERIA

Section 3 of *Specification for the side-lifting variant of the 3 cubic metre Box Waste Package, WPS/310* identifies the standards and performance criteria that are required of 3 cubic metre Box waste packages. This section provides a summary of the justification (which can be found, in full in the GWPS [1]) for each of the criteria together with explanatory material and guidance as to their application. The numerical criteria given in the Specification tend to represent the most onerous values which have been determined using conservative assumptions. Relaxation of these values may be possible when the specific packaging proposals are assessed and this Section indicates the extent to which such relaxation may be taken.

The criteria values are those that would be applicable at the time of transport from the waste packager's site (unless specifically stated otherwise). Because of the lengths of time involved between waste package manufacture and the subsequent stages of waste management, demonstrations of future compliance will in many cases have to be based on reasoned argument, experimental studies and/or modelling studies etc.

Direct quotations from *Specification for the side-lifting variant of the 3 cubic metre Box Waste Package, WPS/310* are in ***bold blue italic type***. Note that the words **should** and **shall** have the following meanings:

- **Shall** denotes a mandatory criterion that is derived from consideration of a regulatory requirement, and/or forms the basis for package standardisation;
- **Should** denotes a criterion that may be considered as a target, and variations may be possible following discussion with the NDA.

The above convention is consistent with the recommendations of BS 7373-1:2001 [6].

4.1 Activity Content

The waste package shall contain conditioned ILW or LLW and the activity content of the waste package shall be restricted to meet the other aspects of the specification (i.e. heat output, dose rate, criticality and normal operational and accident release criteria).

The activity content of the waste package shall not exceed 10^5 A_2.[3]

The 3 cubic metre Box waste package is primarily intended for use in the conditioning, transport and long-term management of ILW, although it is also suitable for use with LLW.

The transport of radioactive waste through the public domain places an absolute upper limit on the activity content of waste packages. It is envisaged that 3 cubic metre Box waste packages will be transported within a SWTC, the combination of which would be classified, by the IAEA Transport Regulations [5], as a Type B transport package. The maximum activity content of such a transport package is 10^5A_2[4] and this limit is applied directly to the 3 cubic metre Box waste package.

The limit on the total activity content of an individual waste package is generic with respect to the future GDF site. However, when a specific site is identified, the authorised

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[3] A_2 is a measure of activity defined in the IAEA Transport Regulations, that applies to the entire range of radionuclides and is linked to radiotoxicity and possible exposure pathways.

[4] This limit applies to transport packages that are not designed to satisfy the requirement of an 'enhanced water immersion test', a standard to which the SWTC is not currently qualified. Work is currently underway to so qualify the SWTC, when this is concluded it may be possible to remove the 10^5A_2 limit.

limits for certain radionuclides may be particularly sensitive to site-specific factors in the post-closure safety case. Work carried out to determine the specific impact of individual radionuclides on the geological disposal [7] has derived Guidance Quantities for each of the 112 'relevant radionuclides' that could, potentially, impact on the safe long-term management of packaged waste. Waste producers intending to package significant quantities of these radionuclides before site-specific Conditions for Acceptance become available are advised to maintain close contact with the NDA.

The quantity of activity that can be packaged in a 3 cubic metre Box may also be limited by other criteria, such as external dose rate (Section 4.2), heat output (Section 4.3), criticality safety (Section 4.11) and the requirements for the impact and fire performance of the waste package (Sections 4.13 and 4.14).

4.2 Dose Rate

The surface dose rate from the waste package shall not exceed a value commensurate with achieving 2mSvh^{-1} at the surface, and 0.1mSvh^{-1} at 1 metre from the surface, of a 280mm steel shield (density 7700kgm^{-3}) in direct contact with the waste package.

The limits on external dose rate from transport packages in the public domain are set by the IAEA Transport Regulations [5], the actual limits depend on the operational procedures applied during transport. In line with a conservative approach, RWMD has adopted the more stringent of the two transport regimes from the point of view of external dose rate, those pertaining to transport carried out under the conditions of 'non-exclusive use[5]'. For packages transported under these conditions:

- the dose rate at 1m from the surface of a transport package shall not exceed 0.1mSvh^{-1} and;
- the dose rate on its external surface shall not exceed 2mSvh^{-1}.

Waste packages resulting in transport packages with higher radiation levels may be permitted but this would be dependent on the approval certificate for the transport container, the operational procedures applied during transport and the operational safety case for a GDF. The ultimate upper limits for the dose rate from transport packages are those defined for 'exclusive use' and are:

- the dose rate at 2m from the surface of a transport package shall not exceed 0.1mSvh^{-1} and;
- the dose rate on its external surface shall not exceed 10mSvh^{-1}.

The regulatory limits apply to the transport package of a 3 cubic metre Box waste packages within a shielded transport container. The limit on the dose rate the waste package therefore depends on the shielding provided by the transport container, the maximum available shielding thickness being determined by the constraints on interior and exterior dimensions [8]. The interior constraint is produced by the dimensions of the waste package, plus appropriate clearances. For a package designed to use all available transport modes, the exterior constraint is derived from the cross-sectional limits of the UK rail system 'loading gauge', minus an allowance for a rail wagon cover and appropriate clearances. These dimensional calculations indicate a maximum possible shielding thickness of 285mm. To derive the package dose rate limit, a nominal shielding

[5] 'Exclusive use' is defined by the IAEA Transport Regulations 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'.* If all of these conditions cannot be met, transport is deemed to take place under 'non-exclusive use'.

thickness of 280mm of carbon steel[6] is assumed. The reduction by 5mm includes an allowance for manufacturing tolerances, and also reflects weight limitations and the need for thermal insulation panels on the transport container.

4.3 Heat Output

The total heat output from the waste package should not exceed 200 watts at the time of transport nor 150 watts at the time of vault backfilling.

Waste packages generate heat as a result of the radioactive decay of their contents (radiogenic heat), as well as from other sources such as biodegradation, cement hydration, corrosion and other chemical reactions. Typically, the radiogenic heat output of ILW is of the order of $1Wm^{-3}$ although variations of up to two orders of magnitude either side of this value are not unusual.

The radiogenic heat output from a waste package can be calculated from the radionuclide content and the effective radioactive decay energy per disintegration for each radionuclide (typically a few W/TBq).

Heat generation by non-radiogenic mechanisms can also be significant and could amount to an additional $3Wm^{-3}$ at times depending on the physical and chemical composition of the waste and conditioning materials. In extreme cases this additional heat could affect thermal performance, particularly following backfilling. Non-radiogenic sources of heat should therefore be included in heat calculations if they are likely to exceed $0.1Wm^{-3}$. The anticipated timescales of such additional heat generation may also need to be considered.

Heat generation by waste packages affects all stages of their long-term management and limits have to be set accordingly. During transport, assurance is required that internal heat generation will not alter the physical state of the package or its contents, or adversely affect the containment or shielding offered by the transport container. Additionally, the IAEA Transport Regulations [5] place limits on the external temperature and surface heat flux of transport packages. Thermal modelling of the SWTC and typical contents [9] has shown that waste package heat outputs of up to 200W would not result in temperature or heat flux limits being exceeded and this value is applied directly to the 3 cubic metre Box waste package.

Following emplacement in a GDF vault, heat generation by waste packages could result in elevated temperatures which could affect the long-term performance of a GDF. Chemical processes such as the corrosion of certain metals etc are known to vary significantly with temperature and the properties of the vault backfill (i.e. speciation, solubility and sorption) may also be deleteriously affected by excessive temperatures.

The Generic Disposal System Specification (GDSS) [10] defines the following temperature targets for the operational period of a GDF:

> *'A long-term temperature target of up to 50°C shall be applied to all waste packages.*
>
> *During the operational phase, short-term excursions above the 50°C target will be tolerable: e.g. experimental data are available to justify increases in waste package temperatures of up to 80°C as acceptable for a period of 5 years. For shorter durations, higher temperatures would probably be acceptable but experimental data are not currently available to demonstrate this.'*

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6 Although it is envisaged that other grades of steel (i.e. martenistic stainless steel) may be used for the manufacture of transport containers, the density and shielding properties of these materials will be similar to those of carbon steel.

Studies of the effects of heat generated by waste packages during the operational period of a GDF [11] have shown that an average heat output of $6Wm^{-3}$ (i.e. ~20W per 3 cubic metre Box) will not challenge these temperature targets. This work has also shown that the presence of limited numbers of waste packages with heat outputs at the maximum level specified for transport (i.e. 200W), will not result in excessive temperatures.

The same GDSS targets also apply after vault backfilling[7] when considerable quantities of heat will be generated by hydration of the cementitious backfill material. Studies have shown that a vault with a mean package heat output of $6Wm^{-3}$ will experience temperature in excess of the long-term target over a timescale of ~20 years [12] although the short term target would not be exceeded. Investigation of the effects of individual ‘hot’ packages [13] have shown that, without a defined heat sink, temperatures in excess of the maximum short-term target of 80°C could result from the presence of individual packages generating in excess of $50Wm^{-3}$. This value translates to a post-backfill heat output limit of 150W for the 3 cubic metre Box waste package.

The post-backfill heat limit of 150W placed on the 3 cubic metre Box waste package is the most bounding of all stages of long-term management and, whilst it would represent a robust limit for all stages, it is over conservative for transport and the operational period of a GDF. The 200W limit is therefore applied to transport and the 150W limit applied only to individual waste packages at the time of backfilling (i.e. 2090). Credit can therefore be claimed for the decay of waste package radiogenic heat output in the intervening time.

Waste packagers should also be aware that large numbers of packages generating heat above $6Wm^{-3}$ (i.e. ~20W per 3 cubic metre Box waste package) could result in excessive temperatures in the post-backfill period and additional assessment of packaging proposals with such heat outputs may be necessary.

4.4 Surface Contamination

The non-fixed surface contamination of the waste package should be as low as reasonably achievable and, when averaged over an area of 300cm² of any part of the surface of the waste package, shall not exceed:

- ***4.0Bqcm⁻² for beta, gamma and low toxicity[8] alpha emitters and;***
- ***0.4Bqcm⁻² for all other alpha emitters.***

Limits on non-fixed surface contamination are specified as a means of ensuring that the contamination of transport systems and waste package handling areas in a GDF are maintained below acceptable levels. The routine handling of contaminated waste packages could lead to a gradual build-up of contamination of transport and GDF equipment with an associated increase in the doses to workers during normal operations and maintenance. This would require decontamination operations which themselves produce liquid and airborne effluents that will require treatment and handling. To ensure that doses to workers and the public are ALARP, and in accordance with good industry practice, limits on surface contamination of waste packages are therefore necessary and those specified above should be considered absolute upper limits.

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.

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[7] The date for backfilling is not fixed but is assumed, for the purposes of this guidance, to be 2090.

[8] 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.

In recognition of the need to minimise the surface contamination of waste packages The NDA has produced specific design guidance on the requirements and the methods for the control of the surface finish of waste containers [14].

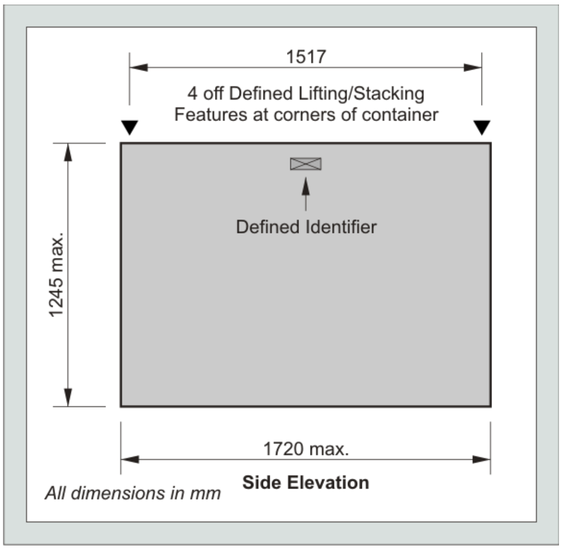
Notwithstanding the above, the proposed radiological classification [15] for a GDF would permit limited numbers of packages with surface contamination levels a factor of 10 higher to be handled. Additionally, work by the National Radiological Protection Board [16] has shown that external doses, inhalation doses and ingestion doses are very low for package surface contamination at the limits quoted.

4.5 Dimensions

The overall dimensional envelope of the waste package shall not exceed:

- ***Height:*** ***1245mm***
- ***Plan:*** ***1720mm x 1720mm***

Figure 2 Standard Features of the 3 cubic metre Box Waste Package



It is essential that all 3 cubic metre Box waste packages fit within a maximum dimensional envelope that is compatible with the systems specified for transport and the various handling systems in a GDF.

Waste containers with standardised dimensions allow for the optimum utilisation of waste packaging facilities, interim surface stores and transport facilities as well as simplifying handling operations at all stages of their long-term management and making most efficient use of GDF vault volume.

The shape, dimensions and lifting/stacking arrangements of the two variants of the 3 cubic metre Box have been chosen to maintain compatibility with the two principle designs of stillage that may be used to handle and stack 2x2 arrays of 500 litre Drum waste packages during storage and, potentially, transport. The key dimensions (i.e. the plan and height) of the two variants are standardised and the only significant difference being the layout of the lifting features.

The corner-lifting variant of the 3 cubic metre Box has dimensions and lifting/stacking arrangements that are standardised with those of the stillage anticipated for use in the repository (the 'disposal stillage') and Sellafield 'compact stillage[9]. The plan dimension is flexible up to a maximum value of 1720mm. Within this standard a plan dimension of 1665mm, maintaining compatibility with the Sellafield compact stillage would also be acceptable.

The upper limit of 1720mm on the plan dimension is set by the loading gauge of the UK rail system and the requirement to transport waste packages in a transport containers with sufficient shielding to satisfy the relevant regulations on external dose rates (Section 4.2).

The overall maximum dimensions given for the 3 cubic metre Box waste package includes any vents, filters or other protrusions from the package.

The base of the 3 cubic metre Box waste package should be flat or, alternatively could have four feet built up by the addition of small steel plates. For either option, consideration should be given to package stability, interface corrosion, potential for wear during transport, and stresses in the package.

One consequence of the flexibility of the key dimensions of this waste package is that the displacement or external volume of the waste package is not fixed and this has consequences for some of the numerical limits that are included in WPS/315. This particularly affects limits that are a function of the transport containers 'free' volume (i.e. the difference between the internal volume of the transport container cavity and the external volume of the waste package) when the waste package is transported. Criteria affected by this variation are those for Gas Generation (Section 4.8), Impact Performance (Section 4.13) and Fire Performance (Section 4.14). To allow these variations to be quantified, waste packages with nominal maximum and minimum displacement volumes have been defined:

- Maximum displacement volume (~3.7m^3) – Plan dimension 1720mm x 1720mm, Height 1245mm.
- Minimum displacement volume (~3.4m^3) – Plan dimension 1665mm x 1665mm, Height 1245mm[10].

It should be noted that, in line with the conservative approach adopted for the setting of waste package standards and specifications, the values for numerical criteria given in WPS/315 are the most onerous of the two values derived for each of these two extreme cases.

4.6 Lifting Feature

The waste package shall incorporate four equally spaced lifting points, in the form of twistlock apertures of dimensions and geometry as defined in Figure 3, located as shown in Figure 4.

The waste package shall be capable of being lifted using any three of the twistlock apertures, without exhibiting any permanent deformation, with a force equivalent to twice the maximum gross mass specified for the waste package.

During their lifetime 3 cubic metre Box waste packages will be handled many times in several locations; in the manufacturing plant, interim surface store(s), during loading and unloading of transport containers, during emplacement at a GDF etc, and accordingly will require specific features to permit safe and efficient vertical lifting. The overall shape of

[9] These two stillage designs are assumed to be the same, at least to the extent that their handling/stacking characteristics and significant dimensions shall be compatible.

[10] These dimensions being those of the Sellafield compact stillage which are deemed to be those of the smallest waste package of this type

the lifting feature at the top end of the package has been precisely specified to ensure that all potential variants of the 3 cubic metre Box waste package can be handled in the same way using the same lifting frame. This is ensured by the specification of a common set of four twistlock apertures with a standard layout as shown in Figure 4.

Figure 3 Twistlock Dimensions and Geometry

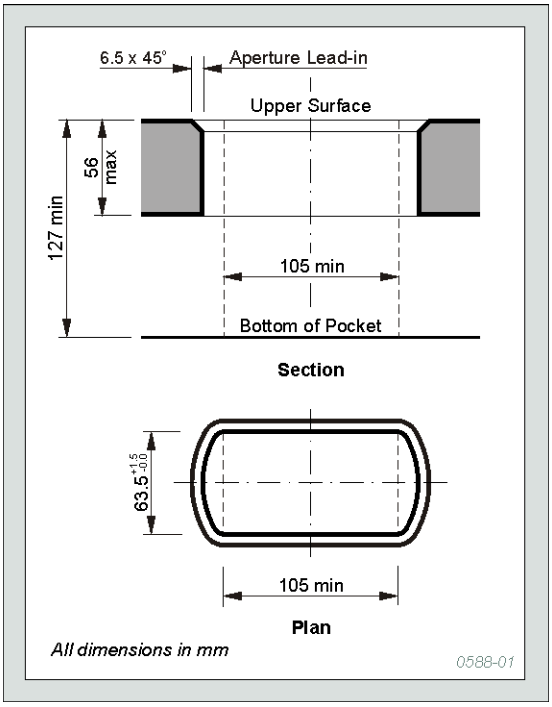
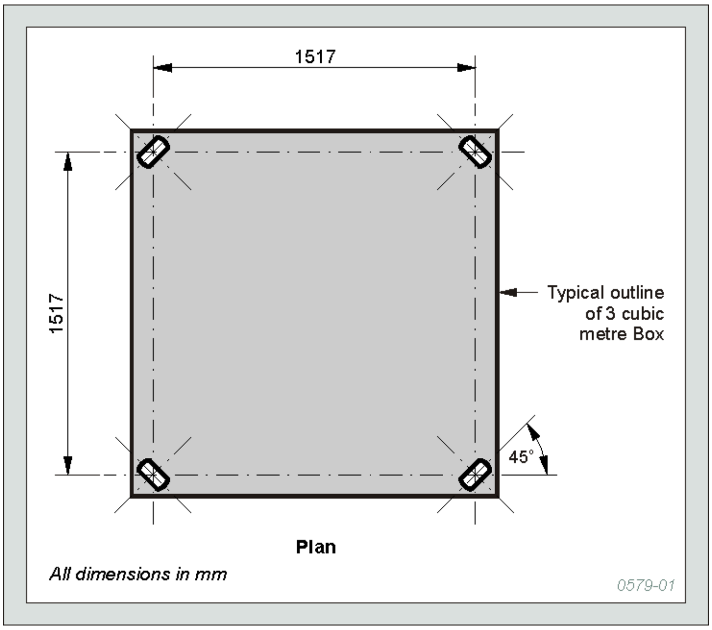


Figure 4 Layout of Lifting Feature of 3 cubic metre Box



The position and orientation of the apertures is in common with that specified for the side lifting variant of the 3 cubic metre Box waste package in order that the same lifting frame can be used to handle either of these two waste packages. Further details on the standard lifting frame are given in *Lifting Frame for 3 cubic metre Drum and side lifting variant of the 3 cubic metre Box. Description and Design Guidelines,WPS/601*.

To comply with GDF lifting requirements, each aperture point must be designed to withstand the weight of the waste package multiplied by a 'snatch factor' of 2, assuming an equi-spaced three-point lift, without any permanent deformation. Three-point lifting is specified to cover the case of uneven distribution of load on the twistlocks. Taken together, and bearing in mind that three-point lifting essentially shares the loading between only two points, each lifting point must be capable of carrying a dynamic load equal to the gross mass of the waste package.

Each waste producer must obviously consider its individual requirements for its own site but must comply with the RWMD specification as a minimum.

Reference should be made to Section 4.10 regarding the integrity timescale of the waste package lifting feature.

4.7 Mass

The gross mass of the waste package shall not exceed 12,000kg.

The maximum allowable mass of waste packages is set by a combination of constraints imposed by the transport system and a GDF handling systems.

Waste packages can be transported by road or rail.[11] Most of the UK rail system limits axle loading to 22.5t per axle, which leads to a maximum loaded rail wagon mass of 90t for a four-axle wagon. As currently designed, the rail wagon anticipated for use in transporting waste packages has an unladen mass of 26t which limits the maximum transport package mass to 64t. The heaviest transport container currently under consideration for such transport, the SWTC with a nominal 285mm of shielding (SWTC-285), has an unladen mass of 52t which sets a limit of 12t on the total mass of waste packages carried within it.

4.8 Gas Generation

The total gas generation rate for the waste package should not exceed 72 litres per day.

Specific limits are placed on the rates of generation of certain toxic, flammable and radioactive gases.

Gases may be generated in wasteforms by a variety of processes including:

- chemical processes, such as corrosion;
- microbial degradation of organic materials;
- radiolysis of water and organic materials;
- radioactive decay producing gaseous products (e.g. Rn);
- release of entrained radioactive gases (e.g. H-3, Ar, Kr).

Gases give rise to a range of potential effects that may have an influence on all stages of the long-term management of waste packages. These include:

- pressurisation and damage of the wasteform, leading to increased release of radionuclides under normal and accident conditions;

[11] The PGRC also considers the possibility of transport by sea, although no specific constraints would arise from that transport mode.

- pressurisation of un-vented waste packages, leading to distortion and/or damage to the waste container;
- pressurisation of the transport container;
- releases of radioactive/toxic/flammable gases from packages;
- alteration of the chemical characteristics of the backfill;
- pressurisation and damage to the surrounding geology;
- generation of additional groundwater flow pathways and modification of flow patterns;
- modification to the rate of re-saturation of backfilled vaults.

4.8.1 Total Gas Generation

The limit for total gas generation by 3 cubic metre Box waste packages is set by the regulatory requirement to prevent the internal pressure of the transport container from exceeding the Maximum Normal Operating Pressure (MNOP) of 700kPa[12].

Box 1 illustrates how a limit for the maximum total gas generation by a 3 cubic metre Box waste package is calculated. The basis for the calculation is that the transport container remains sealed for 28 days, this being maximum anticipated time between sealing and reopening (or controlled venting) and includes the conservatism that the temperature on sealing is the minimum permitted during Normal Conditions of Transport (NCT) and that on opening is the maximum.

Box 1 Calculation of allowable gas generation rate for 3 cubic metre Box waste package carried in SWTC-285

Transport container 'free volume' (i.e. SWTC-285 cavity volume less displacement volume of 3 cubic metre Box waste package) = $0.391m^3$

Atmospheric pressure on SWTC sealing = 1.013×10^5Pa

Temperature on sealing = 263K*

Quantity of gas in cavity on sealing = $\dfrac{1.013 \times 10^5 \times 0.391}{8.31 \times 263}$ = 18.1 moles

Assuming pressure increases to 700kPa (MNOP) = 8.013×10^5 Pa

Assume temperature increases to 348K**

Quantity of gas in cavity on opening = $\dfrac{8.013 \times 10^5 \times 0.391}{8.31 \times 348}$ = 108.4 moles

Quantity of gas generated = 108.4 – 18.1 = 90.3 moles

= 90.3 x 0.0224 = $2.02m^3$

Maximum gas generation rate for 28 day journey = 2.02/28 = $0.072\ m^3$/day

= 72 litres/day

* This being the Minimum Temperature permitted under NCT.

** This being the Maximum Temperature permitted under NCT.

12 All pressures quoted in this document are gauge pressures.

This results in a conservatively based value of 72 litres/day for a 3 cubic metre Box waste package with the maximum volume permitted by the dimensional specification when carried in a transport container with the smallest free volume (i.e. the SWTC-285). Since the limit is a function of the free volume in the transport container cavity, waste packages with a smaller volume, or use of a transport container with a larger internal volume (e.g. the SWTC-70), would result in a relaxation of this limit.

The limit on total gas generation does not take into account the nature of the gas involved. There are more stringent limits on a number of flammable, toxic or radioactive gases as identified below and listed in Appendix A.

The numerical limits placed both on total gas generation and on the generation of the three categories of hazardous gases discussed below will apply at the time of transport of waste packages from the waste packager's site. It is not envisaged that this will take place before 2040, although some packages may be transported to other licensed sites at an earlier date. Waste packagers may therefore choose to take credit for the ageing of wasteforms as it might affect gas generation.

4.8.2 Flammable Gases

Many wastes have the potential to generate flammable gases, notably hydrogen, and controls must be imposed to prevent the accumulation of explosive mixtures of such gases.

In the case of flammable gases, controls can be placed on their rates of generation and/or on the operational procedures adopted for the transport container, the former being the preferable option. Maximum allowable generation rates have therefore been calculated on the basis that the concentration of a flammable gas in the transport container cavity must not exceed 75% of its lower flammability limit in air, within the maximum anticipated transport period of 28 days. Table A.1 lists maximum gas generation rates for a range of flammable gases, given in terms of litres/day per 3 cubic metre Box waste package.

For waste packages generating gases at rates greater than those given in Table A.1, there is the option of purging the transport container cavity with nitrogen before transport, in order to remove atmospheric oxygen and thus eliminate the possibility of ignition. If this procedure is adopted, a limit must also be placed on the allowable rate of oxygen generation by the waste packages; this too is listed in Table A.1.

4.8.3 Toxic Gases

Limitations on toxic gas generation are determined on the basis of limiting the toxic gas concentrations in confined storage areas containing a transport container to levels deemed safe for a worker in that area. Table A.2 lists those toxic gases whose allowable generation limits are less than the total gas generation limits given in Section 4.8.1 above.

4.8.4 Radioactive Gases

Radioactive gases are generated by wasteforms by three principal mechanisms:

- release of entrained tritium and noble gases (e.g. Ar, Kr and Xe) by diffusion or corrosion;
- radioactive decay of radium and the release of radon;
- chemical production of 'labelled' gases (e.g. CO_2, CH_4).

Limits on the release of radioactive gases are imposed by the IAEA Transport Regulations [5] which state that the release of activity from a Type B transport package should not exceed $10^{-6}A_2hr^{-1}$ under NCT. If this value is applied directly to releases of gaseous activity and is combined with the maximum anticipated leakage rate from a SWTC, it results in a limit of $7.7 \times 10^{-4} A_2 day^{-1}$ per 3 cubic metre Box waste package, for all gaseous radionuclides except tritium[13].

Table A.3 lists the allowable activity release rates for a range of radioactive gases released by a 3 cubic metre Box waste package, when transported in a SWTC-285. These limits are for each of the gases in isolation and the method of considering the consequences of mixtures of gases can be found in [17].

In the particular case of Rn-222, a radioactive gas produced by the radioactive decay of Ra-226, control of the generation of this gas by waste packages is primarily by control of the quantity of Ra-226. A consideration of the packaging of radium bearing wastes can be found in *Guidance on Packaging of Radon-generating Wastes*, WPS/901. This gives a maximum allowable Ra-226 inventory is derived for 3 cubic metre Box waste packages above which appropriate measures will need to be taken to limit the release of Rn-222 from the wasteform.

Limits on the allowable releases of radioactive gases can be considered conservative, as they give no credit to number of beneficial factors such as:

- the low pressure drive resulting from the small pressure difference across the transport container seal;
- radioactive decay of short-lived gaseous radionuclides;
- possible venting and/or purging of gases prior to shipment.

These factors may be taken into account during the assessment of waste packages whose predicted releases of radioactive gases are in excess of the values given in Table A.3.

4.9 Venting

Waste packages that, by virtue of the nature of their container and/or contents, could be susceptible to pressurisation due to gas generation at any time during their long-term management, shall incorporate an engineered vent designed to retain significant particulate activity.

The generation of gas within a sealed waste package under normal conditions (see Section 4.8) or accident conditions (in particular, fire) could lead to pressurisation of the waste container and the wasteform, unless steps are taken to avoid it. Pressurisation can lead to swelling, damage to the structure of the wasteform and eventual failure of the waste container. This could compromise the integrity of the barriers provided by the waste package against the release of activity and leads to the requirement for certain waste packages to be provided with a vent. Section 4.8.2 highlighted the possibility of the generation of flammable gases by certain wastes, and the provision of a vent helps to reduce the risk associated with the accumulation of such gases within waste packages.

The requirement to reduce the possibility of waste package pressurisation is important at all stages of their long-term management along with the requirement to minimise the release of particulate activity. This leads to the requirement for the vent to be filtered, which could for example be achieved by the use of a proprietary high efficiency particulate in air (HEPA) or sintered filter as part of the vent, or by using a lidding arrangement that incorporates a device such as a labyrinth seal.

The requirement for venting does, however, potentially conflict with a requirement to minimise ingress of water into waste packages in the post-closure period of a GDF. This

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[13] Because tritium has a significant rate of permeation through the elastomeric material of the transport container seals, its limit is more restrictive: $1.9 \times 10^{-5} A_2 day^{-1}$.

requirement should be taken into account in vent and filter design and the effective area of the vent minimised.

Precautions should be taken in the waste container design to ensure that there is no alternative gas pathway that could bypass the filtering feature (e.g. through an ineffective body/lid seal), particularly during the earlier, more reactive phases of wasteform evolution.

The following are guidelines on the need for the venting of waste packages and the general requirements of such a system, if it needs to be included in a waste package design:

- Waste packages should be vented if gas production by the wasteform, over the period during which the waste package will need to be handled, is considered capable of causing pressurisation of the waste container.
- Un-vented waste packages should be sufficiently leak-tight such that they are capable of satisfying the requirements for retention of activity under normal handling conditions or under specified impact and fire accident conditions (Sections 4.13 and 4.14).
- The design of a venting (and filtration) system should not compromise the ability of the waste package to satisfy the requirements for retention of activity under normal handling conditions or under specified impact and fire accident conditions (Sections 4.13 and 4.14).
- When considering designs of venting systems, waste packagers should take into account the long-term integrity requirements for the waste package (Section 4.10). This should include the longevity of the filter medium under the anticipated conditions of waste package storage.
- The cross-sectional area of the vent should be as small as possible while still satisfying the required performance criteria.
- The sealing of waste packages with a filtered vent should be sufficiently leak-tight to ensure that the filter performance is not compromised by alternative gas pathways.
- The filter should be able to cope with the maximum gas production rate anticipated under normal conditions.
- The dust-holding capacity of the filter should be such that it would be capable of operating with optimum performance over the envisaged storage period and with the potential levels of particulates.
- The filter should be able to satisfy the required performance criteria at temperatures up to 80°C.

4.10 Integrity

The integrity of the waste package shall be such that it is capable of retaining its contents and of being moved and handled safely and efficiently, as required, during all stages of long-term management.

The waste packages should be designed so that:

- ***following a period of interim surface storage, currently assumed to be up to 150 years, the waste package shall meet the requirements for handling and for transport to a GDF;***

- ***following emplacement in a GDF, the waste package should be capable of maintaining its integrity for the operational period, currently assumed to be 50 years;***
- ***upon cessation of a GDF operational period, the waste package should retain integrity during a period of care and maintenance, during which time the waste package must be capable of being retrieved and safely handled. This period could extend to a few hundred years;***
- ***following the period of care and maintenance, a GDF may be backfilled. The waste package should continue to retain its integrity for a period consistent with the containment of short-lived soluble radionuclides.***

A period of 500 years should be considered a target for the integrity of the waste container.

Integrity is defined, in this context, as the ability of a waste package to maintain the containment of its contents, and to maintain the surety of its physical handling features (i.e. lifting locations). The timescale for this requirement is discussed in Section 4.10.1, and is set by the need for waste packages to enter the post-closure period in good condition, and it therefore needs to encompass the periods of interim surface storage, transport, GDF operations and vault backfilling.

Although containment is provided by both components of a waste package – the waste container and the wasteform – the barrier provided by the waste container is particularly important in limiting operational discharges, i.e. liquid and airborne effluents, during the earlier stages of management. The integrity of the waste container must be maintained over the required timescale by:

- appropriate design;
- selection of suitable materials;
- appropriate manufacturing processes;
- provision of appropriate storage environments.

4.10.1 Integrity Timescale

Initially, waste packages will need to remain in interim surface storage for the period until a GDF becomes available. Previously it had been recommended that this storage period be assumed to be 50 years. However, in the light of delays to a GDF programme, a longer period should be assumed as a precautionary measure. In the report on a joint study by the Radioactive Waste Management Advisory Committee (RWMAC) and the Nuclear Safety Advisory Committee (NuSAC) [18] it was noted that:

> *'The main waste producers, the regulators and Nirex all now broadly accept that it would be prudent to plan for a period of interim storage of the order of 100 to 150 years'*

Designing for an interim storage period of 150 years is therefore consistent with such a belief. During this period, waste package physical and structural integrity should be maintained to ensure that it retains its contents and is suitable for subsequent stages of their long-term management. Advice on the appropriate control of the storage environment during this period can be found in *Guidance on Environmental Conditions during Storage of Waste Packages, WPS/630*.

The reference operational strategy for a GDF currently assumes an operational period of 50 years, during which waste packages will be emplaced in the vaults. During this time the package environment will be controlled in accordance with the requirements specified for the vaults in the GDSS [10].

After all of the waste packages have been emplaced in a GDF, future generations will have the option to keep the facility 'open' for a further period of underground storage involving care, maintenance and monitoring, and potential retrieval of waste packages. During that period, the environmental controls maintained during the operational period would be continued until it was decided to backfill the vaults. Waste packages would be required to continue to retain their integrity during that period so that, if required, they could still be retrieved safely for inspection, using the waste package handling feature and standard equipment.

For a disposal concept based on multiple containment barriers there will be no explicit reliance on the waste containers continuing to maintain their integrity after the vault has been backfilled and the chemical barrier provided by the backfill material has been established. However the physical barrier provided by the waste container would still be considered the best practical means of preventing the release of such soluble short-lived radionuclides as Sr-90 and Cs-137 ($T_{½}$ ~30 years) from the engineered system. A period of approximately ten half-lives (i.e. ~300 years) following backfilling is considered sufficient [19] for this purpose. Such a timescale would also provide a more extended period during which future generations would be able to retrieve packages safely for inspection.

Considering the possible durations of all the stages of long-term management discussed above, a requirement for integrity to be maintained for 500 years should be the target for all waste containers.

4.10.2 Threats to Container Integrity

Corrosion is the major potential threat to waste container integrity. In response to this, and to the timescale requirement identified above, the commonly adopted (although not universal) solution is to manufacture containers from austenitic stainless steel to grade 316L (EN 1.4404 [20]) or its equivalent. The corrosion performance and mechanical properties of this material are generally regarded as optimum for the packaging of radioactive waste, and this performance has been demonstrated by experience and research [21]. 'Duplex' stainless steel (notably grade EN 1.4462) has been identified as an alternative material that has the necessary corrosion performance to make it suitable for the manufacture of waste containers. Whichever material is selected it should be noted that quality control of the material, the container manufacturing process and the control of surface finish of the container will also play a fundamental role in maintaining the integrity of the waste container.

A high pH environment is considered to be beneficial in reducing corrosion. A waste conditioning matrix that does not produce high pH conditions could accelerate corrosion.

Stainless steel has various corrosion mechanisms, dominated by:

- general corrosion;
- pitting/crevice corrosion;
- stress corrosion cracking.

The general atmospheric corrosion performance of stainless steel is widely reported [22] and corrosion rates from <0.2μmy^{-1} (>5,000ymm^{-1}) to 3μmy^{-1} (300ymm^{-1}) have been observed in industrial/urban and marine environments. Initial measurements from longer-term testing suggest corrosion rates of ~0.01μmy^{-1} (100,000ymm^{-1}) which can be extrapolated to container lives well in excess of the longest integrity requirements imagined for a GDF.

Pitting or crevice corrosion, although regarded as localised corrosion mechanisms, are considered a greater threat to stainless steel waste packages than general corrosion. Crevices can be formed between container components, between the wasteform and the inside of the container, between packages when stacked, and/or in the container lid area. Container and package designers should therefore bear in mind the requirement to minimise the creation of such crevices.

Nevertheless, data extrapolated from tests [23, 24] have shown that the time for a pit to penetrate 1mm into 316L stainless steel is 700-1000 years. Localised corrosion mechanisms are dependent upon the presence of surface contaminants, in particular, chlorides. Work has been carried out to investigate these effects and to specify requirements for, amongst other factors, the surface finish of stainless steel used for waste containers [14]. Since the molybdenum content of stainless steel is instrumental in preventing the onset and propagation of pitting corrosion, 304 grade stainless steel, which has a lower molybdenum content than 316L grade, is not recommended for the package primary containment. However, the general corrosion performance of 304 grade stainless steel is good, so it is regarded as suitable for structural features of the container such as the lifting/stacking feature, internal package furniture etc whereas 316L grade is more suitable for thinner sections such as the container walls, base and lid.

Atmospheric stress corrosion cracking is dependent on the presence and concentration of soluble chloride deposits, the chemical form of the chloride, temperature, relative humidity and the metallurgical state of the stainless steel [22]. Such corrosion of stainless steel can be accelerated at temperatures above 60°C and may also be significant at lower temperatures. The chloride content therefore should be kept to a minimum and careful consideration given to possible corrosion mechanisms if it exceeds 100ppm. Consideration should be given to mechanisms for the generation of chloride ions, e.g. by the radiolysis or thermal breakdown of chlorine-containing plastics.

Stress corrosion cracking during interim surface storage is not regarded as a significant threat to package integrity, because control of the environmental conditions the stores (as specified in *Guidance on Environmental Conditions During Storage of Waste Packages, WPS/630)* will help to eliminate the conditions under which stress corrosion cracking could occur.

Another pre-requisite for this type of localised corrosion is access by oxygen to the surface of the container material. The elimination or reduction of internal voidage, ullage or gaps between the waste matrix and the container skin can help reduce oxygen access, and also reduce the possibility of water condensation on internal surfaces.

Backfilling will also limit access by oxygen and other contaminants to the external surfaces of the container. In the subsequent period following re-saturation by groundwater, access by oxygen will be further hindered because incoming groundwaters are likely to have a low oxygen concentration due to chemical reduction in the surrounding rock (although oxygen may still be generated locally by radiolysis). The high pH of the porewater in the surrounding backfill will also tend to inhibit corrosion, because the higher the pH, the higher are the chloride levels needed to initiate localised corrosion.

Corrosion inside the waste container can also be accelerated by electrolytic action with dissimilar materials, or with other aggressive components that may be present in the package. Particular consideration should be given to preventing the possibility of metal items in the wasteform from contacting the container walls directly.

Intergranular corrosion or ‘weld decay’ can occur in austenitic stainless steel that has been ‘sensitised’ by the high temperatures experienced during welding. The risk of sensitisation is minimised by use of low carbon or stabilised grades of stainless steel. Nevertheless, excessively high heat inputs should be avoided, as should contamination of the weld by material containing carbon or nitrogen.

4.10.3 Summary

When compared with thickness of the containment typically used for the 3 cubic metre Box (i.e. ~5-10mm) the corrosion rates quoted for the processes identified above do not appear to threaten an integrity target of 500 years. Such a conclusion assumes that container material selection, construction techniques and storage conditions after manufacture are in line with best practice. To assist waste packagers in these areas, guidance has been produced on the general corrosion properties of stainless steel [21], the requirements for surface finish [14] and on welding techniques used during the manufacture of stainless steel containers [25].

4.11 Properties of the Wasteform

All reasonable measures shall be taken during the production of the wasteform, and the interim surface storage of the waste package, to ensure that:

- ***radionuclides in the waste are immobilised;***
- ***loose particulate material is minimised;***
- ***free liquids are excluded;***
- ***hazardous materials are excluded or made safe;***
- ***toxic materials are minimised;***
- ***any gases generated do not result in pressurisation of the wasteform;***
- ***the presence and volume of voids (e.g. ullage, holes etc) is minimised.***

The measures taken to achieve these objectives should include an anticipation of the effects of ageing on the performance of the wasteform.

The Specification of the wasteform for the 3 cubic metre Box waste package can be found in *Specification of Wasteform for 3 cubic metre Box and 3 cubic metre Drum Waste Packages, WPS/510* and *3 cubic metre Box and 3 cubic metre Drum Wasteform Specification: Explanatory Material and Guidance, WPS/810.* This section is therefore limited to guidance on the higher-level requirements given in the *3 cubic metre Box Waste Package Specification, WPS/315.*

4.11.1 Immobilisation of Activity

The main function of the wasteform is to ensure that the waste is rendered passively safe which includes ensuring that activity is not present in a mobile form (gaseous, liquid or fine particulate) which would be more readily released during normal or accident conditions. This can be achieved in the case of liquid or sludge waste by mixing with a suitable binder material and allowing the product to set into a solid mass. Special measures may have to be taken to reduce the mobility of gases, depending on the chemical nature of the gas. Guidance on the need for and means of achieving immobilisation can be found in *Guidance on the Immobilisation of Radionuclides in Wasteforms, WPS/903.*

After emplacement the wasteform provides a diffusive barrier to the escape of radioactivity from the package. To ensure the effectiveness of this barrier, the waste packager should seek to manufacture a wasteform which is, as far as possible, monolithic and of low permeability to gas and water.

4.11.2 Mechanical Strength

In a GDF, 3 cubic metre Box waste packages will be stacked up to seven high without additional means of support. The container itself may not be strong enough to guarantee stability of such a stack without some degree of assistance from the wasteform and waste packagers should therefore seek to achieve a wasteform which provides positive support to the walls of the container. This has been shown to be achieved with a wasteform which has a minimum unconfined compressive strength of 4MPa.

The performance of a package under accident conditions, particularly those involving impacts, is also dependent on the mechanical properties of the wasteform. The waste packager should also design the wasteform to ensure that it responds to the challenges of impact or fire accidents in a progressive and predictable manner. In practice, this can be achieved by ensuring that all activity is well dispersed throughout the solid matrix, and that there are no weak areas in the package where activity is present in potentially dispersible form. Static compressive strengths in the range 4 to 40MPa are deemed satisfactory in this context.

4.11.3 Voidage

Voidage can occur as residual volumes within the waste or wasteform that have been incompletely filled by the encapsulating material, either within the conditioned wasteform, or between the container skin and wasteform. Other examples of voidage include ullage (empty space beneath the lid), enclosed volumes within waste items, and spaces between waste items, waste package furniture etc. that have been poorly infilled with grout. In a different sense, the natural porosity of the cementitious grout can also be considered to be 'voidage'.

Excessive voidage within a waste package has a number of disadvantages which include:

- the potential to make the package weaker and so more susceptible to damage and break-up in the event of impact;
- the possibility of locally accelerated corrosion of the container material, particularly when the material is stressed and when condensation may form in the void;
- the creation of pockets where flammable mixtures might collect;
- the creation of weak areas that might lead to deformation of the package under hydrostatic pressure, which could result once the vault has become re-saturated by groundwater;
- the possibility of interconnected voidage in the vault, which could lead to increased groundwater flow.

It is therefore desirable that voidage be eliminated from waste packages. However, it is also recognised that it will be impractical to achieve zero voidage, and that some limited voidage will be acceptable the tolerable extent of which will be dependent on the nature of the package contents and other features of waste package design. The waste packager must however demonstrate, in the packaging submission, that any voidage within a waste package has been minimised, and that any unavoidable residual voidage does not prejudice the ability of the package to meet the other requirements of the WPS.

4.11.4 Other Restrictions

The properties of the wasteform need to be consistent with all other aspects of the design and safety cases for a GDF and transport system. This may involve specific restrictions on such matters as toxic materials, fissile materials, gas generation, thermal conductivity, activity content, organic materials and oils. For guidance on these requirements, which are related primarily to the nature of the waste rather than the form of packaging, reference should be made to *Specification of Wasteform for 3 cubic metre Box Waste Package, WPS/510.*

4.12 Criticality Safety

The presence of fissile materials[14], neutron moderators and reflectors in the waste package shall be controlled to ensure that they do not present a criticality safety hazard during any of the active stages of their long-term management.

It shall also be ensured that, following closure of a GDF, the possibility of local accumulation of fissile material such as to produce a neutron chain reaction is not a significant concern to the long-term performance of a GDF.

Waste packages must not represent an unacceptable criticality safety hazard, either individually or in arrays, during any stage of their long-term management. A criticality incident involving waste packages during transport or the operational period of a GDF would result in substantially increased heat output, changes to the radionuclide inventory and elevated dose rates. Such an event would therefore present an immediate hazard to the public or workers. During the post-closure period of a GDF, the increased generation of heat could compromise the effectiveness of the barriers to radionuclide release from individual packages and from a GDF.

The RWMD approach to criticality safety is outlined in [26] and is based upon the production of 'benign packages' containing insufficient fissile material for criticality to occur, even in worst-case conditions. This is achieved by controlling the package design, including the quantities of both fissile and moderating materials, to help eliminate the potential for criticality either in individual packages or in assemblies of packages, during routine transport and operations at a GDF. In accident conditions, the physical robustness of waste packages is such none of the credible accidents considered would result in criticality.

This has established the maximum allowable levels of fissile material in waste packages with respect to criticality [27] and this has led to the derivation of a generic package screening level of 50g Pu-239 or equivalent, a value supported by conservative assumptions and calculations. This value is defined as a level below which individual and groups of standard packages, containing undefined waste, will be sub-critical under all circumstances.

Waste packages containing quantities of fissile material at or below the screening level may be considered benign under all circumstances, but those containing larger quantities will need further consideration. Methodologies exist for the control of package design and contents to ensure that these packages would also be benign from a criticality point of view.

A methodology has been devised for producing Criticality Safety Assessments (CSAs) that cover waste package transport, GDF operations and post-closure conditions. Generic and package-specific criticality assessments form an integral and coherent part of the generic safety assessments. These assessments will identify general screening levels for waste packages containing four common categories of fissile material:

- low enriched uranium (LEU);
- highly enriched uranium (HEU);
- plutonium-contaminated material (PCM);
- irradiated natural uranium (NU).

These waste categories are defined mainly by the isotopic and chemical compositions of the uranium and plutonium in the wastes, but they may also be bounded by the types of waste container and methods of waste conditioning that are likely to be used, and by the

[14] Defined as U-233, U-235, Pu-239 and Pu-241 but excepting unirradiated natural or depleted uranium and natural or depleted uranium that has been irradiated in a thermal reactor.

range of other materials likely to be present. The four general assessments should be applicable to a large fraction of the wastes that contain fissile material.

Recently, it has been agreed with the EA and the Nuclear Installations Inspectorate (NII) to update the methodology underpinning the determination of the screening level. The objective is to:

- derive post-closure screening levels based upon an approach that considers both conservative and more-credible assumptions with regard to the likely post-closure evolution of the system;
- apply to criticality safety assessment of the operational period principles from approaches being developed by BNFL in association with NII; this methodology is being developed for application to storage, deriving lower and upper package limits;
- develop waste transport criticality safety assessments through interactions with the Department for Transport.

This additional work is to allow arguments about the 'balance of risk' to be developed, with the aim of ensuring that any agreed limits are proportionate and not unduly restrictive.

To date, this process has resulted in the elicitation of indicative values for a conservatively derived Lower Screening Level (LSL) and an Upper Screening Level (USL) based on more credible assumptions, for each of the four categories for the post-closure period of a GDF. The generic screening level (i.e. 50g Pu-239 or equivalent) is retained for waste packages containing fissile materials not covered by the four categories (e.g. U-233[15]). Table 1 lists the indicative values of LSL and USL for the four categories of fissile material.

Table 1 Indicative values for LSL and USL for Fissile Material Categories.

Category of Fissile Material	LSL	USL
Irradiated natural uranium (g Pu-239 eq)	200	4000
Low enriched uranium (g U-235)	200	600
Highly enriched uranium (g U-235)	100	300
Plutonium contaminated material (g Pu-239)	300	900
Other fissile materials (g Pu-239 eq.)	50	-

[15] The advisory material for the IAEA Transport Regulations [5] also identifies Np-237, Pu-238, Pu-240, Pu-242, Am-241, Am-242m, Am-243, Cm-243, Cm-244, Cm-245, Cm-247, Cm-249 and Cf-251 as materials '*that have the potential for criticality'* although the presence of such materials is only considered to constitute a hazard if they have been deliberately separated from irradiated fuel.

4.13 Impact Performance

The waste package should be designed such that, in the event of an impact accident:

- ***releases of radionuclides and other hazardous materials are low and predictable, exhibit progressive release behaviour with increasing impact severity and do not exhibit significant cliff-edge performance characteristics within the anticipated range of impact conditions;***
- ***both barriers to radionuclide release from the waste package (i.e. the waste container and the wasteform) should play an effective role in minimising those releases.***

The waste package shall be capable of being dropped, in any attitude, from a height of 0.3 metres onto a flat unyielding surface, whilst retaining its radioactive contents, and afterwards shall remain suitable for safe handling during all subsequent stages of their long-term management.

The release of radioactive contents from the waste package, as a result of credible impact accidents during transport and operational period of a GDF, shall not result in the relevant regulatory dose limits to workers and to members of the public being exceeded.

Impact accidents can affect waste packages at a number of stages of their long-term management and are a mechanism by which the radioactive contents of a package could be released into the environment in an uncontrolled manner. Accordingly, waste packages must be capable of withstanding a number of specified impact conditions without excessive loss of contents in order that they will comply with regulatory requirements and with the assumptions which underpin the safety assessments for transport and the operational period of a GDF.

RWMD has defined three types of impact accident to which waste packages could be exposed:

- Minor impacts resulting from normal handling;
- Impacts resulting from transport accidents;
- Impacts resulting from accidents in a GDF.

4.13.1 Impacts resulting from normal handling

Waste packages may be subject to knocks, collisions and rough handling in the course of normal handling operations at any stage of their long-term management. It is expected that all waste packages should be sufficiently robust to withstand such impacts and, following external examination, should be capable of onward management without repair or rework. According, following such impacts, waste packages must continue to be capable of retaining their radioactive contents, and must remain compatible with the transport and GDF systems (although it is accepted that non-standard handling arrangements may have to be used).

Whilst the possible nature of such impacts could be very varied, RWMD has adopted a drop height of 0.3m as being suitably representative of such impacts. The selection of such a height is somewhat arbitrary but it is felt that impacts of a greater severity than that resulting from a drop from ~0.3m would not be considered ‘normal’ and should lead to an inquiry as to the cause of the incident and to a more extensive examination of the waste package(s) involved. The basis for quantifying the value is partly justified by way of consistency with Paragraph 722 of the IAEA Transport Regulations [5] where a free drop from 0.3m is required to simulate NCT for transport packages with gross masses greater than 15t.

The objective of the requirement for the retention of waste package radioactive contents is to ensure that contamination of transport and GDF systems is minimised along with any associated dose to workers and the general public. The interpretation of 'retaining radioactive contents' is not quantified in the GWPS as it will depend on the stage in their long-term management at which it occurs, and the ongoing requirements for safe handling, storage, transport and eventual disposal. For guidance it can be assumed that a total (i.e. in the form of gaseous and particulate activity) release of less than $10^{-6}A_2$ per hour is likely to be acceptable, this being the IAEA Transport Regulations limit for release from a transport package under NCT.

4.13.2 Impacts resulting from transport accidents

It is anticipated that 3 cubic metre Box waste packages will be transported through the public domain within a reusable shielded transport container as a Type B transport package. Under the IAEA Transport Regulations, such a transport package is required to be capable of a withstanding a range of mechanical and thermal challenges. These challenges are specified in such a manner as to define a 'transport accident' comprising an impact followed by a fire. In the week following such an accident the total activity released from the transport package must be not exceed A_2[16].

The mechanical challenges to the transport package comprise a free drop, in its most damaging orientation, from a height of 9m on to a flat horizontal surface, a free drop from 1m on to an aggressive target and a dynamic crush test. These challenges are intended for complete transport packages (i.e. with the waste package protected by the transport container). Whilst the transport container itself will be subjected to a programme of modelling and testing to demonstrate that it is sufficiently robust to withstand the regulatory requirements, an additional challenge is specified for the unprotected waste package. RWMD has adopted a conservative approach to ensuring adequate impact performance by waste packages by specifying a free drop from a height of 10m drop on to a flat unyielding surface as a conservative equivalent for the regulatory mechanical challenge for transport packages. As a result of such an impact, the release of activity from the unprotected waste package should not exceed that which would result in the regulatory limit of A_2 in the week following an accident being exceeded by the transport package.

Box 2 illustrates how the allowable activity release from a waste package can be calculated when such a waste package is held within a transport container.

Since the testing regime specified by the IAEA Transport Regulations requires that the activity release rate of A_2 per week shall not be exceeded following the mechanical and thermal tests in succession (in order to simulate the anticipated nature of a transport accident). To acknowledge this, RWMD has allocated half of the allowable releases from the bare package to the mechanical challenge, and half to the thermal challenge. This results in an allowable impact release of $7.0A_2$ from the contents of the transport container and a rounded allowable waste package release of $3A_2$.

It should be noted that this value is very conservative in that, among other pessimisms, it is assumed that the internal pressure of the transport container is at its maximum allowable (i.e. the MNOP). As discussed in Section 4.8, the MNOP will only be reached if each of the waste packages being carried is generating gas at the maximum allowable rate (i.e. 72 litres per day) and if the transport container remains sealed for 28 days. As most waste packages are expected to generate gas at considerably lower rates (a few litres/day is considered extreme), the impact release criterion can be significantly relaxed in most instances. Figure 5 illustrates the degree of relaxation that is possible if this conservatism is removed. The calculation also assumes that the waste package has the

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[16] Or not more than $10A_2$ for Kr-85.

maximum volume permitted by the dimensional specification, carried in the transport container with the smallest cavity volume (i.e. the SWTC-285). This combination will result in the minimum free volume in the transport container cavity and a correspondingly conservative allowable activity release limit.

Box 2 Calculation of allowable activity release from 3 cubic metre Box waste package following transport accident

Post-accident standardised leak rate of transport container seal = $10^{-3}Pam^3s^{-1}$

Assumed internal pressure of transport container = 700 kPa*

Volumetric leak rate of gas from transport container cavity = $3.25x10^{-4}m^3/hr$

This value equates to a total leakage of $0.055m^3$ in the week following the accident, assuming no drop in pressure.

Activity concentration in transport container cavity = $A_2/0.055 = 18A_2m^{-3}$

'Free volume' (i.e. cavity volume less displacement volume of waste package)

= $0.391m^3$

Maximum allowable activity in cavity = 18 x 0.391 = $7.0A_2$

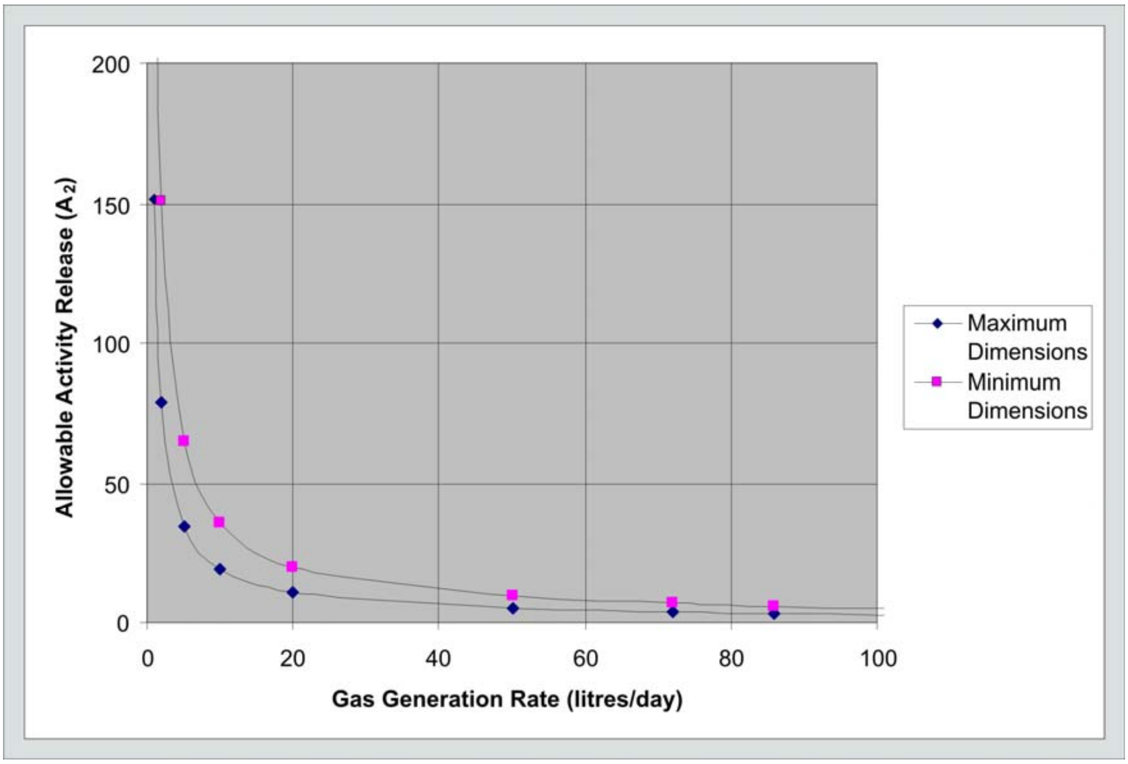
Assuming half of activity release comes from impact and half from fire:

Allowable release from a 3 cubic metre Box waste package

= ½ x 7.0 = $3.5A_2$

* This being the Maximum Normal Operating Pressure (MNOP) permitted under NCT.

Figure 5 Effect of gas generation rate on allowable impact release



A further pessimism incorporated in the calculation shown above is that it is assumed that all of the activity released from a waste package is in a form that can escape through the leaking seal of a transport container and can result in dose to persons in the vicinity of the accident. In practise only gaseous and particulate activity small enough to escape from the transport container needs to be considered. Additionally only particles small enough to be classified as being ‘respirable’ will have significant radiological consequences and larger particles can be ignored.

With regard to the first point, the post-accident standardised leak rate of $10^{-3}Pam^3s^{-1}$ can be shown to be the equivalent of a leak through a single capillary hole of diameter ~30μm in the transport container seal. This is a worst-case figure, which assumes that the leak is via a single circular hole, and that such a hole would not become blocked by escaping particles. Accordingly particles larger than 30μm could not escape from the transport container and would not contribute to dose. A further pessimism exists as it is assumed that no adhesion of particles will take place within the transport container cavity. Such an effect could result in particles significantly smaller than 30μm agglomerating to form particles too large to pass through even the largest anticipated hole in the transport container seal. Work to investigate this effect and the benefit that it could bring has been conducted and is currently being reviewed to determine whether this benefit can be justifiably claimed.

The matter of what constitutes a respirable particle is considered by the International Commission on Radiological Protection (ICRP) in [28] from which RWMD has concluded that particles with a size less than 20μm should be considered respirable.

In view of the above RWMD currently takes the conservative view that all particles with sizes up to 40μm should be considered in any assessment of the consequences of transport accidents. It should however be noted that this assumption is under review and that relaxation of this criterion may be possible in due course.

4.13.3 Impacts resulting from accidents in a GDF

Following receipt at the disposal facility, waste packages will be subject to a series of lifting and handling operations, leading to their emplacement in the vaults. During this period the possibility exists for accidents which could result in waste packages being subject to a range of mechanical impacts. These include:

- the dropping of waste packages during handling;
- the dropping of equipment (including other waste packages, transport container lids etc) on to waste packages;
- more extreme facility mechanical failures, such as vault roof collapses, etc.

Such accidents could result in damage to waste packages, the release of their radioactive contents and radiation dose to both workers on-site and members of the public off-site. The regulatory control of radiation exposure as a result of operations on nuclear licensed sites are by way of criteria defined by the NII SAPs [29]. In the case of Design Basis Accidents (DBAs) the SAPs define Basic Safety Levels (BSLs) for on- and off-site dose consequences, on the basis of the expected frequency of the fault that would result in such an accident. The BSL doses are listed in Table 2.

RWMD have adopted a conservative approach to the treatment of DBAs by assuming that all such events will have a fault frequency of $>10^{-3}$ pa unless it can be shown otherwise. For most faults involving waste package impacts this will mean that the lowest of the BSLs from Table 2 will be adopted (i.e. Post-accident dose limit of 20mSv for workers and 1mSv for members of the public). However the opportunity exists for specific extreme faults to be assigned a lower frequency and corresponding higher BSLs should this prove necessary.

Table 2 Basic Safety Level Doses for DBAs

DBA Fault Frequency	BSL for on-site dose	BSL for off-site dose
>1x10^{-3} pa	20mSv	1mSv
Between 1x10^{-3} and 1x10^{-4} pa	200mSv	10mSv
<1x10^{-4} pa	500mSv	100mSv

As part of the Generic Operational Safety Assessment (GOSA) [30], a HazOp study has been carried out to identify the nature of DBAs that could occur during the operational period. This has identified a number of impact faults for which the highest frequency of occurrence in Table 2 has been assumed, and which are used to define waste package impact performance criteria. These include:

- an impact fault occurring in the inlet cell[17], equivalent to a drop from a height of 3m, and;
- a range of faults, including the dropping of a waste package during lifting or from the top of a stack of waste packages, equivalent to a drop from a height of approximately15m[18].

All such faults are assumed to involve the impact of a waste package on to an unyielding surface (i.e. the vault floor) or on to an unyielding ‘aggressive feature’ such as the corner stacking post of a 3 cubic metre Box waste package or a stillage[19] or, in the case of an accident in the inlet cell, on to a rigid feature of the transport container.

Other, more extreme faults have also been identified by the HazOp study, including the collapse of a vault roof or waste package handling crane on to waste packages, or the consequences of a ‘runaway’ transport package transporter in the drift access to the underground facilities. Such faults could result in more severe impacts than one equivalent to that resulting from a 15m drop and could affect many waste packages. It is assumed that sufficient safety features would be incorporated in the design of relevant systems to ensure that such events would have a lower frequency that 10^{-3} pa and that the higher values of BSL from Table 2 would be applied to their consequences. Historically it had been assumed that such faults would result in an impact to a waste package equivalent that resulting from a fall from 25m and this value was previously used as the basis for defining waste package impact performance. It is now believed that the requirement for the impact performance of waste packages to be ‘*low and predictable’ and to ‘exhibit progressive release behaviour with increasing impact severity……..within the anticipated range of impact conditions’,* backed up by computer modelling and/or impact testing of waste packages, will provide sufficient confidence regarding the performance of waste packages for the full range of anticipated impact faults.

As part of the GOSA [30], RWMD has developed a standardised methodology that allows the radiological consequences of the release of activity from a waste package to be determined [31]. The methodology is applied by way of the GOSA Toolkit, a code that

[17] The inlet cell being the location where waste packages are removed from transport containers, prior to transfer to the disposal vaults

[18] This being a conservative figure for the maximum height to which waste packages would be lifted in a GDF in which waste packages are stacked seven high.

[19] As used for the handling and stacking of 500 litre Drum waste packages.

calculates the dose consequences of the full range of potential accidents identified by the HazOp analysis. The Toolkit incorporates assumptions made regarding the availability and efficiency of protective equipment within a GDF and the ventilation system, the anticipated proximity, exposure times and breathing rates etc of on-site workers to radionuclides released during accidents and the exposure routes to members of the public following an off-site release.

To produce quantified release limits for waste packages the GOSA Toolkit has been used to determine the size of release of individual radionuclides that would result in on- and off-site doses equal to the lowest BSL doses in Table 2. Table 3 lists the results of this process for radionuclides with the highest inventories in UK ILW[20]. Values are given for each of the two impact scenarios identified above, these values being different due to the different levels of protection currently assumed for workers and members of the public for impact accidents occurring in the respective areas of a GDF.

Table 3 shows that, in all cases, the on-site dose is the bounding situation, in some cases by up to four orders of magnitude. This reflects the protection provided to off-site releases of activity by the ventilation system, dispersion of activity by winds following release and the assumptions regarding the manner by which members of the public would be exposed to any released activity.

In the case of accidents in a GDF the maximum size of particles considered to contribute to dose has been historically considered to be 100μm. Although these larger particles are not considered respirable, they are considered to be 'suspendable' in air, and thus capable of contributing to the dose to both workers and the public by external radiation (i.e. 'shine') from deposited activity. Particles larger than 100μm have been deemed too large to be dispersed following an impact accident, and are not considered to contribute to the post-accident dose. The definition of the size of particles that will contribute to dose following an impact accident has significance in the definition of waste package impact performance and RWMD acknowledge that both the 40μm and 100μm values may be over-conservative, especially when the conclusions of the ICRP study [28] are considered. However, in the absence of convincing arguments for a relaxation of this conservatism, these values are to be retained although RWMD will be conducting work in the future with a view to reducing them.

The values given in Table 3 are not direct specifications for waste package impact performance but provide guidance as to whether a proposed waste package would be capable of demonstrating compliance with relevant regulatory requirements following an impact accident. It should first be noted that the values are for individual radionuclides and that the dose resulting form the release of activity following an impact accident in a GDF would be the combination of contributions from all of the radionuclides present in the waste package and released during an accident. The GOSA Toolkit is used to perform an assessment of the dose consequences of impact accident for each proposed waste package type proposed by Site License Companies. Such an assessment takes into account the actual radionuclide inventory of the proposed waste packages together with the release fraction[21] (RF) for waste packages subjected to representative impact challenges. Using generic values for RF [i.e. 32] and expected maximum waste package inventories, the values from Table 3 can be used to judge the potential acceptability of a

[20] Equivalent data on the full range of radionuclides considered by the GOSA Toolkit can be found in Appendix B.

[21] RF is defined as the activity released in and accident as the fraction of the total activity within a waste package. The value of RF for a particular waste package type will depend on the waste container and waste conditioning process(es) used, together with the nature and form of the radionuclides present in the waste. Typically RF's lie in the range 10^{-3} to 10^{-5} for standard waste packages containing wastes conditioned using cementitious grouts.

packaging proposal in advance of a full LoC assessment which would include the use of the GOSA Toolkit to assess the impact accident performance of the proposed waste packages.

Table 3 Allowable activity releases following impact accidents

Nuclide	Waste package activity release resulting in BSL for faults of frequency >10^{-3} pa			
	With protection assumed for Inlet Cell (3m drop)		With protection assumed for Vaults (15m drop)	
	Off-Site (1mSv)	On-Site (20mSv)	Off-Site (1mSv)	On-Site (20mSv)
	TBq	TBq	TBq	TBq
Am-241	4.9E-03	1.5E-06	4.9E-01	1.5E-04
C-14	8.3E+02	1.0E-01	8.3E+04	1.0E+01
Co-58	3.2E+00	2.9E-02	3.2E+02	2.9E+00
Co-60	1.0E-01	2.0E-03	1.0E+01	2.0E-01
Cs-137	1.1E-01	8.6E-03	1.1E+01	8.6E-01
Fe-55	1.8E+00	6.3E-02	1.8E+02	6.3E+00
H-3	2.0E+05	1.4E+00	2.0E+07	1.4E+02
Ni-59	1.9E+03	7.0E-02	1.9E+05	7.0E+00
Ni-63	2.7E+01	2.9E-02	2.7E+03	2.9E+00
Pu-238	9.0E-03	1.3E-06	9.0E-01	1.3E-04
Pu-239	8.6E-03	1.2E-06	8.6E-01	1.2E-04
Pu-240	8.6E-03	1.2E-06	8.6E-01	1.2E-04
Pu-241	6.3E-01	6.8E-05	6.3E+01	6.8E-03
Sm-151	5.2E+01	1.6E-02	5.2E+03	1.6E+00
Sr-90	3.4E-02	3.9E-04	3.4E+00	3.9E-02
Y-91	2.2E+00	6.9E-03	2.2E+02	6.9E-01

4.13.4 Influence of Waste Package Design on Impact Performance

The impact performance of waste packages is strongly dependent on package design, and careful attention should be paid to this from an early stage in the development of a packaging proposal. In particular, the benefits provided by the waste container and the wasteform under impact conditions should both be considered, to ensure that these two components are seen as independent and complementary barriers against the release of contents.

A programme of analytical studies and physical test work has been undertaken to assess the impact performance of waste packages and their detailed design features, e.g. [33]. This work has generated substantial information, improved the understanding of waste package impact performance and aided in the development of design guidelines. Guidance has also produced on waste container design [34] which includes specific guidance on how container design can maximise waste package impact performance.

The following points outline some basic principles that should be considered in waste package design, but it is recommended that waste packagers consult the RWMD in order to gain full benefit from the available information.

- A wasteform that is too weak can obviously result in inadequate impact performance. Less obviously, in certain circumstances a wasteform that is too strong could also result in inadequate performance by over-stressing the waste container. The assessment work performed to date demonstrates that static compressive strengths in the range 4 to 40MPa will be satisfactory, provided that the design details and material properties of the waste container are also appropriate. Wasteforms and matrix strengths outside this range may be acceptable, but would require consideration on a case-by case basis.
- A 'battering ram' effect can occur if the box contents slide inside the box and strike the lid. For this effect to occur, a relatively rigid thick-walled box is needed (especially sidewalls) as well as a smooth internal bore. Under these circumstances the forces on the lid can be very large in a top edge impact, and may be sufficient to cause lid separation. It is beneficial to key the wasteform block to the box walls.
- Generic performance data, has shown the most damaging attitude is likely to be the lid edge impact where an impact is expected to impart sufficient energy to damage the lid/edge interface such that some of the contents could be released. It should be noted that where the potential exists for improvement in the performance of the package due to advancement of technology or increased knowledge base from experience, these should be exploited. In particular generic work has recently been undertaken to investigate the comparative performance of box lid designs with respect to lid retention during impact.
- In general, problems are likely to occur with stiff unyielding features, which may protrude above the box or lid surface, or may be present inside the box alongside soft yielding features. These could lead to rupture by punching, or to concentrations of shear strain. This effect is more likely when the contents are relatively strong (i.e. unconfined compressive strength >20MPa).
- Special consideration should be given to heterogeneous wasteforms, e.g. encapsulated hard wastes, because of the potential for waste items to penetrate the box wall under impact conditions. If the waste has sharp edges, the potential for piercing of the box skin in an impact accident should be guarded against when loading the waste into the box.
- The benefits of the use of an ‘annular grouted’ wasteform should be considered for challenging wastes.
- Under certain circumstances, internal features such as stiffeners, supports or internal secondary lid features could cause a box to rupture during an impact. To reduce this potential, these features should be kept as far away from the sidewalls of the waste container as possible. Features near or attached to the sidewalls should preferably be rounded and non-aggressive, not sharp, and should not create severe localised reductions in ductility that could be prejudicial to impact performance.
- An area that requires particular attention is the box closure or lid. Failure of this can lead to a major breach and loss of contents. In general, test evidence has shown that a welded lid is better than a bolted lid.
- However, welds located at or near to the top or bottom edge of the box are vulnerable to failure on impact. These welds should be located as far as possible from the box edges, particularly the base of the box where the weld should be at least 60mm from the edge. There is less scope for complying with this

requirement at the top of the box, where there may be a welded-on lid flange (see above), but every effort should still be made.

- All butt welds should be full penetration, and fillet welds should be continuous and as strong as the parent material.
- Ductile material behaviour during deformation is highly desirable. This should be a consideration in the selection of box material and in determining any requirement for stress relieving following manufacture.
- Whilst the specification for the side lifting variant of the 3 cubic metre Box no longer includes a requirement for radiussed corners, such a design feature is good practise where the impact performance of the waste package is concerned.
- Designs that involve sudden and large changes in the wall thickness, or include notches and other stress raisers, should be avoided. Such features can lead to stress concentration, and can promote brittle (rather than ductile) material behaviour, particularly at high strain rates.
- Account should be taken of any reduction in waste package impact performance capabilities as a result of corrosion of the waste container material, either internally or externally.

4.14 Fire Performance

The waste package should be designed such that in the event of a fire accident:

- ***releases of radionuclides and other hazardous materials are low and predictable, exhibit progressive release behaviour with increasing fire severity and do not exhibit significant cliff-edge performance characteristics within the anticipated range of impact conditions;***
- ***both of the barriers to radionuclide release from the waste package (i.e. the waste container and the wasteform) should play an effective role in minimising those releases.***

The release of radioactive contents from the waste package, as a result of credible fire accidents during transport and a GDF operational period, shall not result in the relevant regulatory radiation dose limits to workers or to members of the public being exceeded.

The effects of fire accidents can potentially affect waste packages at all stages of their long-term management, up to vault backfilling, in a similar manner to impact accidents as discussed in Section 4.13. Accordingly, waste packages must be capable of withstanding specified fire conditions without excessive loss of contents. For the purposes of this document, the stages at which the effects of a fire accident are considered are limited to transport and subsequent handling and emplacement in a GDF (although the latter will have many similarities to operations during storage by the waste packager prior to transport).

4.14.1 Fire Severity

In developing the criteria for the required fire performance of waste packages it is necessary to define appropriate fire accident conditions (i.e. temperature and duration). The thermal test specified for Type B transport packages by the IAEA Transport Regulations [5] requires such packages to be exposed for 30 minutes to a hydrocarbon fuel/air fire with an average temperature of 800°C, fully engulfing the package. This would result in waste packages being exposed to a thermal transient of significantly less severity. The potential for more challenging thermal transients, when unprotected waste packages could be directly exposed to fires during the operational period of a GDF and this has been considered to allow bounding conditions for such fires to be determined [35]. This work has concluded that, although a fire duration of 30 minutes encompasses 85% of road transport fire accidents, the complicating factors of restricted access and fire fighting capabilities in an underground disposal facility mean that a longer duration (i.e. 1 hour) should be adopted as the fire challenge for waste packages.

With regard to flame temperature, experience from transport accidents suggests that although 800°C is appropriate to transport accidents, it does not take into account the particular conditions that could pertain to an underground fire. Recent fires in similar circumstances, albeit with large fuel inventories, have yielded evidence of flame temperature transients in excess of 1000°C. Accordingly a more conservative temperature than would be required for transport fires alone has been adopted and 1000°C has been specified as the average flame temperature for the fire challenge.

4.14.2 Fires resulting from transport accidents

The approach to specifying allowable activity releases from waste packages following a transport fire accident is similar to that adopted in Section 4.13.2 for transport impact accidents, this results in the same value for the allowable activity release (i.e. $11A_2$). As in the case of the impact release criterion, this value is very conservative and significant relaxation is possible for waste packages generating gas at a rate less than the maximum specified in Section 4.8.1, **Error! Reference source not found.** illustrates the extent of this relaxation.

No particle size is specified in the criteria for fire releases as it is assumed that these releases occur in the form of vapours which are readily transmitted through a leaking transport container seal by thermally-driven air flows, and are then inhaled without restriction by exposed persons.

4.14.3 Fires resulting from accidents in a GDF

The approach to setting allowable waste package activity releases following a fire accident in a GDF once again reflects that adopted for impact accidents (Section 4.13.3). A fire accident is considered a DBA; and in accordance with NII SAPs [29], it should not result in on- or off-site doses in excess of the relevant BSLs. For the purposes of this guidance, the most restrictive BSLs are conservatively assumed, for accidents with a frequency of greater than 10^{-3} pa. The GOSA Toolkit has been used to determine the size of release of individual radionuclides that would result in on- and off-site doses of 20mSv and 1mSv respectively. Table 4 lists the results of this process for radionuclides with the highest inventories in UK ILW[22].

Table 4 shows that, unlike the equivalent values for impact accidents (Section 4.13.3) the on-site dose is not always the bounding situation, and that for gaseous radionuclides (i.e. H-3), or those that could form gases in a fire (i.e. C-14), the off-site case is more bounding. This is due to gaseous species not being removed as affectively as solid particles by the ventilation system HEPA filters.

As discussed above, releases from waste packages as a result of a fire accident are assumed, in the first instance, to be predominantly in the form of gases and vapours. Accordingly no maximum particle size is specified for activity releases following a fire accident and the values in are for total activity release.

The values given in Table 4 are not direct specifications for waste package fire performance but provide guidance as to whether a proposed waste package would be capable of demonstrating compliance with relevant regulatory requirements following a fire accident. It should first be noted that the values are for individual radionuclides and

[22] Equivalent data on the full range of radionuclides considered by the GOSA Toolkit can be found in Appendix B.

that the dose resulting form the release of activity following a fire accident would be the combination of contributions from all of the radionuclides present in the waste package and released during such an accident. The GOSA Toolkit is used to perform an assessment of the dose consequences of a fire accident for each waste package type proposed by Site License Companies. Such an assessment takes into account the actual radionuclide inventory of the proposed waste packages together with the RF for waste packages subjected to representative thermal challenges. Using generic values for RF [i.e. 32] and expected maximum waste package inventories, the values from Table 4 can be used to judge the potential acceptability of a packaging proposal in advance of a full LoC assessment, which would include the use of the GOSA Toolkit to assess the fire accident performance of the proposed waste packages.

Table 4 Allowable activity releases following fire accident

Nuclide	Waste package activity release resulting in BSL for fire faults with frequency $>10^{-3}$ pa	
	Off-Site (1mSv)	On-Site (20mSv)
	TBq	TBq
Am-241	2.4E+00	2.8E-03
C-14	3.6E+00	1.9E+02
Co-58	1.0E+03	5.4E+01
Co-60	3.3E+01	3.7E+00
Cs-137	3.5E+01	1.6E+01
Fe-55	5.6E+02	1.2E+02
H-3	8.6E+02	2.6E+03
Ni-59	1.1E+06	1.3E+02
Ni-63	8.6E+03	5.4E+02
Pu-238	3.8E+00	2.5E-03
Pu-239	3.6E+00	2.3E-03
Pu-240	3.6E+00	2.3E-03
Pu-241	2.4E+02	1.3E-01
Sm-151	3.0E+04	2.9E+01
Sr-90	1.1E+01	7.2E-01
Y-91	9.7E+02	1.3E+01

4.14.4 Influence of Waste Package Design on Thermal Performance

The thermal performance of waste packages is dependent on package design, and careful attention should be paid to this from an early stage in the development of a packaging proposal.

The exposure of a waste package to an external fire provides a driving force that may lead to a release of radioactive material in the form of volatile or gaseous species, contaminated steam or particles entrained in steam or gas. The extent of this hazard should be minimised, and made predictable through application of the principles of wasteform design outlined in Section 4.11. In particular, the presence of free liquids, voidage or excessive heterogeneity of the wasteform could have significant mechanical effects on a wasteform heated during a fire. These effects include uneven expansion and excessive cracking with the consequential loss of radionuclide containment.

If a wasteform could support combustion, that could have a significant effect on the release of activity during and following a fire accident. Modelling and experimental work indicates that the interior of a typical non-combustible wasteform does not suffer a significant temperature rise during an external fire, and this effect makes a major contribution to the limitation of releases; but combustion of the wasteform would lead to higher temperatures, particularly if it took place in the interior of the wasteform. That in turn would lead to physical and chemical changes in the wasteform, resulting in the release of a larger fraction of the radionuclide inventory. Therefore combustion of the wasteform is an unacceptable hazard; wasteforms should be designed neither to burn nor to support combustion.

A programme of analytical studies, experimental work and modelling work has been undertaken to assess the thermal performance of waste packages and the release of radionuclides under fire conditions [36, 37]. This work has generated substantial information, improved the understanding of waste package fire performance and aided in the development of design guidelines. Modelling data on the behaviour of a package in a fire is available to waste package designers, and waste packagers are recommended to consult the NDA in order to gain full benefit from the available information. Guidance on waste container design has also been produced [34] which includes specific guidance on how container design can maximise waste package thermal performance.

The following guidance indicates desirable or undesirable features, and to identifies those aspects that require particular attention:

- The analysis of the effects of a fire should continue for some time after the end of the fire, because the interior temperature of the package will initially continue to increase as temperatures equilibrate throughout the wasteform. A similar effect applies if the package is inside a reusable transport container: after the end of the fire, the thermal mass of the container will continue to heat up the package for some time.
- The emissivity of the flames can be taken to be 0.9. The package emissivity must be justified by analysis.
- Features that protrude above the box or lid surface could be subjected, in some circumstances, to locally higher temperatures or heat fluxes than the remainder of the box. Consideration should be given to the effect of any local hot-spots, both on the container and on the wasteform.
- Designs that involve sudden and large changes in wall thickness should be avoided since these could lead to localised heating of the wasteform, or to local thermally-induced stress concentrations which may not be desirable.
- Any modelling of the performance of the package and its wasteform under fire accident conditions should take account of any internal structures (e.g. internal stiffening of the container or internal furniture such as baskets) since the latter in particular may provide significant heat transfer paths to the interior of the wasteform. As noted above, any demonstration of acceptable performance as a function of time should be continued to the point where falling temperatures, or improving parameters are achieved throughout the package.
- The wasteform should normally be considered homogeneous, with the same release mechanisms applying throughout, and the releases depending only on the

maximum local temperatures reached. However, if the design or nature of the package is such that the distribution of activity is non-uniform, this also should be considered. In this context, the benefits of the use of an annular grouted wasteform should be considered for challenging wastes.

- For non-homogeneous wasteforms (e.g. metal components or compacted drums of soft waste, in-filled with grout) the possibilities of inward transfer paths for heat and outward transfer paths for gases and vapours, and associated radioactivity, must be considered.

4.15 Stackability

The waste package shall be capable of withstanding a stacking load due to a seven high stack of similar waste packages, each with a gross mass of 12,000kg. This shall be the equivalent of a compressive load of 72,000kg applied along the vertical axis of the waste package. Under these load conditions, the waste package should not exhibit any permanent deformation or abnormality that would render it incompatible with any of the requirements defined in WPS/315.

In a GDF vault 3 cubic metre Box waste packages will be stacked seven high and should be capable of being stacked in such a manner while still maintaining the ability to be handled safely.

This requires that the shape of 3 cubic metre Box waste packages remain in conformance with the dimensional envelope defined in Section 4.5, for the lifting feature to remain in accordance with Section 4.6, and for the package to be capable of being handled safely using the standard lifting grab[23]. Waste packages should be capable of satisfying these requirements after having been stacked for an extended period of time (i.e. up to 150 years, for the reasons outlined in Section 4.10.1).

The stackability requirements should also take into account a maximum stack column offset of 25mm in addition to a maximum achievable stack column lean when support pillars and floor slope are at the limit of their tolerances.

It is recognised that it may be necessary to position spacers between stacked waste packages to minimise interface corrosion problems, and to allow stable stacking and prevent blocking of waste package filters. Care should be taken, however, to ensure that the design of any spacer does not impose undue concentrated loads on the waste packages, which might cause damage or permanent deformation when they are stacked. For design purposes it may be assumed that only waste packages of similar design would be stacked together, or alternatively that a flat spacer is placed between them.

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[23] As defined in *Lifting Frame for 3 cubic metre Drum and side lifting variant of 3 cubic metre Box. Description and Design Guidelines,WPS/601*.

4.16 Identification

The waste package shall be marked with an unique ten character identifier as defined in Specification of Waste Package Identification System, WPS/410. The identifier shall be marked on the vertical faces of the four lifting features, 50mm from the top edge as shown in Figure 2. The characters shall be 6 - 10mm high and should be capable of being read during all active stages of the long-term management of waste packages.

The application of a unique identification marking on each waste package enables the identification and tracking of waste packages throughout all stages of their long-term management and permits assignment of the appropriate data record. Making the identification 'machine-readable' and the use of a format containing check digits allows the waste package to be identified remotely and its number verified by an automatic computer check. The use of OCR-A characters [38] also allows for checking by the operator, either by direct sight or through the use of automated reading equipment.

The identifier will consist of ten alpha-numeric characters, the form of which is specified in *Specification for Waste Package Identification System, WPS/410,* supported by *Waste Package Identification System: Explanatory Material and Guidance, WPS/860.*

The identifier is required to be a permanent feature of the waste package that, as a minimum, will be readable accurately by machine and by eye upon receipt of the waste package at a GDF, and remain readable by some means during at least the first 50 years of the operational period of a GDF. As a design basis, a maximum period of interim surface storage of 150 years prior to transport should be assumed, leading to a minimum identifier longevity of 200 years.

For automatic reading systems to operate effectively, it is important to establish standard locations for the identifiers. Four such positions are specified to provide redundancy and minimise the risk of a package becoming unidentifiable. The positions specified, on the vertical edge of the waste package lifting feature (Figure 2), have been selected partly because marking in these positions is unlikely to affect the corrosion performance and associated containment integrity of the waste package.

The recommended method of inscribing the identifier is to laser-etch the characters which, in the case of austenitic stainless steel packages, is expected to satisfy the above requirements.

In-house markings and additional labels may be applied by the waste packager if required for internal purposes, provided that they do not affect waste package performance. However, any additional identification, whether temporary or permanent, must not compromise the integrity requirements of the waste package. This should include a consideration of the materials used for such markings, guidance on which can be found in [39].

4.17 Physical Protection for Nuclear Security

The quantity of Nuclear Material contained within the waste package shall be such that the waste package can be transported subject to standards of physical protection no higher than Category III.

The Nuclear Industries Security Regulations (NISR) 2003 lay down the approvals required for the physical protection of 'Nuclear Material' (NM)[24] in transit between

[24] Defined by OCNS as plutonium, uranium, neptunium, americium and other irradiated materials

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.

It is the RWMD's intention that the NM content of all waste packages destined for emplacement in a GDF will be such that they will require standards of physical protection no higher than those defined for Category III material under the NISR. This intention forms part of the Security Plan.

The assessment of the suitability of waste packages containing NM for transport to Category III standards of physical protection will take account of the following:

- the estimated maximum mass of NM in a single waste package, including the reliability of the estimate;
- the quantity, and activity, of other (non-fissile) radioactive material present;
- the method of conditioning of the waste, including any Safeguards issues raised by the possibility of recovery;
- the potential impact on the physical protection categorisation of a GDF of the proposed waste packages.

Table 5 lists the mass limits applicable to a number of types of NM typically found in ILW that would allow them to be transported with Category III standards of physical protection. A full listing of all NM can be found in [40]. It should be noted that these limits apply to NM in any physical or chemical form and that less restrictive limits may apply to the same material in conditioned wasteforms packaged in a manner that makes their recovery difficult.

During the assessment of a packaging proposal, a physical protection assessment will be carried out to consider the nature and quantity of any NM intended for transport and disposal, in particular, its attractiveness for theft and its dispersability from an act of sabotage. The assessment will conclude with a statement regarding compliance with the current GDF Security Plan. Any issues identified in the assessment that do not comply with the provisions and conditions of the Security Plan will be referred to the OCNS for information, and, if necessary, for direction.

Table 5 Limits on quantities of NM for waste packages transported with Category III levels of physical protection

Type(s) of Material	Maximum quantity categorised as Category III
Pu, U-233	0.5 kg
Enriched uranium containing >20% U-235	1.0 kg
Enriched uranium containing <20% but >10% U-235	10 kg
Enriched uranium containing <10% U-235	Any quantity

4.18 Nuclear Materials Safeguards

The Safeguards status of any fissile or source materials (i.e. isotopes of uranium, plutonium and thorium) contained within a waste package shall be ascertained.

To prevent the potential for the diversion of civil nuclear materials to military use, 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 '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 their long-term management and a Safeguards assessment will be required. Such an assessment will review the proposed management processes for the packaged waste and consider whether they are likely to be adequate to meet the requirements of the Safeguards authorities.

In order that implications of accepting waste packages that contains safeguarded materials can be fully assessed and, in particular, the likely impact on GDF operations, the waste packager will be required to provide sufficient information on the quantity, nature and status of all Safeguarded material that will be incorporated into proposed waste packages.

5 QUALITY MANAGEMENT

Quality management arrangements shall be applied to all aspects of the packaging of radioactive wastes that affect product quality. These arrangements shall be agreed with RWMD prior to the start of the activities to which they relate.

All activities relevant to licensing of a GDF will be conducted in accordance with appropriate quality management arrangements. The objective in establishing and operating a quality management system is to provide an integral framework of procedures which will ensure that the work is adequately controlled, documented and recorded. It is the responsibility of the waste packager to develop, operate and maintain appropriate quality management arrangements which meet all RWMD requirements. These arrangements will be the subject of a separate approval by RWMD, as specified in *Waste Package Quality Management Specification, WPS/200*. Guidance on the quality management requirements can be found in *Waste Package Quality Management Specification. Guidance Material, WPS/210.*

6 WASTE PACKAGE DATA AND INFORMATION RECORDING

Information shall be recorded on all relevant details of the manufacture of each waste package. That information shall be sufficient to enable assessment of the waste package characteristics and performance against the requirements of all stages of their long-term management.

Waste Package Data and Information Recording Specification, WPS/400, describes the information to be supplied by waste packagers on the nature and contents of each waste package. Supporting guidance can be found in *Waste Package Data and Information Recording Specification: Explanatory Material and Guidance, WPS/850*. Most of this data will of necessity have to be recorded at the time the waste is packaged. It will eventually be transferred and stored on a GDF database maintained by RWMD, where it will be enhanced with additional information including the location of each waste package within a GDF. The data will comprise a permanent record of the waste that has been disposed of in a GDF and will be used to ensure that operations are carried out within the limits of the relevant authorisations.

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APPENDIX A WASTE PACKAGE GAS GENERATION LIMITS

Table A.1 Generation Limits for Flammable Gases

Gas	Lower Flammability Limit	Gas Generation Limit above which nitrogen purging of transport container is required (litres/day)	
	(%)	Maximum Displacement Volume	Minimum Displacement Volume
Ammonia	16	1.89	2.65
Benzene	1	0.11	0.15
Butylamine	2	0.22	0.31
Carbon disulphide	1	0.11	0.15
Cyclohexane	1	0.11	0.15
Furfural	2	0.22	0.31
Heptane	1	0.11	0.15
Hydrogen	4	0.43	0.61
Hydrogen sulphide	4.3	0.46	0.64
Methane	5.6	0.61	0.85
Styrene	1	0.11	0.15
Toluene	1	0.11	0.15
Trichloroethylene	13	1.62	2.27
Oxygen generation limit when nitrogen purging is applied		0.24	0.34

Table A.2 Generation Limits for Toxic Gases

Gas	Gas Generation Limit (litres/day)	
	Maximum Displacement Volume	Minimum Displacement Volume
Arsine	27.6	38.6
Hydrogen selenide	27.6	38.6
Nickel carbonyl	27.6	38.6
Ozone	53.7	75.2
Phosphorous oxychloride	53.7	75.2
Stibine	53.7	75.2

Table A.3 Generation Limits for Radioactive Gases

Nuclide	Allowable Package Activity Release (TBq/day)	
Half-life <1 year	**Maximum Displacement Volume**	**Minimum Displacement Volume**
Ar-41	1.3×10^{-4}	1.8×10^{-4}
Kr-85m	1.3×10^{-3}	1.8×10^{-3}
Kr-87	8.6×10^{-5}	1.2×10^{-4}
Rn-222	1.7×10^{-6}	2.4×10^{-6}
Xe-122	1.7×10^{-4}	2.4×10^{-4}
Xe-123	3.0×10^{-4}	4.2×10^{-4}
Xe-127	8.6×10^{-4}	1.2×10^{-3}
Xe-131m	1.7×10^{-2}	2.4×10^{-2}
Xe-133	4.3×10^{-3}	6.0×10^{-3}
Xe-135	8.6×10^{-4}	1.2×10^{-3}
Half-life >1 year		
Ar-39	8.6×10^{-3}	1.2×10^{-2}
C-14	1.3×10^{-3}	1.8×10^{-3}
H-3	4.2×10^{-3}	5.8×10^{-3}
Kr-81	1.7×10^{-2}	2.4×10^{-2}
Kr-85	4.3×10^{-3}	6.0×10^{-3}

APPENDIX B ACTIVITY RELEASE LIMITS FOR DBAs

	Activity Release resulting in BSL for Fault Frequency $>10^{-3}$ pa			
	Impact Accidents		Fire Accidents	
	Off-Site (1mSv)	On-Site (20mSv)	Off-Site (1mSv)	On-Site (20mSv)
Ag-108m	5.5E+00	1.6E-01	1.7E+01	3.1E+00
Am-241	4.9E-01	1.5E-04	2.4E+00	2.8E-03
Am-242m	5.5E-01	1.6E-04	2.6E+00	3.1E-03
Am-243	4.9E-01	1.5E-04	2.4E+00	2.8E-03
Be-10	1.4E+02	1.8E-01	4.9E+02	3.4E+00
C-14	8.3E+04	9.9E+00	3.6E+00	1.9E+02
Ca-41	3.4E+03	3.0E+01	1.1E+04	5.7E+02
Ca-45	2.8E+02	2.1E+00	9.0E+02	4.0E+01
Ce-144	8.6E+01	1.2E-01	2.8E+02	2.2E+00
Cl-36	2.0E+01	8.4E-01	6.3E-04	1.6E+01
Cm-242	4.7E+00	1.2E-03	2.7E+01	2.3E-02
Cm-243	6.2E-01	2.0E-04	3.6E+00	3.7E-03
Cm-244	6.0E-01	2.3E-04	2.9E+00	4.3E-03
Cm-245	4.4E-01	1.4E-04	2.5E+00	2.7E-03
Cm-246	4.4E-01	1.4E-04	2.5E+00	2.7E-03
Co-58	3.2E+02	2.9E+00	1.0E+03	5.4E+01
Co-60	1.0E+01	2.0E-01	3.3E+01	3.7E+00
Cr-51	1.9E+04	1.6E+02	6.0E+04	3.0E+03
Cs-134	1.2E+01	6.0E-01	3.7E+01	1.1E+01
Cs-135	7.9E+00	5.8E+00	3.4E+01	1.1E+02
Cs-137	1.1E+01	8.6E-01	3.5E+01	1.6E+01
Eu-152	2.5E+01	1.5E-01	8.1E+01	2.8E+00
Eu-154	2.9E+01	1.2E-01	9.3E+01	2.2E+00
Eu-155	4.3E+02	8.8E-01	2.0E+03	1.7E+01
Fe-55	1.8E+02	6.3E+00	5.6E+02	1.2E+02
H-3	2.0E+07	1.4E+02	8.6E+02	2.6E+03
I-129	7.9E-01	6.0E-02	3.4E-05	1.1E+00

	Activity Release resulting in BSL for Fault Frequency >10^{-3} pa			
	Impact Accidents		Fire Accidents	
	Off-Site (1mSv)	On-Site (20mSv)	Off-Site (1mSv)	On-Site (20mSv)
I-131	7.2E+01	2.9E-01	3.1E-03	5.4E+00
Mn-54	8.7E+01	3.9E+00	2.8E+02	7.2E+01
Mo-93	6.8E+00	2.6E+00	2.1E+01	4.9E+01
Nb-93m	3.8E+03	3.6E+00	1.3E+04	6.8E+01
Nb-94	6.7E+00	1.3E-01	2.1E+01	2.4E+00
Nb-95	5.8E+04	3.6E+00	2.2E+05	6.8E+01
Ni-59	1.9E+05	7.0E+00	1.1E+06	1.3E+02
Ni-63	2.7E+03	2.9E+00	8.7E+03	5.4E+01
Np-237	1.1E+00	2.7E-04	6.3E+00	5.1E-03
Pa-231	8.6E-01	4.4E-05	3.6E+00	8.3E-04
Pb-210	2.5E-01	5.2E-03	7.9E-01	9.8E-02
Pd-107	2.6E+01	1.0E+01	1.1E+02	2.0E+02
Pm-147	5.2E+03	1.2E+00	3.0E+04	2.3E+01
Po-210	2.5E-01	1.9E-03	7.9E-01	3.6E-02
Pr-144	0.0E+00	1.9E+02	0.0E+00	3.6E+03
Pu-238	9.0E-01	1.3E-04	3.8E+00	2.5E-03
Pu-239	8.6E-01	1.2E-04	3.6E+00	2.3E-03
Pu-240	8.6E-01	1.2E-04	3.6E+00	2.3E-03
Pu-241	6.3E+01	6.8E-03	2.4E+02	1.3E+00
Pu-242	9.2E-01	1.3E-04	3.8E+00	2.5E-03
Ra-226	1.3E+00	1.8E-03	6.5E+00	3.4E-02
Rh-106m	0.0E+00	3.0E+01	0.0E+00	5.7E+02
Ru-103	2.0E+03	2.1E+00	7.4E+03	3.9E+01
Ru-106	5.8E+01	9.3E-02	2.0E+02	1.7E+00
S-35	1.6E+01	4.4E+00	6.7E-04	8.3E+01
Se-79	7.9E+00	1.9E+00	3.4E-04	3.5E+01
Sm-151	5.2E+03	1.6E+00	3.0E+04	2.9E+01
Sn-121m	1.1E+02	1.4E+00	3.6E+02	2.6E+01
Sn-126	1.0E+01	2.1E-01	3.3E+01	4.0E+00
Sr-89	2.8E+02	7.7E-01	9.0E+02	1.4E+01

	Activity Release resulting in BSL for Fault Frequency >10^{-3} pa			
	Impact Accidents		Fire Accidents	
	Off-Site (1mSv)	On-Site (20mSv)	Off-Site (1mSv)	On-Site (20mSv)
Sr-90	3.4E+00	3.9E-02	1.1E+01	7.2E-01
Ta-182	2.2E+02	5.9E-01	9.7E+02	1.1E+01
Tc-99	2.0E+01	5.9E-01	6.3E+01	1.1E+01
Th-229	8.6E-01	5.8E-05	3.6E+00	1.1E-03
Th-230	8.6E-01	1.4E-04	3.6E+00	2.7E-03
Th-232	8.6E-01	1.4E-04	3.6E+00	2.6E-03
U-233	2.8E+00	6.6E-04	1.6E+01	1.2E-02
U-234	2.8E+00	6.8E-04	1.6E+01	1.3E-02
U-235	2.9E+00	7.5E-04	1.6E+01	1.4E-02
U-236	2.8E+00	7.3E-04	1.6E+01	1.4E-02
U-238	2.3E+00	7.9E-04	1.1E+01	1.5E-02
Y-90	5.8E+04	3.4E+00	3.4E+05	6.3E+01
Y-91	2.2E+02	6.9E-01	9.7E+02	1.3E+01
Zn-65	6.7E+00	2.0E+00	2.1E+01	3.7E+01
Zr-93	2.5E+03	2.0E-01	1.5E+04	3.7E+00
Zr-95	4.3E+02	1.0E+00	9.7E+02	2.0E+01

APPENDIX C ABBREVIATIONS AND ACRONYMS

ACT	Accident Conditions of Transport
Bq, GBq, TBq	becquerel, gigabecquerel (10^9 Bq), terabecquerel (10^{12} Bq)
DBA	Design Basis Accident
GDF	Geological Disposal Facility
GDSS	Generic Disposal System Specification
GOSA	Generic Operational Safety Assessment
GRD	Generic Repository Design
GWPS	Generic Waste Package Specification
HEPA	High Efficiency Particulate Air (filter)
HEU	High Enriched Uranium
HSE	Health and Safety Executive
IAEA	International Atomic Energy Agency
ILW	Intermediate Level Waste
kg, g	kilogram, gram
LLW	Low Level Waste
LoC	Letter of Compliance
m, cm, mm, μm	metre, centimetre, millimetre, micrometre or micron (10^{-6}m)
MNOP	Maximum Normal Operating Pressure
NCT	Normal Conditions of Transport
NRVB	Nirex Reference Vault Backfill
OCNS	Office for Civil Nuclear Security
PGRC	Phased Geological Repository Concept
SAP	NII Safety Assessment Principles
SFM	Safe Fissile Mass
Sv, mSv	sievert, millisievert
SWTC	Standard Waste Transport Container
t	metric tonne (1000 kg)
W, kW	watt, kilowatt
WPSGD	Waste Package Specification and Guidance Documentation

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