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# DELIVERABLE REPORT

Thermal treatment for radioactive waste minimisation and hazard reduction

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The report has been reviewed by THERAMIN Consortium members and approved by THERAMIN WP4 leader. The THERAMIN coordinator has reviewed and approved the report for submission.

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## THERAMIN Project Partners

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| Andra | Agence nationale pour la gestion des déchets radioactifs – France |
| ORA | Orano – France |
| CEA | Commissariat à l'énergie atomique et aux énergies alternatives – France |
| GSL | Galson Sciences Limited – UK |
| FZJ | Forschungszentrum Juelich GmbH – Germany |
| LEI | Lithuanian Energy Institute – Lithuania |
| NNL | National Nuclear Laboratory – UK |
| ONDRAF/NIRAS | Organisme National des Déchets RAdioactifs et des matières Fissiles enrichies – Belgium |
| SCK•CEN | The Belgian Nuclear Research Centre – Belgium |
| USFD | University of Sheffield – UK |
| VTT | Teknologian Tutkimuskeskus VTT Oy (VTT Technical Research Centre of Finland Ltd) |
| VUJE | VUJE a.s. – Slovakia |

## THERAMIN End User Group

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|---|---|
| Andra | Agence nationale pour la gestion des déchets radioactifs – France |
| CEA | Commissariat à l'énergie atomique et aux énergies alternatives – France |
| EDF | Electricité de France – France |
| Fortum | Fortum Oyj – Finland |
| IGD-TP | Implementing Geological Disposal of Radioactive Waste Technology Platform |
| Nagra | Nationale Genossenschaft für die Lagerung Radioaktiver Abfälle – Switzerland |
| ONDRAF/NIRAS | Organisme National des Déchets RAdioactifs et des matières Fissiles enrichies – Belgium |
| RWM | Radioactive Waste Management Ltd – UK |
| Sellafield | Sellafied Ltd – UK |
| TVO | Teollisuuden Voima Oyj – Finland |

## List of acronyms

| | |
|---|---|
| ASR | Alkali-Silica-Reaction |
| ATA | Alphatoxic waste |
| DEF | Delayed Ettringite Formation |
| DGR | Deep Geological Repository |
| EC | European Commission |
| ENSI | Eidgenössisches Nuklearsicherheitsinspektorat / Swiss Federal Nuclear Safety Inspectorate |
| EU | European Union |
| FCC | Fibre-reinforced Concrete Containers |
| FSAR | Final Safety Analysis Report |
| GDF | Geological Disposal Facility |
| GR | Geological Repository |
| HAW | Higher Activity Wastes |
| HLW | High Level Waste |
| HM | Heavy Metal |
| IAEA | International Atomic Energy Agency |
| ILW | Intermediate-Level Waste |
| LILW | Low- and Intermediate Level Waste; |
| LL | Long-Lived |
| LLW | Low-Level Waste |
| NPP | Nuclear Power Plant |
| NSR | Near Surface Repository |
| PP | Primary Packaging |
| QA | Quality Assurance |
| RN | Radionuclides |
| RWP | Radioactive Waste Package |

| | |
|---|---|
| SL | Short-Lived |
| SR | Slovak Republic |
| SST | State of Science and Technology |
| TD | Technical Design |
| TRL | Technology Readiness Level |
| VLLW | Very Low-Level Waste |
| VSL | Very Short-Lived |
| WAC | Waste Acceptance Criteria |
| WENRA | Western European Nuclear Regulators Association |
| WMO | Waste Management Organisation |
| WP | Work Package |

# 1 Introduction

## 1.1 Background

The **Th**ermal treatment for **ra**dioactive waste **min**imisation and hazard reduction (THERAMIN) project is a European Commission (EC) programme of work jointly funded by the Horizon 2020 Euratom research and innovation programme and European nuclear waste management organisations (WMOs). The THERAMIN project is running in the period June 2017 – May 2020. Twelve European WMOs and research and consultancy institutions from seven European countries are participating in THERAMIN.

The overall objective of THERAMIN is to provide improved safe long-term storage and disposal of intermediate-level wastes (ILW) and low-level wastes (LLW) suitable for thermal processing. The work programme provides a vehicle for coordinated EU-wide research and technology demonstration designed to provide improved understanding and optimisation of the application of thermal treatment in radioactive waste management programmes across Europe, and will move technologies higher up the Technology Readiness Level (TRL) scale. The THERAMIN project is being carried out in five work packages (WPs). WP1 includes project management and coordination and is being led by VTT. WP2 evaluates the potential for thermal treatment of particular waste streams across Europe; this WP is led by GSL. In WP3, the application of selected thermal treatment technologies to radioactive waste management is demonstrated and evaluated; this WP is led by NNL. In WP4, the disposability of the thermally treated radioactive waste products is assessed; this WP is led by Andra. WP5 concerns synthesis of the project outcomes and their dissemination to other interested organisations; this WP is also led by GSL.

WP4 aims to carry out an evaluation of the disposability of thermally treated waste products and of the manageability of the resulting secondary waste, depending on the waste stream/treatment process combinations and depending on the disposal concepts in each participating country. WP4 is divided into three tasks:

- Task 4.1: Identification and review of criteria and requirements for the disposability of thermally treated waste products

  Under this task, Waste Acceptance Criteria (WAC) of interest and requirements in terms of behaviour and performance of waste products will be identified. Moreover, required characterisation tests will be determined.

- Task 4.2: Study of thermally treated waste products and secondary waste

  Under this task, characterisation tests will be carried out on thermally treated waste products and secondary waste. Some relevant existing data will be shared.

- Task 4.3: Downstream / Safety Case implications

  This task is focused on the disposability of thermally treated waste based on the identified criteria and the experimental data from the 2 previous tasks.

This deliverable is the first of Task 4.1. It covers all the work which has been carried out in this task.

## 1.2 Objectives of this Report

The objectives of this report are to compile data concerning WAC relevant to thermally treated waste products in the eight THERAMIN participating countries. This report presents the context and applicable WAC for management and disposal of radioactive waste in each country and, on this basis, derives generic WAC for thermally treated wasteforms that can be applied more generally. Belgium, Finland, France, Germany, Lithuania, Slovakia, Swiss and the United Kingdom contributed to this report. The generic WAC identified in this report are intended to assist in identifying the requirements in terms of characterisatio. They will also be a useful output of the project for the development of national WAC for thermally treated wastes.

## 1.3 Scope of this Report

This report gathers the data on WAC collected from each THERAMIN participating country. The WAC identified in this report are the basic data which are required to develop the other tasks of the WP. Depending on the country, WAC could be general and/or qualitative rules, or quantified and/or precise criteria. They could be developed especially for the disposal of thermally treated waste, or be derived from WAC which apply to other kind of waste. As a consequence, the WAC presented in this report should be used carefully, with the knowledge of the context of each country. This is considered further in Section 4, which discusses the development of generic disposability criteria for thermally treated wastes that are applicable in any context. This is especially enabled by the broad range of contributing countries. Then, characterisation requirements have been identified and developed on this basis.

## 1.4 Report Structure

The remainder of this report is set out as follows:

- Section 2 provides information on the context in each countries, such as the classification of radioactive waste, the existing or in-development disposal facilities, the main dates, the known issues, etc.
- Section 3 compiles criteria for the disposability of thermally treated waste products, for each participating country.
- Section 4 describes the development of generic WAC for thermally treated waste.
- Section 5 identifies the requirements in terms of characterisation.
- Section 6 sets out the conclusions of this report.
- Section 7 lists the references used in this report.

# 2 National Radioactive Waste Management Strategies

## 2.1 Belgium

In Belgium, ONDRAF/NIRAS implements a four-level hierarchical classification system for conditioned radioactive waste (see Figure 1). The most general classification is the group, which refers to the final disposal concept of the waste that it contains. As such, two groups exist, an open and a closed group. Next, the groups are further subdivided into three categories: category A, B and C. At this second level, distinctions are made based on the radiological activities of the waste and their possible heat generation. Category A is the equivalent of short-lived low and intermediate level waste, category B of long-lived low and intermediate level waste and category C corresponds to high level waste. At the third level, conditioned waste that has similar packaging, storage and potential disposal methods are assigned to twenty classes that are finally allocated to about 60 families. The classification into families is the last level in the classification hierarchy. Each waste family comprises waste packages with individual characteristics for which the deviations from the family average do not produce any significant impact during any step subsequent to its production (ONDRAF/NIRAS, 2001, 2008, 2013).

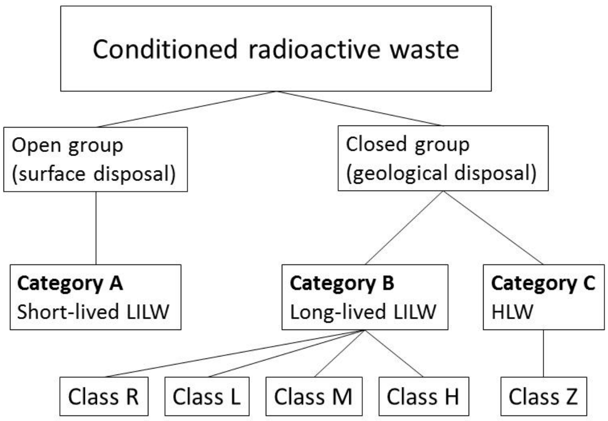


*Figure 1. Classification of conditioned radioactive waste in Belgium. LILW: low- and intermediate level waste; HLW: high level waste*

The national policy for waste that is categorised as category A is directed towards surface disposal. This solution is designed to guarantee long term safety in a passive way. In this disposal concept, waste is placed in concrete caissons then filled with mortar to obtain a monolith. These monoliths are placed in large structures of concrete, the disposal-modules. Once the modules are filled, they are closed by a concrete plate and covered by a layer of soil, which ensures the functions of inclusion, protection and sealing (NIRAS/ONDRAF, 2008).

For this scenario the creation and operation license request is submitted; it is foreseen to obtain the licence in 2019. The first monolith could then be disposed in 2023. It is expected to close the disposal facility 100 years after obtaining the licence.

Currently there is no national policy regarding the long-term management of category B waste and category C waste in Belgium. Therefore, the reference management route for the category

B and C waste is based on the conclusion of the Waste Plan (ONDRAF/NIRAS, 2011) approved by the board of directors of ONDRAF/NIRAS. The reference management option, considered for Research, development and Demonstration (RD&D) is **geological disposal in poorly indurated clays**. A proposition of national policy for the long term management of the category B and C waste was also proposed by ONDRAF/NIRAS based on this reference. This reference management option implies either the Boom Clay, either the Ypresian clays as reference host rock. Based on more than 30 years of RD&D, ONDRAF/NIRAS considers that a GDF in poorly indurated clay could be safe and feasible from a depth of 200 m onward. The proposed reference management route for the category B and C waste is thus a **geological disposal in poorly indurated clays at a depth between 200 m and 600 m**. The disposal packages for category B waste are called monoliths B. They are composed of a concrete caisson in which the packages of B-wastes are inserted. The disposal packages for category C waste are called supercontainers. In this design, the primary waste packages are surrounded by a carbon steel overpack, a filler and a buffer made of concrete and, if needed, a stainless steel envelope. These monoliths and supercontainers are in turn placed in concrete-lined disposal galleries that are excavated at mid-depth in the host rock. The remaining voids between the containers will be backfilled (NIRAS/ONDRAF, 2008).

In the currently considered reference scenario, it is foreseen to obtain the creation and operation license in 2050. Obtaining a license will however, strongly depend on the establishment of the social acceptance of the solution and the duration of the research necessary for the licensing request. The start of the disposal of B waste is planned to take place 20 years after the acquisition of the license. The disposal of the C-waste is foreseen to start in 2110. In this planning, the closure activities would start in 2130. It should be noted that this planning strongly depends on the political decisions that have to be taken in order to make this management route possible.

## 2.2 Finland

### 2.2.1 Classification of Radioactive Wastes

In Finland, radioactive waste is categorised into HLW, ILW, and LLW:

- HLW, such as spent nuclear fuel, has activity concentrations higher than 10 GBq/kg.
- ILW, such as ion-exchange resins used for cleaning the primary circuit in a reactor, has activity concentrations between 1 MBq/kg and 10 GBq/kg.
- LLW, such as maintenance waste, has activity concentrations below 1 MBq/kg.

Additionally, for disposal purposes, radioactive waste is often categorised into short- and long-lived waste. For the short-lived waste, the half-life of the predominant radioactive substance is not more than about 30 years and its activity can be reduced to a safe level within a few hundred years. Low and intermediate-level radioactive waste originating from NPP operations is usually within this category, whereas long-lived waste, such as spent fuel, contains significant concentrations of radioactive substances with half-lives of more than 30 years.

### 2.2.2 Radioactive Waste Management Plans

The producers of nuclear energy (currently Fortum and TVO, and in the future also Fennovoima) are responsible for the safety of nuclear waste and nuclear materials. To fund nuclear waste management, the nuclear energy producers have a statutory obligation to contribute financially to the nuclear waste management fund.

At both Loviisa and Olkiluoto NPP sites, there are fresh and spent fuel storage facilities, and facilities for storage, treatment and disposal of low and intermediate level radioactive wastes. In order to take care of spent fuel disposal, a joint company Posiva OY was established in 1995 by Fortum and TVO. Research, development and planning work as well as construction of a disposal facility are in progress. The ONKALO disposal facility in Olkiluoto is envisaged to be operational in the early 2020s.

## 2.3 France

### 2.3.1 Classification of radioactive waste

Radioactive waste classification in France is based on two parameters, which are important when determining the appropriate management method (Andra, 2015).

The first distinction is made between the following waste activity levels:

- very low-level waste (VLLW);
- low-level waste (LLW);
- intermediate-level waste (ILW);
- high-level waste (HLW).

The second distinction is made between the following half-life:

- very short-lived (VSL) waste, which contains radionuclides with a half-life of less than 100 days;
- short-lived (SL) waste, whose radioactivity comes mainly from radionuclides with a half-life of less than or equal to 31 years;
- long-lived (LL) waste, which contains a significant quantity of radionuclides with a half-life of more than 31 years

Then, there are five categories of waste that require, or will require, special management.

1. High-Level waste (HLW): the activity level of HLW is several billion Becquerel (Bq) per gram. This type of waste comes for the most part from the nuclear power industry and related research, and, to a lesser extent, from the defence industry. It mainly arises from reprocessing spent fuel. Most of this waste is vitrified in stainless steel containers. Because of its high radioactivity, this type of waste gives off heat.
2. Intermediate-level long-lived waste (ILW-LL): this waste mainly comes from spent fuel reprocessing and activities involved in the maintenance and operation of processing plants. It comprises structural waste from fuel assemblies (end caps and cladding hulls), technological waste (used tools, equipment, etc.) and waste resulting from the treatment of effluents, such as certain types of sludge. Other types of ILW-LL originate from components that have been activated while exposed to neutron flux in a reactor. The activity of this waste ranges between one million and one billion Becquerel per gram, i.e. lower than that of HLW by a factor of 10 to 100.

3. Low-level long-lived waste (LLW-LL): this category consists mainly in two types of waste:
   - Radium-bearing waste mostly arises from non-nuclear industrial activities such as some types of research and rare earth minerals processing. Other radium-bearing waste comes from the clean-up of legacy sites contaminated with radium. The level of radioactivity of this waste is usually between a few tens and a few thousands of Becquerel per gram.
   - Graphite waste comes from operation and dismantling of the first nuclear power plants (gas-cooled graphite-moderated reactors, GCRs) and certain experimental reactors that have been shut down. This type of waste has a level of radioactivity between 10,000 and 100,000 Becquerel per gram and contains mainly long-lived beta-emitting radionuclides.
   - This LLW-LL category also comprises other types of waste, such as spent sealed sources, certain legacy bitumen packages and uranium conversion treatment residues.
4. Low-and intermediate-level short-lived waste (LILW-SL): this waste related to maintenance (clothing, tools, filters…) and operation (liquid effluent treatment or gaseous effluent filtering) of nuclear power plants, fuel cycle facilities and research centre. It can also come from dismantling operations on these facilities. The level of radioactivity of this waste is usually between a few hundred and one million Becquerel per gram. LILW-SL waste is disposed of in a surface facility. The category of LILW-SL includes low-and intermediate-level-short-lived waste containing a significant quantity of tritium.
5. Very-low-level waste (VLLW): this waste mainly comes from the operation, maintenance and dismantling of nuclear power plants, fuel cycle facilities and research centres. It also originates from conventional industries using naturally occurring radioactive materials. The level of radioactivity of this waste is generally less than 100 Becquerel per gram.

Sometimes, this classification is not applicable to certain types of waste, either because they cannot be handled using existing management solutions in view of some of their characteristics, and especially their chemical characteristics, or because treatment or conditioning processes are not available or particularly complex to develop, given the sometimes small quantities involved. Examples include some oils and organic liquids that cannot be incinerated, or waste containing mercury.

The Very short lived waste (VSLW) are managed by storing it until its radioactivity decays. Thus, it isn't considered as a category.

The figure below presents the breakdown of radioactive waste volumes by category at the end of 2013.

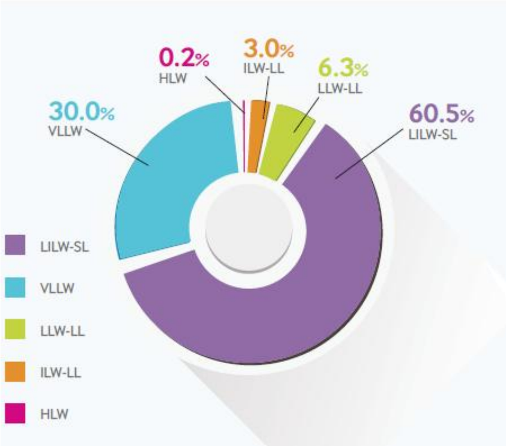


*Figure 2. Radioactive waste volumes by category at the end of 2013 (Andra,2015)*

### 2.3.2 Radioactive waste management plans

The management plan of radioactive waste depends of the category of the waste (see table below).

For both HLW and ILW-LL, deep geological disposal is considered as the reference solution for these wastes. Andra is conducting studies and research to select a site and design a deep reversible disposal facility, called Cigéo, at a depth of 500 meters in a clay formation, to accommodate this waste. This geological disposal facility is due to be commissioned in 2025, subject to licensing. While awaiting the creation of Cigéo, HLW and ILW-LL waste are stored mainly on the sites where the packages are produced.

LLW-LL is currently stored where it is generated or, for certain types of waste not from nuclear power production, at Cires (radioactive waste management, interim storage and disposal facility in the Aube district, Eastern France), pending a management solution. Andra has been tasked with conducting research and studies to develop disposal solutions for this type of waste. Near-surface disposal of LLW-LL waste is currently being studied.

LILW-SL waste is disposed of in a surface facility and monitored during the time taken for its radioactivity to decay to levels with negligible impact. On the Andra disposal sites, it is generally considered that this level is reached after 300 years. These sites will therefore be monitored for at least 300 years. There are two dedicated sites in France for the disposal of LILW-SL: the CSM and CSA waste disposal facilities. No waste has been taken to the CSM disposal facility since 1994; it is currently in the monitoring phase. The CSA facility has been in operation since 1992. Prior to disposal, most of the low-level waste undergoes treatment by compaction to reduce volume. It is then placed in metal or concrete containers.

Since 2003, VLLW has been disposed of at Cires (radioactive waste management, interim storage and disposal facility), operated by Andra in the Aube district (Eastern France). Prior to disposal, VLLW is packaged according to type, either in big bags for ease of handling, or in metal containers. Certain waste items may be subject to specific treatment:

- Compaction for plastic and metallic waste, to reduce volume;
- Solidification then stabilisation for liquid waste (contaminated water, sludge, etc.).

Some waste, mainly hospital waste, contains very-short-lived radionuclides (with a half-life of less than 100 days), which are used for diagnostic or therapeutic purposes. This waste is stored on-site until its radioactivity has decayed, which takes from a few days to a few months. It is then disposed of using conventional methods.

The table below presents a synthesis of the French classification of radioactive waste and associated management solutions.

| | Very-short-lived waste containing radionuclides with a half-life < 100 days | Short-lived waste, whose radioactivity comes mainly from radionuclides with a half-life ≤ 31 years | Long-lived waste, whose radioactivity comes mainly from radionuclides with a half-life > 31 years |
|---|---|---|---|
| Very low-level waste (VLLW) | Management by radioactive decay VSLW | Surface disposal (Cires waste collection, storage and disposal facility) VLLW | |
| Low-level waste (LLW) | | Near Surface Disposal (CSA waste disposal facility) LILW-SL | Near-surface disposal (research ongoing under Section 3 of the Act of 28 June 2006, codified) LLW-LL |
| Intermediate-level waste (ILW) | | | Deep geological disposal (research ongoing under Section 3 of the Act of 28 June 2006, codified) ILW-LL |
| High-level waste (HLW) | Not applicable* | Deep geological disposal (research ongoing under Section 3 of the Act of 28 June 2006, codified) HLW | |

*The category of very-short-lived high-level waste does not exist.

*Table 1. Classification of radioactive waste (Andra, 2015)*

For waste without any specific management solution, development and implementation of treatment processes is monitored under the National Radioactive Materials and Waste management Plan (PNGMDR, 2016).

## 2.4 Germany

### 2.4.1 Waste classification and disposal plans

Radioactive waste disposal policy in the Federal Republic of Germany is based on the Government decision that all types of radioactive waste with short-lived and long-lived radionuclides (RN) are to be disposed of in deep geological repositories within the country (BMUB, 2014). Due to that there is a necessity to differentiate between the heat-generating wastes (e.g. high level wastes, HLW) and wastes with negligible heat generation (e.g. low and intermediate level wastes, LILW).

According to the German Atomic Energy Act (§ 9a para. 3), the provision of disposal facilities for radioactive waste, as well as the establishment of the scientific and technical basis for their realisation, is a federal task. Therefore, deep geological disposal facilities are to be established at two sites: the Konrad disposal facility for LILW and a disposal facility particularly for HLW (BMUB, 2015). The Konrad disposal facility, a former iron ore mine, is expected to become operational in the year 2022; the operational phase of the repository for the emplacement of the licensed waste volume of 303,000 $m^3$ is not to exceed 40 years. On 23rd July 2013, the German Bundestag passed the Act on the search for and selection of a site for a repository for heat-generating radioactive waste and for the amendment of other laws (Standortauswahlgesetz – StandAG). The site selection act entered into force on 27th July 2013. The objective is to find a site for a repository which ensures the best possible safety for storage of HLW over a period of one million years. The site selection procedure is to be concluded by 2031. The Site Selection Act stipulates a science based, transparent and comprehensible selection process with a broad involvement of the public. Rock salt, clay and crystalline rocks are considered as possible host rock types to be evaluated in a comparative site selection process. The German Federal Government plans to take the disposal facility into operation around the year 2050. Until a respective repository for HLW is available, spent nuclear fuel and reprocessing wastes are placed in interim dry storage in transport and storage casks. Apart from on-site storage facilities at the sites of the nuclear power plants, transport cask storage facilities at Gorleben, Ahaus and the Rubenow ("Zwischenlager Nord") are available.

### 2.4.2 Disposal of HLW

Disposal of heat-generating radioactive waste is still a great challenge, since no repository concept and site for HLW disposal has been selected so far. According to the current report on national strategy on management of nuclear wastes (BMUB, 2015) over 8,379 t HM (or 28,994 fuel assemblies) in form of spent nuclear fuel have been generated in Federal Republic of Germany; these wastes have to be disposed directly. This amount is expected to rise up to 10,500 t by the end of decommissioning of commercial NPPs, whereas 10-12 t Mg are expected from research reactors. Besides that, 291 casks with vitrified HLW from reprocessing is awaiting disposal. Generally, about 28,000 $m^3$ of conditioned HLW is expected to accumulate by 2080. The general criteria for HLW disposal were formulated by Ministry for the Interior (BMI - Bundesministerium des Innern) in 1983: the “Safety Criteria for the Disposal of Radioactive Waste in a Mine” (Bundesministerium des Innern, 1983). In 2007 the Company for Plant and Reactor Safety (GRS - Gesellschaft für Anlagen- und Reaktorsicherheit mbH) issued an update of the 1983 safety criteria, considering the state-of-the-art of science and technology as well as international recommendations published in particular by IAEA and ICRP most recently (Batles & Brennecke, 2008). The main points addressed are the isolation of the heat-generating waste in the isolating rock zone, demonstration of safety (i.e. appropriate containment of radionuclides) for approximately one million years, conducting a stepwise approach, and executing a continuous safety-related optimisation process. The most actual document on "Safety Requirements Governing the Final Disposal of Radioactive Waste" was issued by Federal Ministry for Environment, Nature Conservation and Nuclear Safety (BMU - Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit) in July 2009, stipulating the protection objectives, safety principles, repository design requirements and documentation. Subsequent to that document, additional requirements were formulated, stating that the retrievability of the waste packages must be possible during the operational phase of the repository, and the recovery of the waste packages as a measure of an emergency situation must be possible for a period of time of 500 years after closure and sealing of the repository (Arens, 2010).

### 2.4.3 Disposal of LILW

The licensing procedure for the repository LILW, Schacht Konrad, was started on August 31, 1982 (Brennecke, 2011). The license was issued on May 22nd, 2002, for the emplacement of waste packages of 303,000 $m^3$ at maximum. Of this, approximately 150,000 $m^3$ will originate from the operation as well as from the decommissioning and dismantling of various nuclear facilities, respectively. Waste packages are intended to be emplaced at a depth of 800 m to 1,300 m in disposal rooms with a cross-section of 40 $m^2$ and a length of up to 1,000 m using the stacking technique. The operational lifetime is expected to last for 30 or 40 years.

## 2.5 Lithuania

The objectives and goals of management of spent nuclear fuel as well as radioactive waste are defined in the Radioactive Waste Management Development Program (further Program) approved by Government in December 2015 (Gov. of Rep. of Lithuania, 2015).

According to normative document (VATESI, 2017) solid radioactive wastes are segregated into different classes in accordance to their radiological properties and technological peculiarities of their management, i.e. short-lived very low level waste (class A), short-lived low level waste (class B), short-lived intermediate level waste (class C), long-lived low level waste (class D), long-lived intermediate level waste (class E) and high level waste (class G). Spent sealed sources are attributed to separate class F. Waste with radionuclide specific activity below clearance levels are attributed to free release waste which can be managed as conventional non-radioactive waste.

Lithuania's legislation allows very low-level short-lived waste to be disposed of in a simple near-surface repository of landfill type. Here, treated and untreated radioactive waste that meets acceptance criteria defined during safety assessment, could be placed. The construction of the landfill is ongoing. Operation of the landfill repository is planned for 2018 – 2038 period.

A design of a near-surface repository (vault type) for low and intermediate-level short-lived waste in Lithuania at Stabatiske site close to Ignalina NPP has been completed in 2017 and permission for repository construction has been approved by authorities. Operation of the near-surface repository is planned for 2022 – 2038 period.

The Program (Gov. of Rep. of Lithuania, 2015) foresees only one alternative for spent nuclear fuel and long-lived radioactive waste disposal, which is the implementation of a geological repository in a suitable geological formation in Lithuania. The proposed repository concept for Lithuania is based on the KBS-3 concept developed by SKB for disposal of spent nuclear fuel in Sweden (SKB, 1999). The repository would be constructed in the crystalline basement at a depth of 300 – 500 m. Lithuania is in less advanced program stage. The construction and commissioning of the repository is planned for completion in 2066.

Requirements on Nuclear Safety (VATESI, 2015) define the requirements on WAC for NSR.

In the case of the other type of repositories (landfill, geological), responsibilities to develop WAC are established in Requirements on Nuclear Safety (VATESI, 2016). According to the document, the licensee of repository construction is responsible for WAC development.

As implementation of landfill and NSR already started in Lithuania, WAC for landfill facility as well as NSR have been derived and presented as parts of the respective PSARs (Preliminary Safety Analysis Reports) which have been approved by the State nuclear power safety inspectorate (VATESI).

WAC for geological repository is still not defined.

## 2.6 Slovakia

### 2.6.1 Waste classification

In Slovak National Programme, a newly established classification of radioactive waste is highlighted in Regulation of Nuclear Regulatory Authority of Slovak Republic (SR) No. 30/2012 Coll., based on safety standard of IAEA Classification of Radioactive Waste. IAEA Safety Standards (§ 5 of the Regulation (IAEA (2009)) divides radioactive wastes into five classes as follows (Nuclear Regulatory Authority of SR (2012a):

*a) temporary radioactive wastes: activity will decrease under limit value for their release to environment during storage ( very short half-life radionuclides) ,*

*b) very low level radioactive wastes: activity is slightly higher than limit value for their release to environment. They contain primarily radionuclides with short half-life, eventually also radionuclides with long half-life in low concentration. They require low grade of isolation from environment in disposal, thanks to engineered barriers or no engineered barrier at all. The period of institutional control of repository is shorter than in case of a surface type repository,*

*c) low-level radioactive waste: average mass activity with long half-life radionuclides. The content of radionuclides emitting alpha radiation is under 400 Bq/g, but can locally reach a maximal value under 4 000 Bq/g. They do not produced residual heat. After conditioning, they fulfil the limits and conditions for safe operation for surface type repository,*

*d) intermediate-level radioactive wastes: average mass activity with long half-life radionuclides. The content of radionuclides emitting alpha radiation is equal to 400 Bq/g or higher. They can produce residual heat. Measures for its removal are lower than in case of high level active radioactive wastes. After conditioning, they do not fulfil limits and conditions for safe operation for surface type repository,*

*e) high-level radioactive wastes: average mass activity with short and long half-life radionuclides. The content of radionuclides emitting alpha radiation exceeds values stated for low and intermediate-level radioactive wastes. They are disposable only in deep type repository, whereby measures for removal of residual heat are a significant design factor.*

### 2.6.2 Surface disposal of low-level radioactive waste

Slovakia has a National repository in Mochovce. It is a surface repository for low-level radioactive wastes (JAVYS, 2018).

Specified waste arising from NPP A1 and NPP V1 decommissioning (or from decommissioning of other NPPs in SR) which are not acceptable in Repository Mochovce will probably be deposited in deep repository. Pending the availability of a deep repository, a safe long-term storage was built: The Integral Waste of Radioactive Waste.

The National Radioactive Waste Repository is a multi-barrier surface type storage facility. Its purpose is the final storage of solid and solidified low active radioactive waste from the operation and decommissioning of NPPs, or from research institutes, laboratories and hospitals, from all over the Slovak Republic (IAEA, 2000) (IAEA, 2012).

A suitable location for the facility has been selected, following engineering-geological and hydro-geological prospecting. The selection criteria were in compliance with current legislation, IAEA safety procedures, and requirements for the placement of nuclear facilities, while also taking into consideration the actual proposed design of the facility and the properties and the type of the radioactive waste to be stored. Construction started in 1986 with the laying-down of a clay sealing bed, an important separating barrier standing between the stored radioactive waste and the environment. Since the storage compound was classified as a nuclear facility, stringent demands on working standards applied. Due to this, and also due to recommendations of an IAEA international mission, several changes were proposed and realised, the aim of which was the increased nuclear safety of the storage facility. The storage was put into service in 2001.

Protective barriers against the release of radioactivity into environment are in this case a matrix, in which the waste is fixed, the walls of fibre-reinforced concrete containers (FCC), the steel-reinforced concrete structure of the storage facility itself, the filling between individual boxes, multilayer shielding on the top and the clay bed. The final barrier is the almost impermeable geological formation in which the facility is situated. The storage facility also features a drainage system for collecting and monitoring water, in the eventuality of extremely unfavourable circumstances, if water seeped inside the storage boxes. The drainage system is one of several engineering barriers that prevent negative influences of the storage facility on the environment.

The storage facility comprises an array of storage boxes, arranged in rows and double rows. The first double row is shielded by a steel hall. The Boxes are made of steel-reinforced concrete, measuring 18 × 6 × 5.5 m, the walls are 600 mm thick. In stage 1, two double rows, i.e. 80 storage boxes were constructed. Ninety fibre-reinforced concrete containers can be fitted into one box with each of the containers measuring 1.7 × 1.7 × 17 m, the walls of the containers being 0.1 m thick. The total capacity is 7,200 containers with a total volume of 22,320 $m^3$.

The storage facility is receiving treated low activity waste, sealed with a cement mixture, in fibre-reinforced concrete containers that are brought in from the Bohunice Treatment Center as well as the Mochovce LRW FTF. After initial inspection, marked containers are unloaded from the transport into a storage box, to a predetermined position.

Since the beginning of operations, until the 30 June 2017, 4958 FCCs have been stored at the NRWR.

Due to the premature shutdown of the two V1 NPP units at Jaslovské Bohunice, the need for constructing new storing capacities for the deposit of radioactive waste from the decommissioning, within the existing NRWR Mochovce site, is growing.

Technical parameters:

| | |
|---|---|
| Area | 11.2 ha |
| Number of storage boxes in a row / total | 20 / 80 |
| Box sizes | 18 × 6 × 5.5 m |
| Usable box volume | 510 $m^3$ |
| Storage box capacity | 90 containers |
| FCC volume | 3.1 $m^3$ |
| Total storage capacity | 7,200 containers |
| Total useful storage volume | 22,320 $m^3$ |

### 2.6.3 Deep disposal for intermediate-level and high-level radioactive waste

Deep repository for intermediate-level and high-level radioactive wastes and for spent nuclear fuel is in development process. Works on programme for development of deep geological repository in the Slovak Republic began in 1996. A whole dossier and studies was developed concerning:

- selection of locality,
- approaches to the project of deep repository,
- demonstration of repository safety from the method point of view and analysis of processes and phenomena concerning migration of radionuclides from disposed fuel up to the biosphere,
- public participation,
- coordination, planning, assessment and cross-sectional activities (system for quality management, legislative issues, international cooperation, etc.).

While carrying on working within the program itself, the Slovak Republic participated actively in international activities, which would lead to implementation of deep repositories shared by more states of Europe – firstly by participation in scientific-research projects of framework programs of the EU, later by participation in relevant work teams. From 2010 company JAVYS, a. s., which resumed the Programme for development of Deep repository in SR has become the implementer of deep disposal in The Slovak Republic. In updated Feasibility study, which is being prepared within the project for development of Deep repository (2013-2016), the term of Deep repository commissioning is 2065.

### 2.6.4 Main deadline summary

The main objectives and time schedules in Slovak Republic are the following ones:

- To construct a Repository of Very low-level wastes: 2018, JAVYS, a. s.,
- To construct other repository structure after filling of the second double row of National Repository in Mochovce: 2018, JAVYS, a. s.
- To take decision on continuation or termination of double road in development of deep depositing – to review completely the idea of common international deep repository: 2020, Ministry of Economy of SR,

- To develop plan for other stages of renew development of deep disposal: 2026, JAVYS, a. s.
- To take decision on placement of Deep Repository of SR (in case of cancellation of double road): 2030, JAVYS, a. s.
- To commission Deep Repository: 2065, JAVYS, a. s.

### 2.6.5 Waste Acceptance Criteria

WAC are developed by the designer of the repository according to the safety assessment.

WAC are approved by the supervisory bodies of the Slovak Republic:

- The control and supervision of National Repository are carried out by the internal supervisory and control bodies responsible for the following activities:
  - a) Control of all procedures and programs influencing nuclear and radiation safety.
  - b) Approval of all proposed tests and experiments that affect nuclear and radiation safety.
  - c) Assessment of all proposed changes to limits and conditions.
  - d) Assessment of all proposed changes, modifications of repository systems and facilities that affect nuclear and radiation safety.
  - e) Investigation of all cases of breach of the limits and conditions, their evaluation and the development of measures against recurrence.
  - f) Investigation of the causes of safety-relevant events.
  - g) Assessment of operational events and their written notification to superiors and control bodies.
  - h) Assessment of the operation of the repository in view of possible safety problems.
  - i) Performing special checks and analyses, as instructed by senior operating authorities
  - j) Checking and practicing emergency plan of repository, co-ordination with NPP Mochovce in the framework of NPP Mochovce emergency plan and designing their changes.
  - k) Checking the repository´s physical protection plan and proposing its changes.
- Security controls are controlled by the state supervision authorities in the following areas:
  - a) nuclear safety - the Nuclear Regulatory Authority of the Slovak Republic,
  - b) Radiation Protection - Public Health Service of the Slovak Republic Bratislava,
  - c) Safety of work and technical equipment - National Labor Inspectorate of SR
  - d) fire protection: Presidium of the Fire and Rescue Corps,
  - e) environmental protection - Slovak Environmental Inspection,

In addition to state control authorities, the safety of the repository's operation is controlled by the supreme authorities of the operator.

WAC are revised when a new type of waste to be placed in the repository is not yet approved by Nuclear Regulatory Authority of the Slovak Republic.

## 2.7 Switzerland

For the purpose of management and disposal of radioactive waste the Swiss Nuclear Energy Ordinance (Art. 51) states three different waste categories:

- High level waste (HLW): spent fuel and vitrified reprocessing waste
- Alphatoxic waste (ATA): waste in which the content of alpha emitters exceeds 20,000 becquerels per gram of conditioned waste
- Low and intermediate level waste (L/ILW): all other radioactive waste

In 2008, the Swiss Federal Council has adopted the "Sectoral Plan for Deep Geological Repositories" (SGT), which determines the procedure and criteria for the Swiss site selection process. In contrast to the waste management programmes of other countries, all radioactive waste (incl. L/ILW) will be disposed of in a deep geological repository. In Switzerland, it is planned to construct either two separate repositories (HLW and L/ILW[1]) or one combined repository. The SGT is divided into three stages. Currently, the third stage (SGT-E3) is running, in which Nagra as implementer in performing detailed investigations and safety evaluations for potential sites. In the end of stage 3, Nagra will prepare the general license application for one or two deep geological repositories and submit it to the Federal Office of Energy.

In order to obtain waste forms, which are suitable for transport, interim storage, and final storage in a deep geological repository, raw waste is mixed with additives to fix the material in a container. Typical additives are cement, polystyrene, and bitumen. These waste forms are conditioned with filler materials such as cement in the container for intermediate storage. Optional fittings and shielding installations may apply. If necessary, the filled waste container can be put into an additional overpack container.

Previous to the production of a new waste type, the waste producer has to describe the characteristics of the conditioned waste in a specification. Based on this, Nagra evaluates the final disposability of the new waste tape in a GDF. In case of a positive evaluation, Nagra issues a so called final disposability certificate to the waste producer. Among others, this is a requirement for the authority ENSI (Eidgenössisches Nuklearsicherheitsinspektorat / Swiss Federal Nuclear Safety Inspectorate) to give the approval for the production of this new waste type.

The evaluation of the final disposability of radioactive waste is based on several requirements, which have to be fulfilled. These are:

- Principles on the conditioning method for radioactive wastes
- Requirements on the characterisation, technical description, and characteristics of the waste packages

## 2.8 United Kingdom

### 2.8.1 Classification of Radioactive Wastes

In the UK, radioactive wastes are classified according to the type and quantity of radioactivity they contain and how much heat is produced (NDA, 2015; Pöyry Energy Limited & Amec

[1] The L/ILW repository will be designed in a way that ATA waste can be disposed of in this facility.

Foster Wheeler plc, 2017). The UK Radioactive Waste Inventory is broken down by the following classifications:

- HLW – Highly Active Liquor (HAL) generated from reprocessing of spent nuclear fuel is vitrified into a stable wasteform stored within stainless steel canisters. Reprocessing is the only source of HLW in the UK.
- ILW – predominantly steels, graphite, concrete, cement and sand, sludges, ion exchange resins and flocs.
- LLW – includes building rubble, soil, and steels from the dismantling and demolition of nuclear facilities.
- VLLW – a subset of LLW, primarily building structural materials from the dismantling and demolition of nuclear facilities.

In total, 24 different waste groups span these classifications (with the exception of HLW, which has its own single waste group as all HLW is managed through vitrification). The waste groups are defined in Table 2 below.

| | | | |
|---|---|---|---|
| Activated metals | Contaminated other materials | Graphite | Organic ion exchange material |
| Activated other materials | Desiccant and catalysts | HLW | Plutonium contaminated materials (PCM) |
| Asbestos & other insulation materials | Flocs | Inorganic exchange material | Raffinate |
| Concrete & rubble | Fuel cladding & miscellaneous wastes | Miscellaneous contaminated materials | Sludges |
| Conditioned waste | Fuel element debris | Mixed wastes | Soil |
| Contaminated metals | Fuels & uranium residues | Oils & other fluids | Uranium & thorium contaminated materials |

*Table 2. UK radioactive waste groups (Pöyry Energy Limited & Amec Foster Wheeler plc, 2017)*

Radioactive wastes can also be categorised as

- **Higher Activity Waste (HAW)** – HLW, ILW and some LLW which is unsuitable for disposal in the Low Lever Waste Repository (LLWR).
- **Lower Activity Waste (LAW)** – LLW and VLLW.

It is worth noting that there are additional populations of radioactive materials within the UK which are not currently classified as wastes (e.g., plutonium, uranium, spent nuclear fuels) which may need to be managed as wastes in the future in the event that a decision is taken that they will not be re-used.

### 2.8.2 Radioactive Waste Management Plans

The 2007 UK Government policy statement on LLW management ("Policy for the Long-Term Management of Solid Low Level Radioactive Waste in the United Kingdom") and 2008 white paper ("Managing Radioactive Waste Safely") together outline the long-term management approaches within the UK for the management of radioactive waste, and summarise the respective positions of the devolved administrations (Welsh Assembly, Northern Ireland Executive, and the Scottish Government).

**Low Level Waste**

Waste management plans for LLW are focussed on minimising the amount of wastes which require disposal. Most solid UK LLW is routed to the LLWR near Drigg, in West Cumbria; solid LLW produced at the Dounreay and Vulcan sites in Scotland is sent to a LLW disposal facility adjacent to the Dounreay site.

Some landfill sites can accept high volume VLLW alongside non-radioactive wastes – low volume VLLW which meets certain radionuclide limits can be disposed of at unspecified destinations.

Metals recycling, incineration, compaction and grouting are among the main techniques used in the treatment and packaging steps when disposing of LLW. Some or all of these techniques can be employed depending on the nature of the waste.

**Higher Activity Waste**

The UK Government policy for HAW, supported by the Welsh Government and the Northern Ireland Executive, is for ultimate disposal of HLW and ILW (and the small proportion of LLW unsuitable for disposal at LLWR) in a GDF. Current planning assumptions are that ILW emplacement in the GDF will begin in 2040, and HLW and (if required) spent fuel emplacement will begin in 2075 (NDA, 2014a). Intermediate storage of ILW and HLW near to its point of origin is ongoing until the GDF is available.

Work is ongoing to locate a site for the GDF and determine the engineering features it requires. Radioactive Waste Management Ltd (RWM), as a wholly-owned subsidiary of the Nuclear Decommissioning Authority (NDA), is responsible for the GDF design and siting process, and for determining and issuing guidance on acceptable waste package specifications to waste consignors.

The Scottish Government policy for HAW is centred around long-term management in near-surface facilities, located near to the waste consignor. Again, packages will be held in interim stores until the final disposal routes are available.

# 3 Identification of criteria for the disposability of thermally treated waste products

Waste acceptance criteria of interest for the disposal of thermally treated waste products and associated secondary waste are identified in this section. A table has been proposed to partners to carry out the identification work, and used by some then. The empty table can be found in appendix A.

In this section, criteria could be quantified of simply qualitative (see part 1.3).

## 3.1 Belgium

In Belgium, the WAC are established based on the General Rules, which are approved by the appropriate minister. The waste acceptance criteria define the minimum rules on a mechanical, physical, radiological and chemical basis to which the primary package of (un-) conditioned waste have to fulfil in order to be accepted by NIRAS/ONDRAF. In the past, WAC were related exclusively to criteria for interim storage of radioactive waste whether or not in treated and conditioned form.

In the framework of the license procedure for the surface disposal of low and intermediate short lived waste (waste of category A), new waste acceptance criteria are being developed at which the radioactive waste should comply with in order to be conform with the operational and final surface disposal requirements. In the table, the requirements and conformity criteria for surface disposal are given based on the current knowledge. In general, one can distinguish between radiological, physical and chemical requirements. Some of the requirements are specifically defined for a cement conditioning-matrix as this is the most frequently used matrix for waste conditioning. In order to assure the absence of any other disturbance than the ones already described in the criteria, NIRAS/ONDRAF is taking part in various R&D programs which should allow enlarging the current scientific knowledge and excluding any other chemical disturbance. The criteria will be formalised and finalised once the licence for surface disposal is obtained (NIRAS/ONDRAF, 2017).

For waste of category C and B, no conformity criteria are available as there is no national policy on the final disposal concept up to now. The currently existing WAC for this waste category are thus limited to criteria for interim storage and cannot be considered as the final WAC to which the possibly thermally treated waste should fulfil in order to be considered as disposable and are therefore not included in the table.

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility (if applicable): | Surface disposal facility. NIRAS/ONDRAF submitted the nuclear licence application, necessary to start the construction of the surface repository for category A waste. This application is pending. | | | | |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |
| Physical dimensions, weight | All the packaging should be approved by NIRAS/ONDRAF. The used materials have to be physically and chemically compatible with the materials and infrastructure used for transport and storage and the waste itself. The maximum weight depends on the type of packaging. For drums with a primary packaging (PP) of 400L the weight is limited to 1.5 ton. | | relevant | handling and operation | |
| | The integrity of the packing should not be affected by handling operations, conditioning or storage. | | relevant | handling and operation | |

| | | | | | |
|---|---|---|---|---|---|
| Integrity | The 95% confidence interval on the compression strength of the final waste form on 28 days should be determined.<br>The bend strength of the final waste form on 28 days should be larger or equal to 1MPa.<br>After submersion in water for 90 days, the waste form may not show any structural change and the bend strength should be at least equal to 1MPa. | The compression strength gives an indication of the quality of the waste form whereas the bend strength gives an indication of the presence of fractures in the waste. The submersion in water gives an indication of the durability of the waste form and its behaviour in contact with water. This is important both for the operational phase as well as on the long term. | relevant | handling, operation and after closure safety | high priority |
| Activity content | Two criteria X and Y, based on the radiological spectrum expressed respectively in activity concentration and in activity determine the class to which the waste belongs | The activities and activity concentrations are compared with the maximum values of the radionuclides that are considered to be relevant for the safety. | relevant | handeling and operation | |
| | Depending on the storage building, limits exist on the volume-activity concentration and the surface contamination of alpha-emitters and beta-gamma emitters. | | relevant | handling and operation | |
| | In each primary package, the activity concentration of each of the critical nuclides should be lower than the defined specific concentration limit. | | relevant | after closure safety | high priority |

<table>
<tr><td rowspan="3">Radionuclide inventory</td><td></td><td rowspan="2">The maximum radionuclides concentrations for each primary package and each monolith should be limited regarding the acceptable limitations of the mean concentrations in the disposal.</td><td rowspan="3">relevant</td><td rowspan="3">handling, operation and after closure safety</td><td></td></tr>
<tr><td>The radiological heterogeneity factor should be equal or smaller than 60.</td><td></td></tr>
<tr><td>The sum of the activities of Ra-226 and Th-232 divided by the mass of the waste may never be larger than 1000 Bq/kg. The mass of the waste is equal to the total mass of the primary waste package minus the mass of the packaging.</td><td>Radon emission from the waste may never disturb the radiological measurements of the disposal site. Therefore, no radium or thorium containing waste may be allowed in the disposal.</td><td></td></tr>
<tr><td>Dose rate</td><td>The dose rate of monolith may not exceed 20 mSv/h</td><td>The dose rate should be conform to the values of the hypotheses of the design of the disposal. Therefore, only a limited number of monoliths with a dose rate of 20 mSv/h can be allowed in the disposal.</td><td>relevant</td><td></td><td></td></tr>
<tr><td>Surface contamination</td><td>The removable surface contamination of a monolith may not be higher than 0.4 Bq/cm² for the total of beta/gamma-emitters and the low radiotoxic alfa-emitters and not higher than 0,04 Bq/cm² for the total of other alfa emitters.</td><td>The removable surface contamination should apply to the transportation requirements.</td><td>relevant</td><td>handling, operation and after closure safety</td><td></td></tr>
</table>

| | | | | | |
|---|---|---|---|---|---|
| Nuclear criticality | For a monolith containing only 400 L (or 600 L PP): in every PP the sum of the masses of U-235 and Pu-239 may not exceed 50 g or the mass of Pu-241 may not exceed 86 g. For every other type of primary package of conditioned waste, the sum of the masses of U-235, Pu-239 and Pu-241 may not exceed 15 g. | There may not be a risk of criticality in the disposal site. | relevant | Operation and after closure safety | |
| | For a monolith with packages of conditioned waste with uranium with an enrichment of 20% or more, the mass of U-235 may not exceed 15 g. | Waste with a non-negligible amount of nuclear fuel may not be accepted in a surface disposal | | Operation and after closure safety | |
| Containment | In every waste package with waste conditioned in a cement matrix, the amount of hydrated cement should be larger or equal to 10 m% of the mass of the package. The criteria for packages with conditioned waste other than a cement matrix are still to be determined. | The waste should possess the chemical and physical characteristics to limit the leaching of radionuclides in the disposal. | relevant | after closure safety | low |
| Thermal output | Not relevant for waste of this category | | | | |

| | | | | | |
|---|---|---|---|---|---|
| Radiological gas generation | A possible gas generation (by corrosion, radiolysis, and degradation of organics) may not lead to a degradation of the waste form or to an unacceptable deformation of the primary packaging. | Gas production may not disturb the other components of the disposal system | relevant | After closure safety | |
| Non radiological gas generation | The radioactive waste may not contain metals that would dissolve and produce hydrogen gas in an excessive way. This corresponds with metals with a redox potential lower than -0.84 V SHE | Gas production may not disturb the other components of the disposal system | relevant | After closure safety | |
| Chemical content | PP 220 L: 0.08 kg cellulose<br>PP 400 L: 0.1 kg cellulose<br>PP 600 L/1000L/1500L/1600L: 0.4 kg cellulose | The complexation by the presence of cellulosic containing waste may not significantly increase the radiological impact by leaching on the long term. A maximum limit for cellulose per monolith will be defined. (Probably 0.4 kg cellulose/monolith) | relevant | After closure safety | |
| | Chloride limit based on the mass of cement. No limits exist for other conditioning matrices. | Chloride ions may not have a negative impact on the concrete barriers due to an increased corrosion speed of the rebars. In addition, the presence of chloride ions and the resulting complexing may not lead to increase of the radiological impact by | | After closure safety | |

| | | | | | |
|---|---|---|---|---|---|
| | | leaching of the radionuclides. | | | |
| | First, the complexing agents present in the waste should be identified and quantified. Second, research has to be done to show that the amounts present in the waste have no impact on the safe disposal of the waste | The presence of complexing agents in the monolith should be limited as they can increase the radiological impact on the long term by leaching | relevant | After closure safety | |
| | The amount of sulfates ($SO_4^{2-}$) should be limited to 12 g/ kg waste. The mass of the waste is the mass of the primary packages minus the packaging weight. All sulfates should be excluded except for $BaSO_4$ | The presence of sulfates may not lead to a degradation of the concrete barriers in the timeframe during which they are considered as non-degraded. | relevant | After closure safety | |
| | Substances which are according to the CLP regulation (1272/2008) labelled as “physical hazards” and for which a specific hazard statement exist, are prohibited. | Hazardous substances in the radioactive waste may not cause a disturbance of the performance of the monolith or the stability of performance of the disposal | relevant | After closure safety | |
| | The radioactive waste may not contain free liquids | Contaminated liquids may not leach from the waste. In addition, the presence of water as free liquid should be avoided as this can serve as medium for disturbing chemical reactions | relevant | Operation and after closure safety | |

| | | | | | |
|---|---|---|---|---|---|
| Chemical durability | The waste form should be insensitive to ASR (Alkali-Silica-Reaction) and DEF (Delayed Ettringite Formation) and this should be proven by suitable test methods. | During the period at which the physical integrity of the systems, structures and components surrounding the waste have a safety function, processes that could lead to an expansion of the waste should be avoided. | relevant | Operation and after closure safety | |
| Volume of voids | No real criteria exist | The volume of voids are in principle limited to 20 % | | | |
| Stackability | See Integrity | | | | |
| Impact performance | No information available | | | | |
| Fire performance | No information available | | | | |
| Identification | Every package should be provided with an accepted, durable and unique identification code which is unequivocally linked with the production documentation | | | | |
| Quality control | The waste producer must have a quality management system | | | | |

| | | | | | |
|---|---|---|---|---|---|
| Quality assurance | including all steps from the raw material to the conditioned waste form. N/O should agree with the proposed way of working | | | | |
| Data requirements | No information available | | | | |

## 3.2 Finland

In Finland, general guidance from the regulator (YVL) and from international organisation such as IAEA and WENRA are applied. However, as low and intermediate level wastes are mainly managed by waste producers, relatively little information about waste acceptance criteria is available in the open literature.

For WAC, national guidance YVL D.4 and YVL D.5 also have to be applied for LILW. However, it is difficult to set unambiguous acceptance criteria for LILW due to waste type differences, handling technologies and disposal concepts (e.g. waste packages). Therefore, WAC for waste and waste packages have to be derived from the safety analysis report and from the safety case. However, the Final Safety Analysis Report (FSAR) shall include a description of each waste package category to be disposed of; such descriptions shall include at least:

- waste type and conditioning specifications
- surface dose rate
- upper bounds for the activities of the most significant radionuclides
- average values of other properties relevant to safety, such as:
    - mechanical strength
    - chemical durability,
    - radionuclide release characteristics (leaching or diffusion rate),
    - free liquid content,
    - flammability,
    - swelling capacity,
    - gas generation potential,
    - concentrations of substances which may degenerate the waste package or decrease sorption in surrounding media

## 3.3 France

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility (if applicable): | The table is completed assuming that thermal processing will lead to ILW.<br>The French geological disposal dedicated to ILW-LL and HLW is under study. The main safety options are described in a file forwarded to the French Safety Authority in 2016.<br>In the reference solution, the primary packages delivered by the producers will be introduced into specific standardised storage packaging (13 different types).<br>Direct storage of primary packages in underground facilities is also under study. | Characteristics of waste packages must not challenge the operation safety or corrupt the containment properties of the clay layer.<br><br>The role of the reference storage container is mainly to standardise package handling, to provide mechanical and thermal protection (in case of fire) to primary packages, to allow gas release | | | |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |

| | | | | | |
|---|---|---|---|---|---|
| Physical dimensions, weight | Primary packages must be congruent to standard specified designs (specific sizes).<br>The relevant maximum weights are also specified.<br>The maximum weight and volume are respectively about<br>* 11 tons, and 5 $m^3$ for primary packages<br>* 17 tons, and 10 $m^3$ for storage packages | This ensures that packages can be handled with the tools available in the facility and be introduced into standard storage containers and potentially be stacked.<br>The specified dimensions and weights are derived from the waste primary package characteristics already produced. Future primary packages will be produced in line with these specifications. | Relevant | Handling | |
| Integrity | See containment.<br>More precisely, integrity of gripping surfaces has to be maintained. | | Relevant | Handling (including retrievability) and operational safety | |
| Activity content | There is no explicit activity criterion. However, the radioactive content is limited indirectly by other criteria such as dose rate, heat output and material release in case of package drop. | | Relevant | | |
| Radionuclide inventory | The activity of 144 radionuclides must be declared; declaration thresholds have been defined for each of them. | | Relevant | Operational and after closure safety | |

| | | | | | |
|---|---|---|---|---|---|
| Dose rate | Specific external dose rate limits are being defined for radiation protection purpose. | Those limits will derive from the thickness and efficiency of facility's shielding. | Relevant | Operational safety | |
| Surface contamination | Non-fixed contamination on the primary packages external surface must be less than : 0.4 Bq/cm$^2$ for alpha emitters,<br>4 Bq/cm$^2$ for beta and gamma emitters<br>No limit is defined for fixed contamination. | These limits were inspired by transport regulation. | Relevant | Operational safety | |
| Nuclear criticality | Some criteria will be defined in order to avoid criticality risk (fissile material mass limits, isotopic spectrum, moderator type …).<br>To establish these criteria we will take into account all the possible configurations including, if any, the packages accumulation. | In order to avoid criticality risk | Relevant | Operational and after closure safety | |
| Containment | The containment of solid radioactive material inside the primary package is required upon receipt of it on the storage facility. This must be demonstrated and justified by the waste producer.<br>Otherwise, the storage package shall be confining to solid radioactive material all the | The reinforced storage container requirement is derived from the performance of the dynamic containment systems (venting & filtration). | Relevant | Operational safety | |

| | | | | | |
|---|---|---|---|---|---|
| | operation period of the facility (more than a hundred years). | | | | |
| Thermal output | A maximum thermal power is being defined. | In order to prevent the repository structure from temperatures higher than 65 °C for concrete and 90 °C for clay. | Relevant | Operational and after closure safety | |
| Radiological gas generation | A maximum release of radioactive gas per package will be precisely defined. | | Relevant | Operational safety | |
| Non radiological gas generation | A maximum release of hydrogen gas per package will be precisely defined. The limit will be several tenth of liter by year per package.<br>For other gaseous releases no limit is specified at this stage. | This limit is derived from the explosion risk study. | Potentially not relevant after thermal treatment, depend on conditioning and type of matrix. | Operational safety | |
| Chemical content | The following substances are prohibited :<br>- self-explosive substances;<br>- very inflammable substances;<br>- substances and mixtures the more reactive in contact with water (exothermic reaction) and emitting flammable gases, corresponding to category 1 of this class of substances defined by the CLP regulations; | | Probably not relevant for disposal because it would have already been required for treatment and/or conditioning. | Operational safety | |

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| | - free liquids both organic and aqueous;<br>- infectious substances.<br><br>The presence of compressed gas is not formally prohibited but intermediate level waste packages are not supposed to contain compressed gas. | | | | |
| | Pyrophoric or other reactive materials must be declared.<br><br>Producers must demonstrated that such materials/substances are no longer reactive | | See previous comment | Operational safety | |
| | Corrosive substances are not prohibited but should be reported | In order to check containment demonstration | | | |
| | Toxic substances are not submitted to a limit but must be declared. | | | After closure safety | |
| | Complexing compounds or organic materials leading to the release of complexing substances after radiolysis or hydrolysis are not submitted to a limit but must be declared. | | | After closure safety | |
| | Chemical content in general must be declared | In order to check different criteria such as $H_2$ release, containment demonstration | | | |
| Chemical durability | Waste and package characteristics have to be | | Relevant | After closure safety | |

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| | declared as basic data for release models. | | | | |
| Volume of voids | A maximum vacuum of 25% is retained for future packages. | | Relevant | | |
| Stackability | Once received at Cigéo, the primary packages are no supposed to be stacked before introduction in storage container.<br>The storage packages will be staked on a maximum of 3 levels. The dimensional characteristics of the storage packages should not be affected by stacking. | This result from facility design choices. | Relevant | | |
| Impact performance | The requirement is different for primary packages produced or under production and future ones.<br>For future primary packages, the producer shall demonstrate the absence of dispersion of the solid radioactive contents in case of drop from 1.2 meters height onto a non-deformable target. | The activity release limit is derived from the performance of the dynamic containment systems (venting & filtration). The height of 1.2 m corresponds to the maximum drop height during facility operation. | | Operational safety | |
| Fire performance | In case of temperature increase of the primary package the producer has to demonstrate :<br>• the integrity of the package envelope and its closure system; | A primary package temperature increase curve representative of a reference fire will be defined taking into account thermal | | Operational safety | |

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| | • the absence of dispersion of the solid content with the exception of the more volatile radionuclides;<br>• the absence of reaction leading to :<br>- an explosion or self-inflammation of the content;<br>- uncontrolled consequences due to exothermic reactions;<br>• that any geometric changes of the primary package does not challenge its handling. | protection provided by the storage container. | | | |
| Identification | Each primary and storage package shall be marked on the outside with an identification number. The format of this number and the size of characters are not defined. | | Relevant | | |
| Quality control | Packaging data from waste producers (procedure description, etc.)<br>+ monitoring mission from Andra | | | | |
| Quality assurance | | | | | |
| Data requirements | | | | | |

## 3.4 Germany

Radioactive LILW destined for the Konrad repository has to be appropriately conditioned and subjected to product control by the Federal Company for Radioactive Waste Disposal (BGE) to verify compliance with the waste acceptance criteria (WAC). In the current version of the Konrad WAC the following requirements are given (Brennecke, 2014):

1. Limits in surface dose rate for waste packages with different geometries.
2. Limits in surface contamination (per 100 $cm^2$) for a-, b-emitters and other RNs.
3. Maximum pressure inside waste package (1.2 bars) that must not exceed.
4. LILW is categorised into 6 groups ("Abfallproduktgruppen" APG): (APG1) bitumen and plastic, (APG2) solids, (APG3) metals, (APG4) compacts, (APG5) cemented wastes, (APG6) concentrates; the characteristics of each waste group is defined in (Arens, 2010).
5. Acceptance only of solid stable (or fixed) radioactive wastes.
6. Wastes must not rotten or ferment inside the waste packages.
7. Liquid and gaseous waste is excluded from acceptance.
8. No flammable or explosive wastes are accepted.
9. The amount of fissile material (e.g. $^{233}U$, $^{235}U$, $^{239}Pu$ etc.) excluding natural and depleted Uranium is limited to 50 g in 0.1 $m^3$ of waste form; criticality limits.
10. Limits of RN inventory in nuclear wastes for 156 specified RNs.
11. WAC define the type of the waste container to be used for certain waste groups, which results from the incident analysis, e.g. the waste package must remain integer in case of fall from 5 m heights or temperature rise up to 800°C.
12. Activity limits are defined for each RN resulting from the safety analysis for the operational phase and the post-operational phase (Annex II, Tables 2 – 7a,b of (Brennecke, 2014)). Additional limitations of the activity of volatile RNs like $^{3}H$, $^{14}C$ or $^{129}I$ is defined based on their release rates from the waste product, which must not exceed 1% of waste inventory over the operational phase of 40 years. This limitation is introduced in order to assure the safety of repository personnel directly dealing with the waste packages, and to limit exposures to the general public via the air pathway during the operational phase of the repository.
13. WAC also limits the inventory (i.e. mass) of inactive organic and inorganic materials (e.g. Beryllium or tributylphosphate), which may potentially be hazardous to local groundwater. These are defined in the permit based on the Federal Water Act ("Gehobene wasserrechtliche Erlaubnis" (BMJV, 1996)) and are listed in Annex IV, in Tables 14 and 16 of (Brennecke, 2014).
14. WAC also define the documentation (Abfallvoranmeldung and Abfalldatenblatt) for each waste package to be disposed of. An example of the waste declaration is given in Annexes V and VI of (Brennecke, 2014).

These selected requirements are described in details in (Brennecke, 2014) and are continuously updated taking into account the state-of-the-art in science and technology.

The criteria listed above put the limitation on the stability of thermally treated product, RN inventory and chemical composition. For disposal in Konrad, the release behaviour of some volatile RN (e.g. $^{3}H$ or $^{14}C$) is critical in order to prevent uncontrolled incorporation of these RN into the human body through inhalation. Therefore it is essential to provide release prognosis over operational time of 40 years, when personnel of the repository would be in the direct contact with released RN. It is considered to use different packaging of waste product, depending on the RN inventory and release rate of some volatile RN (Annex 1, Table 1 of (Brennecke, 2014)).

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility (if applicable): | Schacht Konrad (near Salzgitter, Low Saxony), a former iron ore mine, an underground repository for LILW. Received an operational approval in 2002. | WAC are based on the assumption that the mine will be flooded in a while, and the accommodated waste would equilibrate with the groundwater. Defined inventory of RNs and accompanying compounds must not bring hazard to the groundwater beyond defined by permission based on the Federal Water Act (Wasserhaushaltsgesetz). | Relevant as soon as thermally treated waste product fall within the definition of LILW. (Applied for all cases below). | N.A. | No information |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |
| Physical dimensions, weight | The types of waste containers and dimensions are listed in Annex I, Table 1 of (Brennecke, 2014). Containers are subdivided into three main groups: concrete-containers (1.2 – 1.3 $m^3$), cast-containers (0.7 – 1,3 $m^3$) and simple containers (3.8 – 10,9 $m^3$). | No information | | Operation | No information |
| | Containers integrity must not be compromised during transport, handling and emplacement. | No information | | Operation | No information |

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| Integrity | Containers for disposal of LILW are licenced to retain their integrity for 40 years (operational phase of repository). After closure of repository presumed that container will rust and waste will come in contact with groundwater. | No information | | Operation | No information |
| Activity content | Activity limits are specified for the repository in total and for each container class (Brennecke, 2014). Additional specification is given for volatile RN, which activity depends on their release behaviour (e.g. the higher the release rates, the lower the activity limit specified for a single container). | Specified limits for radioactivity is based on the assumption that all wastes will be (sooner or later) equilibrate with the groundwater, without posing any hazard to the biosphere. | | Operation and post-closure | No information |
| | There are also requirements regarding surface dose rates, surface contamination and criticality (e.g. simultaneous disposal of moderator materials, like Be, and fissile materials, like $^{239}Pu$ or $^{233}U$). | Safety analysis | | Operation | No information |
| Radionuclide inventory | There are in total 156 radionuclides allowed to be disposed in Schacht Konrad. The detailed list of these RN is given in Annex 2, Table 5 and 6 of (Brennecke, 2014). | Safety analysis | | Operation | No information |

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| Dose rate | Average dose rate of 2 mSv/h; locally max – 10 mSv/h; for cylindrical waste packages at 1 m distance and square packages at 2 m distance – 0.1 mSv/h. | Safety analysis | | Operation | No information |
| Surface contamination | 0.5 Bq/cm$^2$ for a-emitters; 50 Bq/cm$^2$ for b-emitters; 5 Bq/cm$^2$ for other RN. | Safety analysis | | Operation | No information |
| Nuclear criticality | Criticality is of a concern (see above). Detailed amount of fissile materials to be disposed in Schacht Konrad is also listed in Annex II, Table 7a and 7b of (Brennecke, 2014). | Safety analysis | | Operation and post-closure | No information |
| Containment | Containers must be tight; the limits for the RN release rates are given in Annex II, Table 2 of (Brennecke, 2014). | No information | | Operation | No information |
| Thermal output | Thermal load of the waste to be disposed in Schacht Konrad must not exceed 3 K. | No information | | Operation and post-closure | No information |
| Radiological gas generation | The waste to be disposed in Schacht Konrad must not generate any gases. Internal pressure is limited to 1.2 bar. | No information | | Operation and post-closure | No information |

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| Non radiological gas generation | The waste to be disposed in Schacht Konrad must not generate any gases. Internal pressure is limited to 1.2 bar. | Safety analysis | | Operation and post-closure | No information |
| | The waste to be disposed in Schacht Konrad must not generate any explosive gases. The limits for release rates of volatile RN, like $^{3}H$, are given in limited in Annex II, Table 2 of (Brennecke, 2014). | Safety analysis | | Operation and post-closure | No information |
| Chemical content | Chemical content including chemical speciation of RNs in the waste product has to be declared | Safety analysis | | Operation and post-closure | No information |
| | Explosive, pyrophoric, hazardous or combustible materials, fire and explosive materials are not allowed in Schacht Konrad. | Safety analysis | | Operation and post-closure | No information |
| | Waste can be corrosive; limits are not specified. Containers are certified for 40 years, with a possibility of further corrosion, which is accounted to the post-closure period and is not a problem for Schacht Konrad. | Safety analysis | | Operation | No information |

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| | Toxic materials and their limits are listed in Annex IV Table 14 and 15 of (Brennecke, 2011). | Safety analysis | | Operation and post-closure | No information |
| | No requirements on complexing compound except for those listed in Annex IV Table 14 and 15 of (Brennecke, 2014). | No information | | Operation and post-closure | No information |
| | No compressed gases are allowed in Schacht Konrad. | Safety analysis | | Operation and post-closure | No information |
| | No free liquids are allowed in Schacht Konrad. | Safety analysis | | Operation and post-closure | No information |
| | Solid stable waste forms are to be disposed in Schacht Konrad. No specification of how RNs have to be immobilised. | Safety analysis | | Operation and post-closure | No information |
| Chemical durability | The waste product must be stable for at least 40 years. | Safety analysis | | Operation | No information |
| Volume of voids | An emplacement of in total 303,000 $m^3$ of waste packages at maximum is foreseen in Schacht Konrad. Each container type has separated specified volume, e.g. 200-L or 400-L. | | | Operation | No information |
| Stackability | No information. | No information | | No information | No information |

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| Impact performance | There is a drop test from 5 m heights to demonstrate an impact performance. Additionally the waste package must withstand the impact of 300°C (fire scenario) and 30 MPa impact of external pressure. | Safety analysis | | Operation | No information |
| Fire performance | See above | Safety analysis | | Operation | No information |
| Identification | Each waste package has to be accompanied with appropriate documentation, summarising the composition, inventory and mount of the wastes. For details see Annex VII of (Brennecke, 2014). | No information | | Operation and post-closure | No information |
| Quality control | The quality control procedure is carried out by the licensing authority (BMU/BfS). | No information | | Operation and post-closure | No information |
| Quality assurance | The quality assurance is carried out by the licensing authority (BMU/BfS). | No information | | Operation and post-closure | No information |
| Data requirements | The data are recorded in the waste accompanying papers as well as in an electronic database. For details see Annex VII of (Brennecke, 2014). | No information | | Operation and post-closure | No information |

## 3.5 Lithuania

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility - Geological repository (**GR**) | Less advanced program in Lithuania.<br>Disposal concept is proposed. | Preliminary WAC is proposed in the scope of this project in accordance to (VATESI, 2016).<br>It is developed **for long-lived intermediate level** (VATESI, 2017) **cement grouted thermally not treated activated metallic waste**. | | | |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |
| Physical dimensions, weight | Not defined by VATESI requirements (VATESI, 2016). | **GR: cement grouted thermally not treated activated metallic waste**<br>Only specified type containers are accepted for disposal.<br>Maximum dimensions for concrete container, mm:<br>- length: 2 400<br>- width: 1 620<br>- height: 1 650<br>Maximum mass of waste package (container + waste + grout) 15 000 kg | Relevant | handling,<br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| | Not defined by VATESI requirements (VATESI, 2016). | **GR: cement grouted thermally not treated activated metallic waste**<br>Compliance with Handling System in accordance as provided by Technical Design (TD).<br>Compliance with Transport System in accordance as provided by TD.<br>Compliance with Loading System in accordance as provided by TD | Relevant | handling | |
| Integrity | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Cementation according to licensee procedures and test of samples.<br>The ability of the immobilised waste to resist solubility and leaching as well as the integrity of waste package shall be identified by applying the standard test procedures (e.g. confirmed by ISO) if such procedures are established. Otherwise the tests shall be performed according to the licenser procedures describing the identification of immobilised radioactive waste resistance to solubility and leaching | Relevant | handling,<br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
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| Activity content | VATESI requirements (VATESI, 2016): Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste** Mass of fissile radionuclides per package according to European treaty on international transportation of hazardous goods by roads (ADR).<br>Total activity in the GR as well as specific activity per package is derived for each radionuclide under consideration. | Relevant | handling | |
| | VATESI requirements (VATESI, 2016): Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>See criteria “Surface contamination” | Relevant | handling | |
| Radionuclide inventory | VATESI requirements (VATESI, 2016):<br>Radioactive waste acceptance criteria must be prepared according to radiological, mechanical, physical, chemical and biological properties of radioactive waste (e.g. radioactivity, gas release, heat | **GR: cement grouted thermally not treated activated metallic waste**<br>Immobilised waste shall be characterised as follows:<br>- Radionuclide content (list of relevant radionuclides); | Relevant | handling,<br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
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| | release, criticality and other properties). | - Specific activity of individual radionuclides in the solidified waste, Bq/kg;<br>- Deviation (uncertainty) of specific activity, Bq/kg;<br>- Total activity of alpha emitters, Bq;<br>- Total activity of beta emitters, Bq;<br>- Total activity of gamma emitters, Bq. | | | |
| Dose rate | VATESI requirements (VATESI, 2016):<br>Radioactive waste acceptance criteria must be prepared according to radiological, mechanical, physical, chemical and biological properties of radioactive waste (e.g. radioactivity, gas release, heat release, criticality and other properties). | **GR: cement grouted thermally not treated activated metallic waste:**<br>According to VATESI requirements (VATESI, 2017):<br><br>Dose rate at contact (10 cm from surface of waste) for long-lived intermediate level waste: > 10 mSv/h | Relevant | handling | |
| | Specific external dose rate limits are not specified by VATESI requirements (VATESI, 2016). | **GR: cement grouted thermally not treated activated metallic waste**<br><br>Not specified | | | |
| Surface contamination | VATESI requirements (VATESI, 2016): | **GR: cement grouted thermally not treated activated metallic waste** | Relevant | handling | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
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| | Radioactive waste acceptance criteria must be prepared according to radiological, mechanical, physical, chemical and biological properties of radioactive waste (e.g. radioactivity, gas release, heat release, criticality and other properties). | Upper limit of beta, gamma and low toxicity alpha surface contamination: 4 Bq/cm$^2$.<br>Upper limit of other alpha surface contamination: 0.4 Bq/cm$^2$. | | | |
| Nuclear criticality | VATESI requirements (VATESI, 2016):<br>Radioactive waste acceptance criteria must be prepared according to radiological, mechanical, physical, chemical and biological properties of radioactive waste (e.g. radioactivity, gas release, heat release, criticality and other properties). | **GR: cement grouted thermally not treated activated metallic waste**<br>Not relevant concerning criticality | | handling,<br>post closure | |
| Containment | Not defined by VATESI requirements (VATESI, 2016). | **GR: cement grouted thermally not treated activated metallic waste**<br>Only conditioned waste shall be accepted for disposal. | Relevant | handling,<br>post closure | |
| Thermal output | VATESI requirements (VATESI, 2016): | **GR: cement grouted thermally not treated activated metallic waste** | | | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
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| | Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | Not relevant for metallic waste and not specified | | | |
| Radiological gas generation | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Not relevant. No radioactive gas release from metallic waste. | | | |
| Non radiological gas generation | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Container water tightness class W6 (K≤2E-11 m/s)<br>For waste matrix K≤5E-05 m/s (as for totally degraded concrete)<br>Container porosity: ≤0.15<br>Waste matrix porosity: ≤0.25 (as for totally degraded concrete) | Relevant | handling,<br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Releases of $H_2$ gas due to metal corrosion must be assessed. If necessary WAC must be specified. | | | |
| Chemical content | VATESI requirements (VATESI, 2016): Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Total ban on reactive chemicals. | Relevant | handling,<br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
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| | VATESI requirements (VATESI, 2016): Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety.<br><br>Radioactive waste must not be easily flammable. | **GR: cement grouted thermally not treated activated metallic waste**<br><br>Total ban of aggressive materials.<br><br>Total ban on biological, pathogenic, and/or infectious materials.<br><br>No materials with ignition temperatures of less than 60ºC.<br><br>Total ban on pyrophoric materials. | Relevant | handling,<br><br>post closure | |
| | VATESI requirements (VATESI, 2016):<br><br>Radioactive waste acceptance criteria must be prepared according to radiological, mechanical, physical, chemical and biological properties of radioactive waste (e.g. radioactivity, gas release, heat release, criticality and other properties). | **GR: cement grouted thermally not treated activated metallic waste**<br><br>Information on tests, including information on corrosion and any damage, performed periodically during interim storage of waste packages at waste generator premises according to existing standard procedures must be provided. | Relevant | handling,<br><br>post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| | VATESI requirements (VATESI, 2016): Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste** Criteria will depend on content of toxic materials.<br>Content of toxic materials is still not defined | Relevant | post closure | |
| | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste** Criteria on complexing compounds not relevant for metallic waste.<br>Not specified. | | | |
| | Criteria on compressed gases not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste** Criteria on compressed gases not relevant for metallic waste.<br>Not specified. | | | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| | VATESI requirements (VATESI, 2016):<br>Radioactive waste must be solid and do not contain free liquids. | **GR: cement grouted thermally not treated activated metallic waste**<br>Total ban on free liquids. | Relevant | handling,<br>post closure | |
| | VATESI requirements (VATESI, 2016):<br>Radioactive waste must be solid and do not contain free liquids. | **GR: cement grouted thermally not treated activated metallic waste**<br>Only conditioned (concrete grouted, glass immobilised) waste shall be accepted for disposal. | **Relevant** | handling,<br>post closure | |
| Chemical durability | Not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste**<br>Depends on content of waste.<br>Content of waste is still not defined. | | | |
| Volume of voids | VATESI requirements (VATESI, 2016):<br>Radiological, mechanical, physical, chemical and biological properties of radioactive waste must not have detrimental effect on repository safety. | **GR: cement grouted thermally not treated activated metallic waste**<br>Voids in radioactive waste package should be avoided strictly following Quality assurance programme (QA). | Relevant | post closure | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Stackability | Not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste**<br>Must comply with Technical design. | Relevant | | |
| Impact performance | Not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste**<br>Container concrete class B30 (SNiP) (compressive strength: ≥25 MPa)<br>Grouting concrete class C40/50(required compressive strength: ≥40 MPa)<br>Container shall withstand free fall from the height of 0.6 m.<br>Minimum compressive strength, for container in accordance to maximum stacking height (up to 6 levels).<br>Resistance to the dynamic loads: shall withstand design basis earthquake of seismic intensity ≤6 (MSK). | Relevant | | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Fire performance | Not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste**<br>Not specified | | | |
| Identification | Not defined by VATESI requirements | **GR: cement grouted thermally not treated activated metallic waste**<br>Unique waste package identification (barcode) for container. | Relevant | handling | |
| Quality control | VATESI requirements (VATESI, 2016):<br>Licensee for repository operation must prepare the procedure which describes actions and means relevant to the verification of compliance of radioactive waste packages to the WAC. | **GR: cement grouted thermally not treated activated metallic waste**<br>Procedure must be prepared by licensee for repository operation | | | |

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Quality assurance | VATESI requirements (VATESI, 2016):<br>Licensee for repository operation must prepare the procedure which describes actions and means relevant to the verification of compliance of radioactive waste packages to the WAC. | **GR: cement grouted thermally not treated activated metallic waste**<br>Must be prepared by licensee for repository operation before repository commissioning | | | |
| Data requirements | VATESI requirements (VATESI, 2016):<br>It must be ensured that the nuclear material accounting and management means will not have negative influence to repository safety and vice versa, repository safety insurance measures will not influence the accounting and management means. | **GR: cement grouted thermally not treated activated metallic waste**<br>According to waste package specification (to be prepared) | | handling,<br>post-closure | |

## 3.6 Slovakia

The limits and conditions contain a summary of the essential organisational, technical and technological conditions that must be respected to ensure a safe decommissioning process of a nuclear facility.

The limits and conditions are processed on the basis of requirements of Slovak legislation, international standards and recommendations, and experience gained in the previous phase of decommissioning of JAVYS, a.s.. They meet the requirements of § 25 of Decree of Nuclear Regulatory Authority of the Slovak Republic No. 58.

The radioactive waste disposal site will be determined according to the classification in Act No. 541, Decree of Nuclear Regulatory Authority of the Slovak Republic No. 53/2006. Radioactive waste is broken down by activity into the following classes (National Council of the Slovak Republic, 2004):

a) temporaly radioactive waste whose storage activity decrease below the limit value for its release into the environment,

b) low-level radioactive waste and intermediate radioactive waste with an activity higher than the limit value for its release into the environment and with a residual heat of less than 2 $kW/m^3$:

   1. short-term radioactive waste which, after treatment, meets the limits and conditions of safe operation for the surface storage of radioactive waste and whose average alpha nuclide mass activity is less than 400 Bq/g,
   2. long-term radioactive waste which, after treatment, does not meet the safe operating limits and conditions for surface storage of radioactive waste or whose average mass of alpha nuclides is equal to 400 Bq/g or higher.
   3. Deep repository in Slovakia is only in planning phase, so we are no able to answer these questions. According Slovak National Programme we will be able to respond to these issues early in 2020 (Board of Governors of NJF (2013).

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility (if applicable): | Surface National repository for low- level radioactive waste in Mochovce in operation since 200. Deep geological disposal is the development. | | | | |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |
| Physical dimensions, weight | Approved package for Mochovce repository is fibre concrete container, made of fibre-reinforced concrete.<br>- Dimensions: 1.7 x 1.7 x 1.7 m<br>- Container weight: 4,200 kg<br>- The maximum weight of the waste container: 15,000 kg.<br>The weight of the waste containers in the expansion unit (4 boxes) of one row must not exceed 3,600 t. | | | | |
| | Each container must be marked with a serial number printed on the specified container. The designation must ensure its unambiguous identification. | | | | |
| Integrity | The outer surface of the waste container shall be free from defects:<br>1. cracks wider than 0.3 mm and at the same time longer than 50 mm,<br>2. angles, tangled edges and damage with dropped concrete pieces of more than 50 mm, more than 200 mm and more than 20 mm; | | | | |

| | | | | | |
|---|---|---|---|---|---|
| | 3. a tear-off surface of the fibre which is larger than 100 $cm^2$, a depth greater than 3 mm;<br>4. Container walls and grip anchorage beds must not be dirty<br>The outer surface of the waste container with reduced INTEGRITY must be free from defects:<br>1. cracks wider than 0.3 mm<br>2. Container walls and grip anchor beds must not be dirty.<br>The total number of waste containers with reduced INTEGRITY must not exceed 200 pc.<br>The container shall be sealed to meet the integrity requirements and the container lid and plugs comply with the loading and unloading conditions of the container after loading onto the Mochovce repository in accordance with the fibre concrete container performance tests (pressure test).<br>The compressive strength (after 28 days for samples of fibre-reinforced concrete) for the body, the lid, the plugs, the lid seals and the cove measured on the cubes shall be greater than 71,5 N.<br>The shrinkage (after 28 days for samples of fibre-reinforced concrete) for the body, lid, plugs, seal of the lid and the bay must be less than or equal to 350μm.<br>Waterborne body fibre concrete container (after 7 days) - must not show any leakage.<br>The value of the leaching index of the active compound mixtures must be greater than 6. | | | | |
| Activity content | The total inventory of selected radionuclides stored in the repository must not exceed the values given in the table below. | | | | |

| Radionuclide | Limits [Bq ] |
|---|---|
| **C-14** | 2.01E+15 |
| **Ca-41** | 3.78E+14 |
| **Ni-59** | 2.00E+16 |
| **Ni-63** | N |
| **Se-79** | 7.68E+14 |
| **Sr-90** | 6.14E+18 |
| **Mo-93** | 1.80E+15 |
| **Zr-93** | 5.08E+15 |
| **Nb-94** | N |
| **Tc-99** | N |
| **Pd-107** | N |
| **Sn-126** | N |
| **I-129** | 4.58E+11 |
| **Cs-135** | 4.72E+15 |
| **Cs-137** | N |
| **Sm-151** | N |
| **Pu-238** | N |
| **Pu-239** | 1.80E+15 |
| **Am-241** | N |

"N" - the total inventory of these radionuclides is not limited but there are maximum concentrations in the container

| | | | | | |
|---|---|---|---|---|---|
| | The sum of the shares of the currently stored activity of the i-th radionuclide and its limit value (Table) must be less than or equal to one. | | | | |
| | The radionuclide activity in the waste container must not exceed the values given in next table.<br>The maximum mass activity of radionuclides emitting alpha radiation must not exceed 4000 Bq at any point within the VBK.<br>The sum of the actual concentrations of radionuclides in the waste container to the limit value (see Table 2) shall be less than or equal to one, the sum of the concentrations being calculated by:<br>the amount of the ratio $= \sum_{i=1}^{n} \frac{A_i}{LA_i} + \frac{S_a}{L_a \; m} \le 1$<br>where:<br>$S_\alpha$ is the summary activity of radionuclides emitting alpha radiation in fibre concret container in [Bq]<br>m is the mass of treated waste in fibre concret container in [g] $L\alpha$ = 400 Bq/g limits for total alpha mass activity in fibre concrete container<br>Ai - real activity of i-radionuclides in fibre concrete container [Bq/continer.]<br>Lai- limits for activity of i-radionuclides in fibre concrete container from next table [Bq/container.] | | | | |

| Radionuclide | Limit for upper layer [Bq.m-3 ] | Limit for lower and middle layer [Bq.m-3] | for upper layer [Bq/cont] | Limit for lower and middle layer [Bq/cont] |
|---|---|---|---|---|
| C-14 | 1.35E+10 | 9.01E+10 | 4.19E+10 | 2.79E+11 |
| Ca-41 | 1.70E+10 | 1.70E+10 | 5.27E+10 | 5.27E+10 |
| Ni-59 | 7.35E+11 | 8.96E+11 | 2.28E+10 | 2.78E+12 |
| Ni-63 | 1.14E+13 | 3.01E+14 | 3.53E+13 | 9.33E+14 |
| Se-79 | 3.44E+10 | 3.44E+10 | 1.07E+11 | 1.07E+11 |
| Sr-90 | 1.90E+13 | 2.75E+14 | 5.89E+13 | 8.53E+14 |
| Mo-93 | 1.70E+10 | 8.06E+10 | 5.27E+10 | 2.50E+11 |
| Zr-93 | 2.28E+11 | 2.28E+11 | 7.07E+11 | 7.07E+11 |
| Nb-94 | 4.57E+07 | 4.97E+07 | 1.42E+08 | 1.54E+08 |
| Tc-99 | 4.48E+09 | 6.69E+11 | 1.39E+10 | 2.07E+12 |
| Pd-107 | 1.84E+12 | 1.79E+13 | 5.70E+12 | 5.55E+13 |
| Sn-126 | 2.93E+07 | 3.19E+07 | 9.08E+07 | 9.89E+07 |
| I-129 | 1.91E+07 | 1.91E+07 | 5.92E+07 | 5.92E+07 |
| Cs-135 | 1.43E+10 | 2.11E+11 | 4.43E+10 | 6.54E+11 |
| Cs-137 | 1.01E+13 | 1.10E+13 | 3.13E+13 | 3.41E+13 |
| Sm-151 | 1.14E+14 | 1.24E+14 | 3.53E+14 | 3.84E+14 |
| The total mass of alpha radiation of radionuclides: $^{238}Pu$, $^{239}Pu$ , $^{241}Am$ | | | average on container max. 400 Bq/g | average on container max. max. 400Bq/g |

Note: storing of other radionuclides emitting alpha - radiation is not allowed without evidence of their storage in repository by safety analyses and subsequent assessment of Nuclear Regulatory Authority of SR.

| | | | | | |
|---|---|---|---|---|---|
| Radionuclide inventory | Max. inventory is in next table | | | | |

<table>
<tr><td></td><td><table>
<tr><th>Radionuclide</th><th>activity [Bq ]</th></tr>
<tr><td>C-14</td><td>2.01E+15</td></tr>
<tr><td>Ca-41</td><td>3.78E+14</td></tr>
<tr><td>Ni-59</td><td>2.00E+16</td></tr>
<tr><td>Ni-63</td><td>N</td></tr>
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<tr><td>Am-241</td><td>N</td></tr>
</table></td><td></td><td></td><td></td><td></td></tr>
<tr><td>Dose rate</td><td>DP≤ 2 mSv/h</td><td></td><td></td><td></td><td></td></tr>
</table>

| | | | | | |
|---|---|---|---|---|---|
| | DP≤ 2 mSv/h | | | | |
| Surface contamination | surface contamination on the surface of the outer package of the stored materials is less than 0.3 Bq/cm$^2$ for beta emitters and 0.03 Bq/cm$^2$ for alpha emitters | | | | |
| Nuclear criticality | - | | | | |
| Containment | Only PACKAGED RADIOACTIVE WASTE FORMS are accepted for storage:<br>Containers with radioactive waste uniformly dispersed in a compact coupling substance or with RADIOACTIVE WASTES in solid substrates embedded in a cementitious compound into a compact form and a combination thereof.<br>Containers are filled to a minimum of 95% of the internal volume<br>The ratio of the volume of concrete available for radionuclide sorption (fibre concrete volume + cement matrix) to the total volume (including waste) through which the radionuclides pass must be equal to or greater than 0,62 at all times for the entire storage. | | | | |
| Thermal output | - | | | | |
| Radiological gas generation | No, only liquid discharges of the nuclear facility are limited. | | | | |
| Non radiological gas generation | The concentration of hydrogen must not exceed 2% by volume. | | | | |
| Chemical content | The container must not contain free liquids.<br>THE WASTE OF THE CONTAINER must not contain substances that can create conditions under which microbiological decomposition occurs under the development of gases to the extent that it may cause container integrity to be impaired. | | | | |

| | | | | | |
|---|---|---|---|---|---|
| | CONTAINER filling must not contain flammable substances whose flash point is lower than the flash point of flammable liquids listed in I and II. Classes according to Decree of the Ministry of the Interior No. 86/1999 in the original form or in contact with water, an explosive or exothermic reaction of this substance can be expected to such an extent that the integrity of container with radioactive waste is not impaired.<br><br>CONTAINER filling must not contain dangerous waste except radioactive waste.<br><br>The container must not contain wastes which may only or in contact with water, generate gases, smoke and flammable vapors.<br><br>THE CONTAINER filling shall not contain chelating or complexing agents in an amount greater than 0.1% by weight. | | | | |
| | CONTAINER filling must not contain dangeroud waste except radioactive waste.<br><br>The container must not contain wastes which may only or in contact with water, generate gases, smoke and flammable vapors.<br><br>THE CONTAINER filling shall not contain chelating or complexing agents in an amount greater than 0.1% by weight. | | | | |
| | The container must not contain free liquids.<br><br>Note: Liquid is considered to be free liquid on the surface of the cement matrix, which cannot be ruled out if container wall leaks. Liquids that are in the matrix are fixed, respectively. Absorbent, or adherently bonded to the surface of the filler material, is not among the free liquids. | | | | |

| | | | | | |
|---|---|---|---|---|---|
| Chemical durability | It is necessary to ensure the long-term preservation of the properties of the waste container even when it is loaded. Ensure safe handling of the container under normal conditions and under foreseeable abnormal conditions.<br>RAO can only be deposited in fibre-reinforced concrete containers, the properties of which are in accordance with the specification, technical conditions for their production. | | | | |
| Volume of voids | - | | | | |
| Stackability | The storage facility comprises an array of storage boxes, arranged in rows and double rows. The first double row is shielded by a steel hall. The Boxes are made of steel-reinforced concrete, measuring 18 × 6 × 5.5 m, the walls are 600 mm thick. In stage 1, two double rows, i.e. 80 storage boxes were constructed. Ninety fibre-reinforced concrete containers can be fitted into one box with each of the containers measuring 1.7 × 1.7 × 17 m, the walls of the containers being 0.1 m thick. The total capacity is 7,200 containers with a total volume of 22,320 $m^3$.<br>The weight of the waste containers in the expansion unit (4 boxes) of one row must not exceed 3600 t. | | | | |
| Impact performance | - | | | | |
| Fire performance | CONTAINER filling must not contain flammable substances whose flash point is lower than the flash point of flammable liquids listed in I and II. classes according to Decree of the Ministry of Interior of the Slovak Republic No. 86/1999 Z.z. and substances which, in their original form or in contact with water, may be expected to have an explosive or exothermic reaction of this substance to such an extent that the integrity of container with radioactive waste is not impaired. | | | | |
| Identification | Described above. | | | | |
| Quality control | Quality control assurance is based on each condition within the limits and conditions for the repository, and fulfill the requirement is a secured the QC. | | | | |

| | | | | | |
|---|---|---|---|---|---|
| Quality assurance | - | | | | |
| Data requirements | - | | | | |

## 3.7 Switzerland

### 3.7.1 Preliminary Waste Acceptance Criteria (extraction), requirements on waste package properties

#### 3.7.1.1.1 Purpose

The properties of waste packages for disposal have to meet various criteria in order to ensure that, considering the design of the repository, the waste emplacement process is technically feasible and that the necessary level of safety can be assured during the operational and post-operational phases (protection of personnel, population and the environment).

#### 3.7.1.1.2 Principles for the design and content of waste packages

1 Minimisation of organic material in the conditioned waste: minimisation of organics in raw waste, abandonment of organic or organic-containing additives in the waste conditioning if possible at the state of science and technology (SST).

*Purpose:* Minimisation of radionuclide mobilisation via complexation in the closed deep geological repository (DGR), minimisation of gas generation and gas pressure build-up in the closed DGR, minimisation of fire load during operation.

2 Minimisation of mass and available surface of inorganic material in the conditioned waste, which can generate gases when in contact with underground water (e.g. Zn, Al, Fe, steel), and if possible considering SST: Avoidance of corresponding raw waste; chemical reaction (oxidation) of corresponding raw waste, abandonment of corresponding additives in the waste conditioning.

*Purpose:* Minimisation of gas generation and gas pressure build-up in the closed DGR.

3 Minimisation of secondary waste production in the waste conditioning

*Purpose:* Minimisation of conditioned waste amounts

4 Stabilisation of gaseous, liquid of solid dispersed waste or intermediate products by transforming into forms which are difficult to disperse (gastight enclosure, immobilisation on sorbents, or solidification via the production of waste/additive matrices)

*Purpose:* Minimisation of radionuclide release during the operational and post-operational phases of the DGR

5 Protection of heat sensitive and combustible waste against thermal impact: thermal insulation by non-combustible packaging and avoidance of thermal bridges if technically feasible.

*Purpose:* Minimisation of radionuclide release upon accidents with thermal impact (fire).

6 Well-balanced mass distributions in waste packages that remains balanced at transport, operation and stacking.

*Purpose:* safe transport and handling of waste packages.

7 Resistance (chemical transformation, dissolution) of waste packages and their components against aqueous media in the DGR after closure.

*Purpose:* barrier function of the waste packages and their components (retarded resp. slow radionuclide release after repository resaturation).

### 3.7.2 Further qualitative and quantitative criteria with relevance for thermally treated products

Several qualitative and quantitative criteria aim to ensure a safe handling and storage of the radioactive waste. In terms of thermally treated waste products, the most important criteria are as follows:

- Resistance against aqueous media
- Non-ignitable/combustible or hard to ignite/combust
- Non-disperse, mechanically stable
- Pressure resistance of min. 10 MPa

## 3.8 United Kingdom

The UK is now examining options to thermally treat Higher Activity Wastes (HAW)/Intermediate Level Waste (ILW) with the intention of disposing to the ILW repository.

In the absence of an established repository, Radioactive Waste Management Ltd (RWM) operates a Letter of Compliance (LoC) procedure which allows waste owners to develop and implement packaging solutions for their radioactive wastes. The majority of ILW treated to date has been encapsulated in a cementitious medium although conceptual letters of compliance have been submitted for vitreous and ceramic materials. The needs for compliance are not material related and as such no specific requirements have been placed on vitreous or thermally treated products. The exception are studies currently being undertaken which are examining the possible effect of a vitreous product on the efficacy of the cementitious backfill to fulfil its requirements of preventing or retarding radionuclide transport within the repository.

Submissions for Letter of Compliance will require among other things to demonstrate the following. These are not waste acceptance criteria per se, any submission will need to show why the packaged waste product meets these requirements. Such compliance is adjudicated by RWM following submission and will award a letter of compliance if it deems that requirements have been met.

The requirements of encapsulated products are covered by 6 attributes:

- Physical immobilisation
- Mechanical and Physical properties
- Chemical containment
- Hazardous materials
- Gas generation
- Wasteform evolution

### 3.8.1 Physical Immobilisation:

The wasteform shall be designed to immobilise radionuclides and other hazardous materials so as to make an appropriate contribution to waste package performance during all stages of long-term management.

All reasonable measures shall be taken to ensure that radionuclides and hazardous materials in the wastes are immobilised and that loose particulate material is minimised

All reasonable measures shall be taken to ensure that in the event of an impact accident, the quantity of potentially mobile radionuclides present within the wastes package, including those generates as a result of the impact accident, is commensurate with the waste package meeting the impact performance requirement defined by the relevant waste package specification.

All reasonable measures shall be taken to ensure that, in the event of a fire accident the quantity of potentially mobile radionuclides present within the waste package including those generated as a result of the fire accident is commensurate with the waste package meeting the impact accident performance requirements as defined by the relevant waste package specification.

All reasonable measures shall be taken to exclude free liquids from the wasteform. This should include materials that may degrade to generate liquids. Free liquids not removed from the waste prior to waste packaging should be immobilised by a suitable waste conditioning process.

### 3.8.2 Mechanical and Physical properties:

The wasteform shall be designed to provide the mechanical and physical properties necessary to ensure appropriate performance of the waste package during all stages of long term management.

The development and production of the wasteform should ensure that the volume of voidage within the waste package (such as ullage, holes or other spaces) is appropriately minimised.

The wasteform shall be sufficiently permeable to allow gases generated within the wasteform to be released without compromising the ability of the waste package to meet any aspect of the relevant waste package specification. The mass transport properties of the wasteform (e.g. diffusivity and permeability) shall provide best practicable means for the containment of water-soluble radionuclides within the waste package.

Local concentrations of materials within the wasteform that may compromise the ability of the waste package to meet any aspect of the relevant waste package specification should be minimised.

The thermal conductivity of the wasteform shall be sufficient to dissipate any heat generated within the waste package when emplaced in a GDF without unacceptable temperature rise.

### 3.8.3 Chemical Containment:

The wasteform shall not be incompatible with the chemical containment of radionuclides and hazardous materials as embodied in the requirements of a GDF. Where they may affect chemical containment, the following items should not be introduced through waste conditioning or packaging, and their presence in wastes should be minimised wherever possible.

- Oxidising agents
- Acids and or materials that degrade to acids
- Cellulose and other organic materials
- Complexants and chelating agents, and or materials that degrade to such compounds
- Non aqueous phase liquids and or materials that generate them
- Any other materials that could detrimentally affect chemical containment

### 3.8.4 Hazardous materials:

The wasteform shall not contain any hazardous materials or have the potential to generate such materials unless the treatment or packaging of such materials makes them safe. The means by which any of these materials is made safe shall be demonstrate for all relevant periods of long term management.

### 3.8.5 Gas generation

Gases generated by the wasteform shall not compromise the ability of the waste package to meet any aspect of the relevant waste package specification.

### 3.8.6 Wasteform evolution

Changes in the characteristics of the wasteform as it evolves shall not result in degradation that will compromise the ability of the waste package to meet any aspect of the relevant waste package specification. The deleterious effect of the following processes should be considered.

- Dimensional changes, e.g. shrinkage
- Corrosion including, but not limited to, the production of gases and particulate material, and wasteform expansion resulting from the formation of lower density solid corrosions products.
- Microbial activity.
- Self-irradiation and irradiation by surrounding waste packages
- Heat generation by the waste form and its surroundings including, but not limited to localised heat sources within the wasteform, the effects on the curing of the encapsulant material and the consequential effects on longer term performance.

### 3.8.7 Criticality Safety

The presence of fissile material, neutron moderators and reflectors in the waste package shall be controlled to ensure that

- Criticality during transport is prevented.
- The risk of criticality during GDF operational phase is tolerable and as low as reasonably practicable and in the GDF post closure period both the likelihood and the consequences of criticality are low.

# 4 Development of generic criteria

## 4.1 Introduction

Section 3 collated WAC and other disposability requirements in countries participating in the THERAMIN project. In Section 4, the national criteria are used to derive generic criteria that can be applied to evaluate the disposability of thermally treated wastes.

### 4.1.1 Objectives

Disposability criteria identify the characteristics required in a waste product in order to ensure that the waste cannot have a significant detrimental impact on the long-term safety provided by a disposal facility. They also provide a measure of the ‘quality’ of a waste product and, if applied sufficiently broadly, give a basis for consistent comparison between the products generated from the treatment / conditioning of different wastes and/or via different routes.

The objective of the work reported here is to develop a set of ‘generic disposability criteria’ that can be used to evaluate any products from any form of thermal treatment for disposal at any type of facility, and regardless of the political, regulatory or socio-economic context. Generic disposability criteria are defined here as: “*Factors affecting the disposability of conditioned waste produced from application of some form of thermal treatment*”.

The generic disposability criteria discussed below have been developed for use in THERAMIN and beyond. The intention is that they will provide a well-underpinned starting point for WMOs to develop their own disposability criteria relating to thermally treated wastes, tailored to a particular context, such that users can have confidence that they are taking account of all relevant factors. However, it is not mandatory to apply these criteria. They will be applied later in WP4, under task 4.3 (Downstream / safety case implications).

### 4.1.2 Approach and Scope

Generic disposability criteria have been developed based on examination of the criteria applicable to the disposability of thermally treated waste in each THERAMIN participating country, as set out in Section 3, and taking account of the national waste management strategies (i.e. the national context) within which these criteria are applied, as summarised in Section 2. They also reflect typical characteristics of thermally treated waste products, many of which underpin a decision to implement thermal treatment. These include:

- Reduced volume (at least of the raw waste – conditioning / packaging of the waste will increase the volume to some extent, which may or may not result in a smaller total packaged waste volume compared to other treatment / conditioning routes).
- Activity / fissile material may be concentrated into a smaller volume.
- Void spaces within waste packages may be reduced.
- Many waste constituents are oxidised / combusted and end up in an unreactive form (e.g. an ash, char or solid monolithic wasteform where they are combined with an unreactive matrix).
- Unreactive waste constituents are often (although not always) encapsulated in a vitreous waste matrix.

- Volatile species are driven off, chemically reactive species are destroyed, and many potential gas generating species are 'pre-reacted' such that they will not undergo further gas generating reactions in the repository.
- Thermal treatment may generate different types / quantities of secondary waste / off-gases / liquid effluents compared to other treatment / conditioning routes, which need to be factored into planning for their application.

The scope of this sub-task (and wider WP4 activities) is bounded by focusing on the nature of the product from thermal treatment, i.e. the treated / conditioned wasteform. Raw waste characteristics are not considered.

Also, as noted above, the intention is to develop criteria that can be used to evaluate the disposability of *any* products from *any* form of thermal treatment at *any* type of disposal facility, i.e. criteria that are wholly 'generic'. This means that the criteria are not linked to waste product characteristics associated with a particular treatment technology or processing route. It is noted, however, that some products of thermal treatment are likely to require additional processing before they are in a passively safe form that is suitable for disposal and, as such, may not conform with certain generic disposability criteria.

The generic disposability criteria discussed here are intended to be applicable to any packaging or disposal concept, regardless of the nature of engineered barriers that are present, and in any disposal environment, regardless of its characteristics and the nature of the host rock / geology. An emphasis is placed on long-term safety in a geological repository, in line with the scope of the THERAMIN description of work. However, consideration of surface / near-surface disposal is also necessary, given the breadth and scope of national criteria provided in Section 3 (many of which apply to facilities that are at, or close to, the surface).

An important distinction is drawn between:

- 'Guidance' or 'criteria', which tend to be general, and not related to requirements or actions associated with any particular disposal facility. Associated terminology generally indicates that it is not mandatory to follow guidance, through the use of words such as 'could', 'should', etc.
- 'Waste Acceptance Criteria' (WAC), which tend to be specific to the acceptance of waste at a particular disposal facility. Associated terminology tends to indicate a stronger requirement to comply, through the use of words such as 'shall', 'must', etc.

The national criteria provided in Section 3 include both WAC and more general guidance.

## 4.2 General observations on national criteria inputs

This section provides general observations on the national criteria inputs provided by THERAMIN participating countries in Section 3, to provide context to subsequent discussion and development of generic disposability criteria.

Inputs were received from each of the eight countries participating in THERAMIN: Belgium, Finland, France, Germany, Lithuania, Slovakia, Switzerland and the United Kingdom. Five countries (Belgium, France, Germany, Lithuania and Slovakia) structured their inputs using a table which lists different categories of relevance when evaluating disposability. The remaining three countries (Finland, Switzerland and the United Kingdom) provided a more general discussion of requirements relating to waste package design and acceptance of manufactured waste packages for disposal.

National criteria inputs have been provided for various types of disposal facility, including surface disposal (in the case of responses from Belgium and Slovakia), near-surface disposal (Finland) and geological disposal (France, Germany, Lithuania, Switzerland, United Kingdom). The importance of certain criteria may vary depending on the depth of disposal and the safety functions applicable to the wasteform in different disposal contexts. This point is discussed further later on in Section 4.

Some national criteria have been provided that refer to generic plans for disposal (particularly in the cases of geological disposal in countries that have not yet identified a proposed site or host rock in which to construct a repository). In such cases, the criteria provided are preliminary / provisional, and will be developed further as plans for the disposal facility progress. Other national criteria are site-specific and reflect formal WAC associated with an existing disposal facility (or one that is in the advanced stages of planning).

For the most part, the national criteria inputs focus on the disposability of LL-LILW, although some countries have provided inputs relating to SL-LILW management (Belgium) or inputs that apply to both LL-LILW and HLW/spent fuel disposal (United Kingdom). The packaging approach(es) applicable to the different waste types (and reflected in the national criteria) depend on the disposal concepts that have been adopted for the relevant disposal facilities in each country, but generally involve disposal in concrete, steel or cast-iron containers.

It is particularly important to note that the national criteria inputs provided are not specific to the disposal of thermally treated wastes; they tend to be more generally applicable to all wastes of a certain classification or destined for a certain disposal route. This appears to reflect the status of disposability criteria in participating countries and underpins the need for explicit consideration of criteria for thermally treated wastes, which is a key driver for the present task.

Finally, it appears that the national criteria inputs provided are not complete, and do not reflect all applicable WAC for radioactive waste disposal in the participating countries. This may reflect the scope of wastes that are being considered for thermal treatment in these countries. For example, Finland is not considering thermal treatment of spent fuel prior to disposal, so no WAC associated with planned disposal at the ONKALO spent nuclear fuel repository have been provided.

Table 3 summarises the scope of national criteria inputs provided. From this table it can clearly be seen that the scope and context of the national criteria inputs received varies considerably from country to country. The criteria themselves are heavily dependent on this context (e.g. waste classifications, packaging concept, current status of radioactive waste management / disposal operations, type of disposal facility…). This means that as defined, the criteria may not be directly comparable, or transferable to other disposal situations. In Section 4.3, we draw out those factors that are more broadly applicable and those that are of particular relevance for thermally treated waste.

This is achieved by defining generic disposability criteria at a relatively high level, and by setting out qualitative, rather than quantitative, metrics against which disposability can be assessed. Numerical requirements identified within national criteria tend to be strongly linked to the national context (e.g. activity limits for different waste classifications), so have limited transferability for wider use outside the country of origin.

As noted in the objectives (Section 4.1.1), the generic disposability criteria are intended to be a starting point for WMOs to tailor their own national disposability criteria relating to thermally treated waste, which give confidence that relevant factors are being considered.

| Country | Form & scope of response | Type of disposal | Packaging concept | Formal WAC? |
|---|---|---|---|---|
| Belgium | Table covering SL-LILW disposal at Dessel in north-east Belgium | Surface | Storage packages grouted in concrete monoliths | No – preliminary – formal WAC will be finalised once a licence for disposal is obtained |
| Finland | Text response based on LILW (reactor operating waste) disposal | Near-surface (Olkiluoto VLJ); intermediate-depth (Loviisa VLJ) | Concrete boxes | Yes |
| France | Table covering LL-LILW disposal in clay | Geological (in a clay host rock) | Storage packages grouted in concrete boxes / direct disposal | Yes |
| Germany | Table, plus additional context, discussing LL-LILW disposal at Konrad in northern Germany | Geological (host rock to be determined) | Cast iron / steel / concrete boxes | Yes |
| Lithuania | Table covering planned disposal of LILW | Geological (host rock to be determined) | Cement-grouted metallic waste | No – preliminary WAC |
| Slovakia | Table, plus additional context, discussing LLW disposal at Mochovce | Surface | Fibre-reinforced concrete packages | Yes |
| Switzerland | Text response based on disposal in clay (LL-LILW and spent fuel) | Geological (clay) | Steel canisters / storage drums in concrete boxes | No – preliminary WAC |
| United Kingdom | Text response based on disposal of ILW (also applies to HLW/SF disposal) | Geological (salt / clay / crystalline – host rock to be determined) | Waste grouted in steel / cast iron / concrete containers | Not WAC, but formal requirements – specifications to be met for issue of a Letter of Compliance (LoC) |

*Table 3. Summary of the scope of national disposability criteria inputs*

## 4.3 Generic criteria development

### 4.3.1 Comparison of national criteria and proposal of generic disposability criteria

Discussion in this section considers the national criteria inputs received from different countries under each of the categories within the table template, as recorded in Section 3. Requirements are summarised and any common criteria are flagged. Generic disposability criteria for thermally treated wastes are then proposed, drawing on this comparison exercise together with expert judgement of factors relevant to thermal treatment and the typical characteristics of products generated via thermal treatment / conditioning.

The following headings / categories from the Section 3 table are discussed in turn:

- Dimensions / mass of packages
- Provisions for transport, handling and emplacement
- Package integrity and required lifetime
- Activity content
- Radionuclide inventory
- Dose rate limits
- Surface contamination
- Nuclear criticality
- Thermal output
- Gas generation
- Chemical content
- Chemical durability
- Voids
- Waste package stacking
- Waste package impact performance
- Waste package fire performance
- ID / labelling
- QA / QC requirements
- Data management

#### *4.3.1.1 Dimensions / mass of packages*

Standard waste containers with fixed dimensions and masses are specified in many national programmes, with their design depending on typical waste characteristics, package handling and transport capabilities, and the design of storage and disposal facilities.

It is not appropriate to define values of these properties for general application, since specific values already apply in different countries, and vary considerably. It may be possible for national waste management programmes to adapt existing container designs to take thermally treated waste. However, the following considerations could potentially affect the dimensions and mass of waste containers that it is appropriate to use for thermally treated wastes (and might require new package designs to be developed):

- Thermal treatment can concentrate activity and/or fissile material into a smaller volume than that of the raw waste. This may limit the quantity of thermally treated waste that can be placed in any one container, thereby promoting the use of smaller containers. It is noted that under certain circumstances, e.g. where the inventory of volatile radionuclides is high, thermal treatment could conceivably generate a waste product with reduced total activity (although non-volatile species could still be concentrated into a smaller volume)
- The integrity / robustness of the container that is required to ensure safety during transport, handling operations and in the post-closure period will depend, to some extent, on the characteristics of the wasteform produced via thermal treatment (and the

safety functions applicable to the wasteform and the package). A more robust container may need to be used to package wasteforms that are potentially more mobile / less durable, e.g. unencapsulated ashes or chars (if indeed these can be accepted for disposal without further conditioning – this is discussed further in Section 4.3.1.3).

- The choice of container may be influenced by the thermal treatment processing route. For example, if waste is to be transferred directly to the disposal container following melting / pyrolysis, then the container dimensions need to be compatible with those of the processing equipment and the container must be robust enough to withstand exposure to high temperatures without loss of integrity. Some form of refractory lining may be required. On the other hand, if the disposal container forms an overpack around an inner vessel incorporating the treated waste (e.g. to ensure no contamination of the outer surface, to ensure a regular shape, or to ensure no other sub-optimal characteristics resulting from exposure to high temperatures), then the container dimensions should be designed to provide a snug fit around the inner vessel, to maximise packing efficiency. An exception to this would arise if it is desired to place some form of annulus material, such as cement or glass, between the inner and outer containers as an additional barrier to release or recovery of the waste.
- The low voidage typically associated with thermally treated waste (see Section 4.3.1.13) means it is relatively dense. This could limit the size of waste packages that it is practicable to handle.

Proposed generic disposability criteria reflect these considerations.

**Proposed generic disposability criteria:** *The dimensions and mass of containers used to package thermally treatment waste (and other aspects of the container design) should be compatible with the thermal processing route being employed.*

*The dimensions and mass of containers used to package thermally treated waste (and other aspects of the container design) should be compatible with relevant safety functions for storage and disposal, and with all applicable constraints on waste classification, handling, transport and disposal, taking account of the processed waste characteristics.*

#### *4.3.1.2 Provisions for transport, handling and emplacement*

National criteria inputs generally specify a requirement that waste package / container integrity must not be compromised during transport, handling or emplacement. This requirement applies for packages of thermally treated waste, as for all other waste packages.

In demonstrating that this requirement is met, waste packagers should take account of the different characteristics of thermally treated waste (e.g. potentially different susceptibility in the event of an impact accident) and should be able to demonstrate that any new design of waste package required for thermally treated waste can be safely handled, transported and emplaced, once it has been produced and filled.

**Proposed generic disposability criterion:** *Waste packages containing thermally treated waste should comply with all applicable criteria relating to the transport, handling and emplacement of other waste packages. The characteristics of thermally treated waste should be considered as part of demonstrating compliance with existing requirements on transport, handling and emplacement.*

#### 4.3.1.3 Package integrity and required lifetime

Requirements on the integrity and lifetime of packages containing thermally treated waste will depend on the safety functions to be provided by the wasteform and the disposal container (which together make up the waste package). These are specific to a particular disposal facility and the associated safety case. Generic disposability criteria therefore cannot easily be defined in association with this category.

Safety functions applicable to thermally treated wastes may or may not be the same as those applicable to other wastes to be disposed of at the same repository. This depends on the drivers for implementing thermal treatment in a particular context – if thermal treatment is implemented to enhance the robustness and durability of the wasteform, such that it can be relied upon to provide containment of radionuclides for long timescales, then one would expect to see this reflected in associated safety functions. On the other hand, if thermal treatment is driven by factors such as waste volume reduction, then there may be no difference in the safety functions associated with the resulting wasteform.

From a disposability perspective, a solid, passively safe wasteform is required, which incorporates radionuclides in such a way that they are not easily released during transport, operations or following disposal. A monolithic block is generally preferred over a powdered wasteform, although disposal of non-encapsulated waste has been considered by e.g. RWM for certain wastes and materials – see, for example, NDA (2009) and Neall (2016). Some thermal treatments (e.g. pyrolysis) do not produce a monolithic block of conditioned waste. Depending on the acceptability of the potentially more mobile wasteforms that they produce, it could be necessary to conduct additional processing steps to condition the waste into a more disposable form (e.g. via compaction, cementation or vitrification).

**Proposed generic disposability criterion:** *Package integrity requirements for the applicable disposal route should be applied to thermally treated waste. Any additional criteria on package integrity defined for thermally treated waste should be linked to safety functions applied to such waste.*

#### 4.3.1.4 Activity content

National criteria in THERAMIN countries include limits on the activity, activity concentration, concentration of alpha emitters and beta-gamma emitters, and activity levels at the surface of the wasteform. Specific activity limits are often defined for particular radionuclides that are of concern for safety (e.g. fissile, volatile, mobile or chemotoxic radionuclides, or species that can act as a moderator, thereby promoting the risk of criticality occurring). Where specific criteria relating to the activity content are not explicitly defined, the activity content of waste is limited indirectly by other criteria such as dose rate limits, heat output limits, fissile material limits and material release fractions in the event of a dropped package.

Criteria related to the activity content of waste to be disposed of at a particular facility are linked to the acceptable waste classification(s) for the facility in question and, hence, to the relevant national context (e.g. the regulatory basis for waste classifications). It is therefore not appropriate to define limits on activity content for general application, since specific values already apply in different countries, and vary considerably. However, compliance with existing criteria should consider the potential for activity to become concentrated in a smaller volume during thermal treatment and, hence, for the product of thermal treatment to potentially require additional shielding during handling or even reclassification into a higher activity waste category than the parent raw waste. Several of the national inputs to this report remark that thermally treated waste is assumed to be ILW, rather than LLW, for this reason. The appropriate classification of the waste product would need to be confirmed through representative waste processing and characterisation tests.

**Proposed generic disposability criterion:** *Existing limits on the activity content of waste proposed for a particular disposal route should be applied to thermally treated waste. No additional criteria on activity content for thermally treated waste are considered necessary. However, demonstrating compliance with existing criteria should take account of the potential for activity to become concentrated in a smaller volume during thermal treatment. Associated implications for waste classification and waste package handling should be considered.*

#### *4.3.1.5 Radionuclide inventory*

National criteria typically require the activity or concentration of specific radionuclides present in a waste stream to be declared. The radionuclides of interest vary between countries and between disposal facilities and are linked to the potential risk that different species pose to transport, operational and post-closure safety. As with activity limits discussed in Section 4.3.1.5, it is not appropriate to define generic disposability criteria relating to the radionuclide inventory of wastes, since specific requirements already apply, and vary considerably depending on the disposal context.

If thermal treatment is implemented in order to generate a wasteform with enhanced durability, then it is appropriate to consider the chemical compatibility of radionuclide species that are present with the proposed immobilisation matrix. For example, the solubility of species present in the waste within a glass matrix, or their ease of incorporation into ceramic phases of interest, will affect how well these radionuclides are immobilised over the long term.

**Proposed generic disposability criteria:** *Existing requirements applicable to declaration of the radionuclide inventory of a waste stream should be applied to thermally treated waste.*

*The choice of thermal processing route and wasteform morphology / formulation should be tailored to the radionuclide inventory (and other characteristics) of the waste, particularly if thermal treatment is driven by the need to produce a durable, long-lived wasteform.*

#### *4.3.1.6 Dose rate limits*

Many national criteria (including those provided by Belgium, France, German, Slovakia and the United Kingdom), include limits on the external dose rate applicable on contact with (or at some specified distance from) waste packages for radiation protection purposes. As with activity limits discussed in Section 4.3.1.5, it is not appropriate to define generic disposability criteria relating to dose rate limits, since specific requirements already apply, and vary considerably depending on the disposal context.

However (also as with activity limits), compliance with existing criteria should consider the potential for activity to become concentrated in a smaller volume during thermal treatment and, hence, for there to be higher dose rates associated with packages of thermally treated waste.

**Proposed generic disposability criterion:** *Existing waste package dose rate limits for a particular disposal route should be applied to thermally treated waste. No additional criteria on* *dose rate limits for thermally treated waste are considered necessary. However, demonstrating compliance with existing criteria should take account of the potential for activity to become concentrated in a smaller volume during thermal treatment.*

#### *4.3.1.7 Surface contamination*

Contamination of the outer surface of disposal containers with radioactivity could affect work safety during handling operations. Non-radioactive contamination with e.g. grease, water or salts as a result of improper handling or storage could potentially accelerate corrosion of the container and reduce its lifetime.

Similar requirements should be applied to minimise surface contamination of thermally treated waste packages (and to ensure appropriate handling) as apply during production of waste packages via other conditioning routes.

The absence of radioactive contamination on the outer surface of waste packages would need to be confirmed following completion of conditioning and packaging steps, particularly for waste that is to be contact handled during storage and disposal. However, this is, in any case, generally required, so should be reflected in existing WAC for a disposal facility.

There may be different mechanisms for surface contamination of waste containers to occur in association with thermal treatment. For example, molten residues or drips could potentially be deposited on the outer surface of containers. This is particularly relevant for processing routes where the disposal container is filled directly with the product from thermal treatment, as opposed to placing the filled vessel in some form of overpack prior to disposal.

**Proposed generic disposability criterion:** *Requirements associated with minimising and identifying contamination of the outer surface of waste containers for the applicable disposal route should be applied to thermally treated waste. No additional criteria on surface contamination for thermally treated waste are considered necessary. However, ensuring compliance with existing criteria should account for potential contamination mechanisms that are specific to the thermal treatment route employed.*

#### *4.3.1.8 Nuclear criticality*

Where waste streams contain significant amounts of fissile material, national criteria set out limits in order to minimise or avoid risks associated with criticality. These include mass limits on fissile material, fertile material and potential moderators (including water). In some cases (e.g. France), national criteria also indicate that repository designs take account of scenarios for fissile material accumulation as part of their layout and waste package emplacement arrangements (e.g. waste package stacking).

There is no reason to apply different fissile mass limits to thermally treated wastes, so existing criteria applicable to other conditioned waste are considered sufficient. However, as with previous criteria (e.g. activity limits and dose rate limits), it is important to take account of the potential for fissile material to become concentrated into a smaller volume during thermal treatment, which might increase criticality risks. Certainly, this factor needs to be considered during thermal treatment itself and may limit the scale at which a thermal treatment processing route can be applied for certain wastes.

**Proposed generic disposability criterion:** *Existing criteria designed to minimise the likelihood and/or consequences of a nuclear criticality event occurring should be applied to thermally treated waste. No additional criteria on criticality safety for thermally treated waste are considered necessary. However, the potential impacts of fissile material concentration on transport, operational and post-closure safety should be considered.*

#### *4.3.1.9 Thermal output*

A thermally treated wasteform is unlikely to be susceptible to heat output from the waste contained within it, having already been subjected to high temperatures during its production. It is therefore not necessary to define thermal limits to protect the integrity of the wasteform itself.

The thermal output of the waste will instead be limited by what can be tolerated without a detrimental impact on the surrounding engineered and natural barriers of the disposal system. This will depend upon the barriers that are present and their functions. A maximum thermal power for different waste classifications is often defined (e.g. by France, Germany and the United Kingdom). For wastes that generate considerable amounts of heat (e.g. HLW and spent fuel), it may be necessary to limit the amount of waste in any one waste package, or to space packages some distance apart from each other.

Such requirements are less likely to be needed for ILW and LLW, although again, it should be borne in mind that thermal treatment could concentrate activity into a smaller volume of waste, thereby increasing the thermal output from a given waste volume.

**Proposed generic disposability criterion:** *The thermal output of thermally treated waste should not have a detrimental impact on performance of the engineered and natural barriers that make up the disposal system, taking account of the potential for activity to be concentrated during thermal treatment.*

#### *4.3.1.10 Gas generation*

National criteria inputs generally include requirements that gas generation as a result of e.g. corrosion, radiolysis and degradation of organic material should not have a detrimental impact on the waste package, or on other engineered or natural barriers within the disposal system. In addition, some national programmes (e.g. France) define specific limits on the amount of radioactive gas that can be released from individual waste packages (this links to the discussion of radioactive inventory in Section 4.3.1.5).

Gas generation (both radiological and non-radiological) from thermally treated waste should be relatively low. Many raw waste constituents will decompose during thermal treatment and volatile species / water will be lost. One notable exception is that metallic waste constituents such as iron and steel are unlikely to react (although they may melt), so will often be present in the thermally treated product. These could potentially generate significant quantities of gas (hydrogen) over the long term in a disposal facility as they corrode, which could be of concern for repository performance, particularly in a clay / salt host rock where pressurisation may be an issue. However, if such materials are present in wastes requiring disposal, the approach to dispose of them safely without affecting the long-term performance of the repository will, in any case, need to be considered (regardless of whether thermal treatment is employed). It is therefore considered that thermally treated waste should comply with existing criteria for gas generation (which will be specific to a particular national context / disposal facility). Additional / alternative generic disposability criteria are not needed.

**Proposed generic disposability criterion:** *Existing gas generation requirements for the applicable disposal route should be applied to thermally treated waste. No additional criteria on gas generation for thermally treated waste are considered necessary.*

#### *4.3.1.11 Chemical content*

A stable, passively safe wasteform is generally required for disposal. In addition to criteria relating to the radiological content, fissile material content, potential for gas generation and morphology of the wasteform, all of which are described above, national criteria inputs typically require one or more of the following waste constituents to be minimised or avoided:

- Flammables / explosives / pyrophoric materials.
- Free liquids (organic and inorganic).
- Biological / infectious materials or materials that would promote enhanced microbiological gas generation (which could lead to enhanced barrier degradation / gas generation).
- Toxic materials, such as asbestos and chemotoxic elements.
- Compressed gases / sealed containers.
- Complexing agents (“complexants”).

Some countries (e.g. Germany) also require the chemical speciation of radionuclides present in the waste to be declared. Limits may also be imposed on the amount of organic or combustible material that is present.

Most of the above species will be destroyed or will boil off during thermal treatment; this is one of the key benefits of thermal treatment. The product from thermal treatment generally has a low chemical reactivity and is much less likely to undergo detrimental chemical interactions in a repository than wastes that are simply encapsulated in a cement, bitumen or polymer matrix. As such, it is considered that WAC applicable to the chemical content of wastes that have not been thermally treated would easily be met by the products from thermal treatment, and it is not necessary to define any additional criteria for thermally treated waste under this category.

It is also important to consider the chemical compatibility of thermally treated wasteforms with other components of the disposal system, including other wastes and other engineered barriers. An important interaction that has been studied in the United Kingdom is the potential for detrimental interactions between silica-rich wasteforms (e.g. vitrified ILW) and cementitious buffer or backfill materials. Dissolution rates for the former can be enhanced under high-pH conditions. Silica-rich porewater can disrupt the calcium-silicate ratio of cements, thereby affecting their durability and ability to buffer pH to high values.

**Proposed generic disposability criterion:** *Chemical content WAC for the applicable disposal route should be applied to thermally treated waste (and are expected to be easily met).*

*The choice of thermal treatment route and the design of the associated disposal facility should ensure the chemical compatibility of thermally treated waste with other disposal system components.*

#### 4.3.1.12 Chemical durability

Some national criteria inputs (e.g. Germany) specify a minimum lifetime over which disposed wasteforms should be stable. Others (e.g. Belgium) specify particular reactions that the wasteform should be resilient to (e.g. alkali-silica reactions, as discussed in Section 4.3.1.11, or delayed ettringite formation). No evidence has been provided of a maximum dissolution rate being specified as a criterion for waste acceptance.

As discussed in Section 4.3.1.3, the emphasis placed on the long-term integrity of the wasteform (including its chemical durability) could vary considerably, depending on the nature of the waste (e.g. its radiological / chemical inventory, fissile material content and how long it will present a hazard for), as well as the safety functions placed on the wasteform, i.e. the extent to which it is expected to provide a barrier to radionuclide release in the long term (which may reflect the drivers for selecting thermal treatment / conditioning).

Short-lived wastes may not need to be immobilised in a highly durable wasteform in order to be safely disposed of, so one would expect relatively few criteria relating to wasteform 'quality' or longevity. On the other hand, wastes that pose a long-term hazard (e.g. due to the presence of long-lived radionuclides or fissile material) may benefit from the enhanced engineered containment that could potentially be provided via certain thermal treatment routes (e.g. vitrification or ceramification) compared to more conventional encapsulation routes such as grouting in cement. There is generally an expectation that wasteforms for HLW, spent fuel and nuclear materials such as plutonium will provide a safety function by limiting the release of radionuclides on contact with groundwater. However, this is not always the case for LILW wasteforms. For example, in the United Kingdom, no explicit credit is currently taken for containment provided by (cementitious) ILW wasteforms in post-closure safety assessments for geological disposal.

If the longevity of the wasteform is an important contributor to the safety provided by the disposal system and, hence, the ability to make a post-closure safety case, then it seems reasonable to specify WAC relating to the longevity of the wasteform and its ability to immobilise species of concern. However, there are many other reasons why one might choose to thermally treat waste (as discussed in Section 4.1.2), which do not relate to enhanced long-term safety. Given that the applicable drivers will vary depending on the specific disposal context, a generic disposability criterion relating to the chemical durability of thermally treated waste is not proposed.

If criteria relating to the durability of a thermally treated wasteform are deemed to be required for application in a particular context, then it is recommended that these should be linked to a required containment lifetime (as assumed in the relevant post-closure safety case), rather than to a threshold dissolution rate. There are two main reasons for this recommendation:

- It can be extremely difficult to obtain consistent, comparable and reproducible data on wasteform chemical durability from leach tests, given the dependence of dissolution rates on many different factors (including temperature, pH, leachate composition, substrate morphology and surface area, and experimental method). This could hamper the ability to demonstrate compliance with a criterion based on the wasteform dissolution rate.

- Many thermally treated wasteforms are significantly more durable than those produced via more conventional conditioning routes such as cement encapsulation or bituminisation. It is questionable whether it is justified to apply a durability criterion to such wasteforms that might be stricter (in terms of permissible dissolution rate) than any comparable criterion that could be applied to other wasteforms. A more durable wasteform would, in any case, be produced and it may not be possible to improve the wasteform durability further without compromising other characteristics. It would be preferable to demonstrate the durability of a proposed wasteform through testing of candidate phases as part of wasteform development work, and to use this as evidence supporting the longevity of the wasteform and its ability to provide containment.

**Proposed generic disposability criterion:** *Existing requirements on chemical durability for the applicable disposal route should be applied to thermally treated waste. No additional criteria on chemical durability for thermally treated waste are considered necessary, although requirements relating to the containment provided by a wasteform may be justified, depending on the post-closure safety case.*

#### *4.3.1.13 Voids*

Minimising voids within waste packages is important in order to:

- Maximise packaging efficiency and reduce costs and risks associated with waste package handling, storage and disposal.
- Avoid compromising waste package integrity to the extent that it could limit safe waste package stacking in a repository.
- Avoid detrimental impacts on repository post-closure evolution. For example, the migration of void spaces originating within waste packages could generate pathways for more rapid groundwater transport.

Relatively few national programmes define acceptance criteria relating to voids within waste packages, although Belgium and France give indicative limits of 20% and 25% respectively. Considerable work on this topic has been conducted in the United Kingdom on behalf of RWM (see, for example, Towler *et al.*, 2017) and 'voidage' screening levels may be adopted by RWM in future.

Thermal treatment may help to reduce the extent of void space within waste packages, by melting or degrading porous or awkwardly shaped waste constituents (e.g. filters or metal poles). For some wastes, thermal treatment may be a more straightforward route to minimise void space than alternatives such as size reduction, compaction or encapsulation.

The desire to minimise voids in disposed waste packages may influence aspects of how thermal treatment (and any subsequent conditioning / packaging) is implemented, as illustrated by the following scenario: Volume reduction is often observed during thermal treatment as the raw waste melts / degrades. If thermal treatment is conducted in a vessel that is designed to be disposed of alongside the waste (as is the case for in-can vitrification techniques), then this could result in significant void space being present in the vessel. If this void space is deemed to be an issue for disposability (as well as packaging efficiency) then approaches to reduce this could include:

- Adding additional raw waste or matrix forming species (e.g. glass frit) as thermal treatment of the initial waste charge progresses.
- Topping off the vessel with a non-compressible material after cooling.

**Proposed generic disposability criterion:** *Void space within packages of thermally treated waste should be minimised wherever practicable; this may influence aspects of how thermal treatment is implemented.*

#### *4.3.1.14 Waste package stacking*

The number of waste packages that can be stacked on top of each other in a disposal facility is governed mainly by the design of the waste container(s), which varies from country to country (and may need to be adapted for thermally treated waste, as discussed above). The wasteform may contribute additional strength and rigidity, particularly where a dense wasteform with a high compressive strength is generated, such as a glass or a ceramic. Conceivably, a wasteform with a high compressive strength could allow package stack heights to be increased. However, thermally treated waste is relatively dense, so the mass of associated waste packages could increase. Moreover, increasing stack heights would probably be driven by site-specific constraints, such as limitations on disposal vault / tunnel height or floor loading. On balance, therefore, it is not considered necessary or appropriate to factor requirements on waste package stacking into generic disposability criteria.

**Proposed generic disposability criterion:** *Existing requirements on waste package stacking for the applicable disposal route should be applied to thermally treated waste. No additional criteria related to waste package stacking of thermally treated waste are considered necessary.*

#### *4.3.1.15 Waste package impact performance*

Where defined, impact performance criteria are typically linked to limits on the dispersion of solid radioactive content in the event of a waste package being dropped from a defined height, sometimes in a particular orientation, onto a defined target (e.g. a flat floor or a spike).

Powdered / unconsolidated wastes, such as those resulting from pyrolysis or gasification could potentially be readily mobilised and released into the surrounding environment in an impact event, e.g. if a package incorporating such waste were dropped and broke open. This is one of the main drivers that could potentially require further conditioning of the products from such thermal treatment routes in order to ensure that they are passively safe (as discussed in Section 4.3.1.3). This consideration is mainly relevant during the operational period of a repository.

Vitreous and ceramic wasteforms could be susceptible to cracking if dropped, which would increase the surface area available for interaction with groundwater in a repository and, hence, affect their dissolution and ability to immobilise radionuclides. This consideration is relevant during the post-closure period and is of particular relevance where the safety functions applicable to the wasteform place an emphasis on long-term containment of radionuclides

**Proposed generic disposability criterion:** *Existing requirements on impact performance for the applicable disposal route should be applied to thermally treated wastes. Consideration* *should also be given to how an impact event could affect the long-term durability of a wasteform resulting from thermal treatment / conditioning and the safety functions it provides.*

#### *4.3.1.16 Waste package fire performance*

The following national requirements have been identified in relation to temperature excursions in the event of a fire:

- Continued integrity of waste packages and their closure systems.
- Limited or no dispersion of waste package contents.
- No uncontrolled or runaway processes initiated by temperature excursion (e.g. explosion / exothermic reaction).
- No significant geometric changes to the waste package.

In addition, disposed wastes should not contain flammable substances or substances that could undergo significant exothermic reaction (e.g. pyrophoric material).

As for criteria relating to the chemical content / reactivity (discussed in Section 4.3.1.11) and gas generation (Section 4.3.1.10), criteria relating to fire performance are considered to be of relatively limited importance for thermally treated wastes. By definition, thermal treatment involves the application of high-temperature treatment to waste, and thermal treatment processes will generally cause the decomposition (through incineration, oxidation or gasification) of most, if not all, waste constituents that are susceptible to degradation under high temperatures. The resulting waste products are expected to be non-flammable and relatively refractory. They are unlikely to undergo further reaction in the event of a fire in the repository; nor are they expected to provide combustible materials that might initiate such a fire. As such, it is considered that fire performance WAC applicable to wastes that have not been thermally treated would easily be met by the products from thermal treatment, and it is not necessary to propose any additional criteria for thermally treated waste under this category.

**Proposed generic disposability criterion:** *Fire impact WAC for the applicable disposal route should be applied to thermally treated waste (and are expected to be easily met). No additional criteria related to waste package performance for thermally treated waste during fires are considered necessary.*

#### *4.3.1.17 ID / labelling*

Most radioactive waste management programmes require that each individual waste package should incorporate a systematic, durable and unique identification code, which links it to associated production documentation and information about the waste package inventory. Some national programmes (e.g. the United Kingdom) give precise specifications for the location, number and format of such labels, whereas other programmes leave this open for waste producers to decide.

General requirements for labelling waste packages that enable them to be identified during handling, storage and disposal are considered to be sufficient for thermally treated waste, and no additional / alternative disposability criteria are deemed necessary. It is noted that if the disposal container itself is involved in the thermal treatment / conditioning step (e.g. if it is filled directly with molten waste, rather than acting as an overpack for an inner vessel), then it could be advisable to apply waste IDs / labels after the thermal step has been completed, to ensure that the labels do not become damaged in any way as a result of heat treatment.

**Proposed generic disposability criterion:** *ID / labelling requirements for the applicable disposal route should be applied to thermally treated waste. No additional criteria on labelling for thermally treated waste are considered necessary.*

#### *4.3.1.18 QA / QC requirements*

Waste producers must have a quality management system (QMS), covering all steps to process raw waste into a passively safe conditioned and packaged wasteform suitable for disposal and setting out associated management arrangements. Compliance with the QMS ensures that wastes have been processed in line with applicable specifications / criteria, such that there is confidence in their acceptability for disposal.

QA / QC requirements are generally written:

- either at a high level, such that they can be applied to a wide variety of wastes and conditioning approaches, in which case, existing requirements should apply equally well to thermally treated wastes (and no additional criteria for thermally treated wastes are necessary); or
- in a manner that applies specifically to a particular packaging approach, in which case it is difficult to extract requirements that are relevant for a list of generic disposability criteria.

**Proposed generic disposability criterion:** *QA / QC requirements for the applicable disposal route should be applied to thermally treated waste. No additional criteria related to QA / QC for thermally treated waste are considered necessary.*

#### *4.3.1.19 Data management / data requirements*

The following national requirements have been identified in relation to data management:

- Typical data inputs required include packaging data from waste producers and repository monitoring data.
- Data inputs should support effective nuclear material accountancy (for any wastes with a significant content of fissile material).
- Written and/or electronic records are generally required.
- Data management as part of management and insurance measures should be independent of repository safety measures, i.e. it must not be conducted in a way that affects repository safety, and implementation of safe disposal should not influence the achievement of planned data management arrangements.

None of these requirements are of specific relevance to thermally treated waste. It is therefore considered that it would be largely sufficient require that thermally treated wastes meet all applicable data management WAC that apply to other wastes consigned for disposal via a particular route. The one possible addition is that there may be benefit in requiring records of the thermal treatment regime applied to the waste (including the maximum temperature, hold time and any matrix-forming additives) to be kept as part of the waste package production documentation. This would support any future evaluation of waste properties / potential interactions with other components of the disposal system that might be needed. Data of this type would, in any case, be needed to enable compliance with many of the criteria already discussed to be determined.

**Proposed generic disposability criterion:** *Data management requirements for the relevant disposal route should be applied to thermally treated waste. In addition, records of the thermal treatment regime applied to the waste should be kept.*

##### 4.3.1.20 Secondary waste

The potential to generate secondary wastes is not included as a category in the Section 3 table. However, it is mentioned in the text response from Switzerland. The potential to generate secondary waste or effluents is an important consideration for operational safety and efficiency that relates directly to the processing route. It is already relevant from a disposability perspective, since secondary wastes can add to the total volume of radioactive waste requiring disposal and may pose their own disposability challenges. The secondary wastes associated with thermal treatment routes may differ considerably from those associated with other, more conventional, encapsulation routes (e.g. there may be relatively large volumes of filters or decommissioning wastes associated with enhanced off-gas treatment systems). Depending on the thermal treatment technology and the characteristics of the secondary wastes, it may be possible to incorporate some secondary wastes in later thermal treatment runs, to reduce the quantity arising as a separate waste stream.

**Proposed generic disposability criterion:** *Secondary waste associated with thermal treatment should be minimised to the extent that is practicable.*

### 4.3.2 Summary of generic disposability criteria

Table 4 summarises the proposed generic disposability criteria derived in Section 4.3.1 for thermally treated waste. If used, these criteria would need to be applied in conjunction with existing criteria applicable to other wastes that are planned for disposal in a particular facility. They do not stand alone as a complete set of requirements that could underpin WAC for thermally treated waste.

| Topic / Category | Proposed generic disposability criterion | Considerations applicable to measure compliance |
|---|---|---|
| Dimensions / mass of packages | *The dimensions and mass of containers used to package thermally treatment waste (and other aspects of the container design) should be compatible with the thermal processing route being employed.*<br>*The dimensions and mass of containers used to package thermally treated waste (and other aspects of the container design) should be compatible with relevant safety functions for storage and disposal, and with all applicable constraints on waste classification, handling, transport and disposal, taking account of the processed waste characteristics.* | *None.* |
| Provisions for handling, transport and emplacement | *No additional criteria on provisions for handling, transport and emplacement for thermally treated waste – apply existing criteria for the disposal context in question.* | *The characteristics of thermally treated waste should be considered as part of demonstrating compliance with existing requirements on transport, handling and emplacement.* |
| Package integrity and required lifetime | *Apply existing criteria for the disposal context in question.*<br>*Any additional criteria on package integrity defined for thermally treated waste should be linked to safety functions applied to such waste.* | *The characteristics of thermally treated waste should be considered as part of demonstrating compliance with existing requirements.* |
| Activity content | *No additional criteria on activity content for thermally treated waste – apply existing criteria for the disposal context in question.* | *Demonstrating compliance with existing criteria should take account of the potential for activity to become concentrated in a smaller volume during thermal treatment. Associated implications for waste classification and waste package handling should be considered.* |
| Radionuclide inventory | *No additional criteria on declaration of the radionuclide inventory for thermally treated waste – apply existing criteria for the disposal context in question.* | *The choice of thermal processing route and wasteform morphology / formulation should be tailored to the radionuclide inventory (and* |

| Topic / Category | Proposed generic disposability criterion | Considerations applicable to measure compliance |
|---|---|---|
| | | *other characteristics) of the waste, particularly if thermal treatment is driven by the need to produce a durable, long-lived wasteform.* |
| Dose rate limits | *No additional criteria on dose rate limits for thermally treated waste – apply existing criteria for the disposal context in question.* | *Demonstrating compliance with existing criteria should take account of the potential for activity to become concentrated in a smaller volume during thermal treatment.* |
| Surface contamination | *No additional criteria on surface contamination for thermally treated waste – apply existing criteria for the disposal context in question.* | *Ensuring compliance with existing criteria should account for potential contamination mechanisms that are specific to the thermal treatment route employed.* |
| Nuclear criticality | *No additional criteria relating to criticality safety for thermally treated waste – apply existing criteria for the disposal context in question.* | *The potential impacts of fissile material concentration on transport, operational and post-closure safety should be considered.* |
| Thermal output | *The thermal output of thermally treated waste should not have a detrimental impact on performance of the engineered and natural barriers that make up the disposal system, taking account of the potential for activity to be concentrated during thermal treatment.* | *None.* |
| Gas generation | *No additional criteria on gas generation for thermally treated waste – apply existing criteria for the disposal context in question.* | *None.* |
| Chemical content | *Apply existing criteria for the disposal context in question.*<br>*The choice of thermal treatment route and the design of the associated disposal facility should ensure the chemical compatibility of thermally treated waste with other disposal system components.* | *None.* |

| Topic / Category | Proposed generic disposability criterion | Considerations applicable to measure compliance |
|---|---|---|
| Chemical durability | *Existing requirements on chemical durability for the applicable disposal route should be applied to thermally treated waste. No additional generic disposability criteria for thermally treated waste are considered necessary, although requirements relating to the containment provided by a wasteform may be justified, depending on the post-closure safety case.* | *If criteria relating to the durability of a thermally treated wasteform are deemed to be required for application in a particular context, then it is recommended that these should be linked to a required containment lifetime (as assumed in the relevant post-closure safety case), rather than to a threshold dissolution rate.* |
| Voids | *Void space within packages of thermally treated waste should be minimised wherever practicable; this may influence aspects of how thermal treatment is implemented.* | *None.* |
| Waste package stacking | *No additional criteria related to waste package stacking for thermally treated waste – apply existing criteria for the disposal context in question.* | *None.* |
| Waste package impact performance | *No additional criteria on impact performance for thermally treated waste – apply existing criteria for the disposal context in question.* | *Consideration should be given to how an impact event could affect the long-term durability of a wasteform resulting from thermal treatment / conditioning and the safety functions it provides.* |
| Waste package fire performance | *No additional criteria on fire performance for thermally treated waste – apply existing criteria for the disposal context in question.* | *None.* |
| ID / labelling | *No additional criteria on labelling for thermally treated waste – apply existing criteria for the disposal context in question.* | *None.* |
| QA / QC requirements | *No additional criteria related to QA / QC for thermally treated waste – apply existing criteria for the disposal context in question.* | *None.* |

| Topic / Category | Proposed generic disposability criterion | Considerations applicable to measure compliance |
| --- | --- | --- |
| Data management | *Data management requirements for the relevant disposal route should be applied to thermally treated waste. In addition, records of the thermal treatment regime applied to the waste should be kept.* | |
| Secondary waste | *Secondary waste associated with thermal treatment should be minimised to the extent that is practicable.* | None. |

*Table 4. Summary of proposed generic disposability criteria*

# 5 Identification of characterisation requirements in WP4

## 5.1 Methodology

The final purpose of work package 4 is to analyse the impact of thermal treatments on the disposability of radioactive waste. This disposability is evaluated through the Waste Acceptance Criteria (WAC) previously identified in part 3. Some of these criteria are related to a procedure or to some data independent of the waste product, but the other one are related to a physicochemical parameter. In this case, some characterisation tests have to be carried out on the thermally treated waste product. Moreover, at a R&D stage, it is required to understand and to have a good knowledge of the waste products coming from thermal treatment processes. This can be achieved through characterisation tests, which can be defined thanks to the methodology described hereafter:

- Following their identification (see part 3), WAC are reviewed. It is not required to identify quantified criteria at this stage of the work. Generic criteria are sufficient to identify the characterisation tests which will be carried out. Moreover, the context of each country is taken into account: quantified/qualitative criteria, more or less advanced program, safety requirements, etc. At the end of this step, a list of generic criteria is available. (see part 5.2)
- The next step consists in the identification of the physicochemical parameters which relate to the list of the previously identified criteria. More than one physicochemical parameter can correspond to one criterion, and one parameter can correspond to more than one criterion. (see part 5.3)
- In parallel, the available characterisation tools which have been proposed by WP4 partners are reviewed. The objective is to identify which kind of characterisation tests can be carried out. (see part 5.4)
- Then, the characterisation tests which correspond to the physicochemical parameters are identified. These characterisation tests enable the study of the previously identified physicochemical properties. (see part 5.5)

The results of this work are described in the next parts (from 5.2 to 5.5).

Following this work, tests will be compared and contrasted before being adapted to the requirements. Sometimes more than one option is available. In this case, the best compromise will have to be chosen. This work will be detailed in a forthcoming report.

## 5.2 Identification of WAC requiring characterisation

Following their identification in part 3, and the development of generic criteria in part 4, WAC have been reviewed to determine the requirements in terms of characterisation. A list of criteria and parameters has been derived from these WAC. This list is based on the contribution of WP4 partners. It has to be noted that the issues and the challenges can be different between countries. As a consequence, some of the criteria identified hereafter are not an issue for some countries, whereas it is the case for the other ones. It should also be underlined that these criteria can apply to a broad range of waste, such as vitrified waste, cemented waste, etc.

The criteria and parameters which request characterisation tests are the following ones:

1- No free liquid or gas: generally, free liquids are completely prohibited. Gas generation can be limited or prohibited, with, sometimes, a limit in terms of pressure or of flowrate.

2- Permeability and/or diffusivity of the waste: this parameter has to be sufficient to evacuate gas (or other products).

3- No or limited content of hazardous materials: this criterion is common for most countries. Thermally treated waste products must not contain material which could be combustible, pyrophoric, reactive, etc.

4- Immobilisation of radionuclides: in most cases, radionuclides have to be immobilised in the waste matrix. This could be met thanks to a solid matrix for the waste. But, it should be reminded that a solid matrix is not always necessary in some countries.

5- Limited voids / limited porosity: these parameters have to be measured for different purposes, depending on the country.

6- The thermally treated waste product should not contain hot spots, that is to say an accumulation of radioactive material.

7- Knowledge of the leaching behaviour of the waste product: this parameter has to be studied to understand the long term behaviour and the chemical durability of the waste.

8- Mechanical resistance of the waste product: in some countries, the thermally treated waste product has to resist to the mechanical constraints of transport, of disposal, to the impacts, etc.

9- No metal with a redox lower than 0.84 V HSE: This criterion is specific to Belgium but can require specific characterisation tests. More generally, the content of metal can be of interest for some countries.

10- Thermal behaviour of the waste: it covers thermal conductivity and thermal resistance of the waste product. It is especially important for self-heating waste, or in case of fire.

## 5.3 Physicochemical properties

In order to define the right characterisation tests enabling the evaluation of the criteria and parameters (see part 5.2), the corresponding physicochemical properties has to be identified. This identification is summarised in the following table.

| Waste Acceptance Criteria | Physicochemical properties |
|---|---|
| No free liquid or gas | Homogeneity of the waste |
| Permeability and/or diffusivity of the waste sufficient to evacuate gas or other products | Permeability + diffusivity |
| No or limited content of hazardous materials (combustible, pyrophoric, reactive, etc.) | Homogeneity of the waste (no untreated area) + identification of chemical species in the waste |
| Immobilisation of radionuclides | Distribution of radionuclides in the waste |
| Limited voids / limited porosity | Porosity |
| No hot spots | Homogeneity of the waste / microstructure |
| Leaching behaviour of the waste product | Chemical durability |
| Mechanical resistance of the waste product (mechanical constraint in disposal, impacts, etc.) | Mechanical behavior |
| No metal with a redox lower than 0.84 V HSE | Homogeneity of the waste / microstructure |
| Thermal conductivity of the waste product (especially for self-heating waste) | Thermal conductivity / thermal behavior |

This table highlights that most of physicochemical properties are very specific and can be applied to only one criteria or parameter. However, the characterisation of the homogeneity of the waste can provide information on a broad range of criteria and parameters, such as the presence of liquid, gas, hot spots, hazardous materials, etc.

## 5.4 Available characterisation tools

In the first part of the project, partner laboratories identified different characterisation techniques they could use in the framework of WP4[2]. The results of this inventory are detailed in the table below. A short description of these techniques is proposed under this table.

| Characterisation technique | FZJ | USFD | SCK | CEA | VUJE |
|---|---|---|---|---|---|
| a spectroscopy | ✓ | ✓ | | | |
| Atomic Force Microscopy (AFM) | | ✓ | | | |
| Autoradiography | ✓ | | | | |
| Extended X-ray Absorption Fine Structure spectroscopy (EXAFS) | | ✓ | | | |
| Gas chromatography (GC) | ✓ | | | | |
| Gas physisorption | ✓ | ✓ | | ✓ | |
| Gas pycnometry | | ✓ | | | |
| Inductively Coupled Plasma - Mass Spectrometry (ICP-MS) | ✓ | ✓ | | ✓ | |
| Inductively Coupled Plasma - Optical Emission Spectrometry (ICP-OES) | ✓ | ✓ | ✓ | ✓ | |
| Infrared spectroscopy (IR) | | | | ✓ | |
| Ion Chromatography (IC) | | ✓ | | ✓ | |
| Leaching tests | ✓ | ✓ | ✓ | ✓ | ✓ |
| Liquid Scintillation Counting (LSC) | ✓ | ✓ | | | |
| Mössbauer spectroscopy $^{57}Fe$ | | ✓ | | | |
| Nuclear Magnetic Resonance spectroscopy (NMR) | | | | ✓ | |

[2] The identified characterisation techniques doesn't cover all the available techniques in each laboratory, but the ones that could be used in the framework of Theramin.

| Characterisation technique | FZJ | USFD | SCK | CEA | VUJE |
|---|---|---|---|---|---|
| Optical microscopy | | ✓ | | ✓ | |
| Raman spectroscopy | | ✓ | | ✓ | |
| Scanning Electron Microscopy (SEM/EDX) | ✓ | ✓ | ✓ | ✓ | |
| Thermogravimetric Analysis (TGA) | | ✓ | | ✓ | |
| Time-of-Flight Secondary Ion Mass Spectrometry (ToF-SIMS) | | | | ✓ | |
| Total Organic and Inorganic Carbon analyzes (TOC/IC) | ✓ | | | ✓ | |
| Transmission Electron Microscopy (TEM/EDX) | | | | ✓ | |
| Ultraviolet-visible spectroscopy (UV-Vis) | | | | ✓ | |
| Vertical Scanning Interferometry (VSI) | | ✓ | | | |
| Wide-Angle X-ray Scattering (WAXS) | | | | ✓ | |
| X-Ray Diffraction (XRD) | ✓ | ✓ | ✓ | ✓ | |
| X-Ray Fluorescence (XRF) | | ✓ | ✓ | ✓ | |

**Short description of characterisation techniques:**

**a spectrometry** is used in radiochemistry to measure activity of a sample of a emitters.

**Atomic Force Microscopy (AFM)** is a local probe microscopy technique used to visualize the surface topography of a sample.

**Autoradiography** is an imaging technique made from a radioactive sample placed in contact with an emulsion or a photographic film.

**Extended X-ray Absorption Fine Structure spectroscopy (EXAFS)** is an X-ray absorption spectrometric technique that mainly uses synchrotron radiation. It provides information on the local atomic environment (distance and coordinence) of a given element and is applicable in any type of environment: solid, liquid, gas and interfaces.

**Gas Chromatography (GC)** is a technique that separates molecules from a mixture. It applies mainly to gaseous compounds or compounds that can be vaporized by heating.

**Gas physisorption**: the specific surface area of a powder sample is estimated from the amount of gas adsorbed on its surface. The information is interpreted according to the model of Brunauer, Emmett and Teller (BET method).

**Gas pycnometry** (Helium) determines the volume of a solid sample (massive, particle or porous) of known mass, allowing the calculation of its density.

**Inductively Coupled Plasma (ICP)** spectrometry is a physical method of chemical analysis that allows for the quantification of almost all elements simultaneously. After separation, the atoms are detected by Optical Emission Spectrometry (OES) or Mass Spectrometry (MS).

**Infrared spectroscopy (IR)** considers the infrared region of the electromagnetic spectrum and is used for the identification of compounds or to determine the composition of a sample.

**Ion Chromatography (IC)** means the identification of ions using ion exchange resins.

**Leaching tests** are dedicated to the study of the chemical durability of a material. It involves contacting a solid and a solution under conditions set by a variety of standards. Leaching tests are associated with solid and solution characterizations.

**Liquid Scintillation Counting (LSC)** is the measurement of activity of a radioactive sample generally used for a and b particle detection.

**Mössbauer spectroscopy** is a highly sensitive spectroscopy method based on g-rays absorption by atomic nuclei. It helps to study the valence states of atoms, their chemical bonds, and their coordination within solid phases. It is applicable to a fairly limited number of elements among which $^{57}$Fe is the most studied.

**Nuclear Magnetic Resonance spectroscopy (NMR)** exploits the magnetic properties of nonzero spin atomic nuclei, whether in solutions or solids, to provide information on their local atomic environment.

**Optical microscopy** makes it possible to magnify the image of a small object.

**Raman spectroscopy** is a characterisation method used to identify the various structural groups in a material, which is sensitive to local atomic vibrations generated by optical phonons.

**Scanning Electron Microscopy (SEM)** is a technique that produces high resolution images of a sample surface using electron-matter interactions. It can be associated with X-ray Energy-Dispersive microanalysis (EDX) to study the chemical composition of the sample by using the X-radiation caused by the electron beam.

**Thermogravimetric Analysis (TGA)** is a thermal analysis technique that measures the mass change of a sample over time for a given temperature or temperature profile.

**Time-of-Flight Secondary Ion Mass Spectrometry (ToF-SIMS)** is a surface analysis method that consists of bombarding a sample with an ion beam to measure the elemental, isotopic or molecular composition of the sample surface.

**Total organic and inorganic carbon analyzes.** Total organic carbon (TOC) is the amount of carbon found in an organic compound and "inorganic carbon" (IC) represents the content of dissolved carbon dioxide and carbonic acid salts.

**Transmission Electron Microscopy (SEM)** combines high resolution images with diffraction. It can be associated with EDX.

**Ultraviolet–visible spectroscopy (UV-Vis)** refers to absorption spectroscopy in the ultraviolet-visible spectral region and is used for the quantitative determination of different analytes.

**Vertical Scanning Interferometry (VSI)** is a local probe microscopy technique used to visualize the surface topography of a sample.

**Wide-Angle X-ray Scattering (WAXS)** is based on elastic scattering of monochromatic X-rays. It may inform about the medium range order in glass materials.

**X-Ray Diffraction (XRD)** is an X-ray diffraction-based analysis technique that provides access to a variety of information contained in the arrangement of elements within a material.

**X-Ray Fluorescence (XRF)** spectrometry is a technique for the chemical analysis of the elemental composition of a sample.

## 5.5 Selected characterisation tools

In the following table, the WAC described in section 5.3 are associated with some of the characterisation tools inventoried in section 5.4. When it is possible, these techniques are chosen from those available in the majority of the partner laboratories. The purpose of this table is to identify which tests will be compared and contrasted in order to define the characterisation tests to be carried out in WP4. This work will be developed in a further report.

| Waste Acceptance Criteria | Measurements |
|---|---|
| No free liquid or gas | TGA, XRF, electron microscopy<br>... |
| Permeability and/or diffusivity of the waste sufficient to evacuate gas or other products | XRF, electron microscopy<br>... |
| No or limited content of hazardous materials (combustible, pyrophoric, reactive, etc.) | XRF, XRD, ICP after dissolution<br>... |
| Immobilisation of radionuclides | a spectrometry, autoradiography, Raman spectroscopy<br>... |
| Limited voids / limited porosity | WAXS, BET (open porosity) |

| Waste Acceptance Criteria | Measurements |
|---|---|
| | ... |
| No hot spots | XRF, electron microscopy<br>... |
| Leaching behaviour of the waste product | leaching tests, ICP, IC, UV-Vis spectroscopy, a spectrometry<br>... |
| Mechanical resistance of the waste product (mechanical constraint in disposal, impacts, etc.) | hardness, Young's modulus, toughness<br>... |
| No metal with a redox lower than 0.84 V HSE | XRF, electron microscopy<br>... |
| Thermal conductivity of the waste product (especially for self-heating waste) | thermal conductivity measurement<br>... |

This table emphasizes that most of the WAC can be verified by the use—possibly combined—of electron microscopy and analyses of the chemical and radiological compositions of the samples. These characterisations tools are of interest for the next phases of WP4. However, some WAC require specific tools. In this case, a selection will be done in a further report.

# 6 Conclusion

This report covers the whole scope of the first task of WP4, task 4.1. It compiles the data concerning WAC relevant to thermally treated waste products based on the contribution of eight European countries (Belgium, Finland, France, Germany, Lithuania, Slovakia, Swiss and the United Kingdom), but also the development of generic criteria, and the preliminary work on thermally treated waste product characterisation, the identification of the characterisation requirements based on the identified WAC.

The first part is an overview of the national radioactive waste management strategies. It gives information about categorisation of waste in each country, existing disposals or disposal concepts under study and, depending on the partner, main deadlines, dates, potential issues and challenges, etc.

The second part of the report is dedicated to the identification of WAC in each country. These criteria cover a large range of parameters which have been identified through the help of a table (see appendix A). Depending on the context of each country (more or less advanced waste management programs), the information shared are more or less detailed. Sometimes, WAC are quantified, but sometimes simply qualitative. Moreover, some criteria can be directly applied to thermally treated waste, but some of them are derived from other waste type such as cemented waste.

Based on these WAC, the third part describes the development of generic criteria. This work was especially focused on the identification of criteria which could be specific to thermally treated waste products. A table sums up these generic criteria at the end of the chapter.

The last part of this document is a preliminary work preparing the characterisation test phase (task 4.2). The requirements in terms of characterisation were identified. Following this preliminary work, characterisation tests will be discussed and adapted with Theramin partners.

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None

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None

## 7.9 General

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Neall, F.B., (2016) *Integrated Project Team on Uranium: Phase 2 [Task DCD-5]: Preferred Disposal Concept Options for Depleted, Natural and Low Enriched Uranium*, Galson Sciences Report to RWM 1207-DCD-5-1 Version 2.1, September 2016.

G. Towler, L. Harvey, S. Watson, R. MacKean, F. Neall, J. Sharp, J. Knights, T. Hicks, J. Wilson, *Managing Voidage in a GDF: Towards a Methodology for Evaluating Voidage in Waste Packaging Proposal*s, QRS-1698F-R1, Version 3.0, November 2017.

## Appendix A: Template of the section 3 table

| | CRITERIA | DERIVATION BASIS / ORIGIN OF CRITERIA | RELEVANCE FOR THERMALLY TREATED WASTE PRODUCTS | RELEVANCE FOR HANDLING, OPERATION OR AFTER CLOSURE SAFETY | PRIORITY OF THE CRITERIA |
|---|---|---|---|---|---|
| Facility (if applicable): | If applicable, please briefly describe, 1 paragraph, site and repository design, e.g. near surface facility vs. deep geological disposal. Also mention if this is an operating licensed facility, a proposed facility or a facility under construction | If applicable, please explain what are the drivers for developing WAC, including how are they used and on what basis are they developed? What is the context of the country (ongoing project or less advanced program)? | | | |
| *Note: Criteria or requirements can be quantified (i.e. a limit value), but it can also be an obligation to declare or to demonstrate/prove/justify.* | | | | | |
| Physical dimensions, weight | Are there package-specific dimension and weight requirements in the WAC? What is the maximum size/ weight that could be accepted, i.e. bounding values? | If available, please state your derivation basis for developing the below criteria. For example, how were the requirements of say radiological properties derived? Based on government regulations (e.g. reversibility of disposal) / international guidelines / safety assessment or others? | | | |

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| | | **=> Applicable for all the cells below.** | | | |
| | Any specific requirements to ensure compatibility with provisions for transport, handling and emplacement? | | | | |
| Integrity | Please list the requirements for maintaining integrity of waste packages and the required lifetime. | | | | |
| Activity content | How are the requirements of radioactivity level specified for transportation, handling, storage and emplacement, by radionuclide, total activity or class (e.g. alpha)? Please provide example if possible. | | | | |
| | Are there other requirements / limits, e.g. low specific activity, surface contamination limits? | | | | |
| Radionuclide inventory | Please describe the requirements of the type, characteristics, and contents of radionuclides. | | | | |
| Dose rate | Please describe the package dose rate limits for your packages. | | | | |

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| | Are there specific external dose rate limits for waste material or unshielded packages? How to control radiation rate to ensure radiation induced processes (e.g. radiolysis) and degradation of material properties of the packages, repository components and the host rock do not occur to an unacceptable degree? | | | | |
| Surface contamination | Is the maximum surface contamination (non-fixed) specified? Is the same value applicable to all transportation, handling, storage and emplacement packages? | | | | |
| Nuclear criticality | Is criticality a concern? If so, what is the max fissile material? Please also describe how packages are designed to preclude nuclear criticality. If applicable, please also describe how are your handling, storage, disposal systems designed and operated to ensure criticality of arrays of packages cannot occur. | | | | |
| Containment | What are the requirements in terms of containments? | | | | |
| Thermal output | What is the maximum allowable thermal load or heat output of a waste package? | | | | |

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| Radiological gas generation | Are there specific rules in term of radiological gas release? | | | | |
| Non radiological gas generation | Are there specific requirements to ensure gas generation in waste packages will not jeopardize the performance of the container, surrounding media and the overall disposal system? | | | | |
| | Are there specific requirements to limit explosive gas generation ($H_2$…)? | | | | |
| Chemical content | Is there some requirements to declare chemical content of the waste form? Please describe. | | | | |
| | Explosive, pyrophoric, hazardous or combustible materials, fire and explosion hazards: Are there specific requirements in your WAC to reject packages containing these materials / ignition hazards? | | | | |
| | Corrosive materials: What are the requirements for corrosive materials? If applicable, please describe how to determine the contents of corrosive materials? How to determine measurement accuracy to demonstrate compliance with authorized limits? | | | | |

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| | Toxic materials: What are the requirements for toxic materials? If applicable, please describe how to determine the contents of toxic materials? How to determine measurement accuracy to demonstrate compliance with authorized limits? | | | | |
| | Complexing compound: What are the requirements concerning complexing compound? | | | | |
| | Compressed gases: Are there specific requirements in your WAC to reject containers of compressed gases for disposal? | | | | |
| | Free liquids: Are there requirements to limit the quantity of free liquids in waste packages? | | | | |
| | Other properties of waste form: Please describe the required properties of waste forms, e.g. immobilized radionuclides, etc. How are these properties measured? | | | | |
| Chemical durability | Please state the requirement s for ensuring chemical durability. | | | | |
| Volume of voids | Is there requirement concerning the volume of voids? Please describe. | | | | |

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| Stackability | Please describe the maximum number of stacked packages and requirements of deformation/abnormality exhibit. Please explain how to ensure stacking stability. | | | | |
| Impact performance | Please describe the requirements on impact performance of different waste packages. Please include the maximum loss of content in all considered drop test scenarios, if applicable. | | | | |
| Fire performance | Please describe the fire performance requirement of your different packages, i.e. temperature, duration and the maximum loss of contents | | | | |
| Identification | Please describe requirements on waste package identification, e.g. locations, format of identifier, size of characters, | | | | |
| Quality control | Please describe your quality control procedure, i.e. how to identify defective packages | | | | |
| Quality assurance | Please describe your quality assurance arrangements / requirements | | | | |
| Data requirements | Please state any specific requirements for data to be recorded. | | | | |