

National Inventories and Management Strategies for Spent Nuclear Fuel and Radioactive Waste

Final Methodology



Radioactive Waste Management and Decommissioning

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for Spent Nuclear Fuel and Radioactive Waste**

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Cover photos: Maritime shipment of high-level radioactive waste, Cherbourg, France (Dean Calma, IAEA); Presentation of a container for the storage of medium-level long-lived waste at the Meuse and Haute-Marne underground research laboratory in Bure.

Foreword

Radioactive waste and spent fuel inventory data are one of the most important elements in developing a national radioactive waste management programme, as they affect the design and selection of the ultimate disposal methods. Inventory data are generally presented in the form of radioactive waste amounts under various waste classes, according to the waste classification scheme developed and adopted by the host nation and/or national programme. Spent fuel is generally inventoried separately.

International conventions and treaties, such as the Joint Convention on the Safety of Spent Fuel Management and Safety of Radioactive Waste Management (JC) and requirements in the context of the European Council Directive 2011/70/EURATOM of 19 July 2011, establish a community framework for the responsible and safe management of spent fuel and radioactive waste. These international recommendations or requirements oblige the member countries to develop a credible strategy that demonstrates that radioactive waste and spent fuel can be safely managed from generation to final disposal. NEA member countries are also being forced to contend with this growing need to conform to such conventions and treaties.

Various waste classification schemes have evolved around the world, with most countries classifying their waste according to its origin, dose protection criteria or to the physical, chemical and radiological properties of the waste and/or its planned disposal method(s). Such variance has restricted the comparison of waste inventories generated in different countries and has also given rise to challenges in comparing waste management practices, both nationally and internationally. The waste schemes that have evolved in different countries have also prevented the development of a global radioactive waste inventory and have limited comparisons of how waste is managed among countries. A methodology to ensure the consistency of national waste inventory data when placed in a common scheme would address such shortcomings.

The OECD Nuclear Energy Agency (NEA) Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) was established by the NEA Radioactive Waste Management Committee (RWMC) in 2014 to develop a methodology that would ensure consistency of national radioactive waste inventory data presented in a common scheme and provide the best achievable comparability of data within the framework of international programmes and initiatives. The need to develop such a methodology arose in the context of a joint initiative among three organisations – the NEA, the International Atomic Energy Agency (IAEA) and the European Commission (EC) – entitled the “Status and Trends Project on Spent Fuel and Radioactive Waste”.

The NEA expert group was mandated to review the radioactive waste and spent fuel management strategies of NEA member countries with the goal of developing a common presentation format for the national inventory in terms of the radioactive waste and spent fuel management strategies that have been established in member countries. The expert group was thus tasked with developing a methodology to support the Status and Trends Project. The Project published the first edition of a global spent fuel and radioactive waste inventory in January 2018, with plans to update the publication on a regular basis.

The NEA published the completed methodology and presentation scheme in September 2017. The present report provides a presentation scheme and a methodology for spent nuclear fuel and for all types of radioactive waste that could be included in inventories worldwide. The expert group has also been making efforts to facilitate the application of the methodology and completion of the presentation table through the development of a web-based tool. This tool, after it is finalised, will help to organise the process for filling in the presentation table in the form of a dialogue between the interface and the methodology user in order to standardise the population of the table and use it for the presentation of national inventory, not only for the Status and Trends Project but also for other cases.

Acknowledgements

The Nuclear Energy Agency (NEA) wishes to express its gratitude to the members of the Expert Group on Waste Inventorizing and Reporting Methodology (EGIRM) for their successful work on the development of a methodology to ensure consistency of interpretation of national radioactive waste inventory data. The NEA would also like to thank the following experts for their contributions to this report: Geert Volckaert (Federal Agency for Nuclear Control [FANC], Belgium), Mario Dionisi (Ispra, Italy), Maurice Heath (Nuclear Regulatory Commission [NRC], United States), Karin Kugel (Federal Office for Radiation Protection [BfS], Germany), Miklos Garamszeghy (Nuclear Waste Management Organization [NWMO], Canada), Sergey Deryabin (Rosatom, Russia), Elodie Petry (Andra, France), Bengt Hedberg (Swedish Radiation Safety Authority [SSM], Sweden), Dominique Dapei (World Nuclear Association [WNA]). Vladimir Lebedev, formerly with the NEA Secretariat, was responsible for overseeing the drafting of the report. Annex D was drafted by Niccolo Giani, intern at the Nuclear Energy Agency (NEA) and member of the NEA Expert Group on Waste Inventorizing and Reporting Methodology (EGIRM). Other members of the EGIRM have also had an active role in activities that support the reporting of country information on spent fuel and radioactive waste inventory data at the international level.

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List of abbreviations and acronyms

BWR	Boiling water reactor
DGR	Deep geological repository
DSRS	Disused sealed radioactive sources
EBS	Engineered barrier system
EC	European Commission
EGIRM	Expert Group on Waste Inventorizing and Reporting Methodology (NEA)
ENSREG	European Nuclear Safety Regulators Group
HLW	High-level waste
IAEA	International Atomic Energy Agency
ILW	Intermediate-level waste
LLW	Low-level waste
MS	Member states
NEA	Nuclear Energy Agency
NEWMDB	Net-Enabled Waste Management Database (IAEA)
NPP	Nuclear power plant
OECD	Organisation for Economic Co-operation and Development
OPC	Ordinary Portland cement
PWR	Pressurised water reactor
RW	Radioactive waste
RWM	Radioactive waste management
RWMC	Radioactive Waste Management Committee (NEA)
SF	Spent fuel
VLLW	Very low-level waste
WAC	Waste acceptance criteria
WNA	World Nuclear Association

Chapter 1. Objectives and background

In 2013, the “Status and Trends” Project on Spent Fuel and Radioactive Waste was initiated as a joint activity among three organisations – the Nuclear Energy Agency (NEA), the International Atomic Energy Agency (IAEA) and the European Commission (EC).

The Status and Trends Project set out to establish an instrument that would provide a better understanding of the global picture of spent fuel and radioactive waste management, and the main contribution of the NEA would be to create a methodology that would help underpin this understanding.

The NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) was thus established following a decision of the Radioactive Waste Management Committee (RWMC) at the 47th meeting in March 2014. The primary objectives of this expert group, as specified in the mandate, were set out as follows:

- to develop a methodology that would ensure consistency of national radioactive waste inventory data when they are included in a common presentation scheme (this scheme will be used only to compare and combine radioactive waste inventory data), with a focus in the first phase on spent fuel and radioactive waste from the reprocessing of spent fuel;
- to support NEA member countries in preparing their national report for the “Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management” (IAEA, 1997) and the “Council Directive 2011/70/EURATOM” (EU, 2011) with the above-mentioned method being used as a tool for the comparison and compilation of data;
- to propose the above method as a tool for presenting spent fuel/radioactive waste data when preparing publications in the context of the joint “Status and Trends” Project.

The expert group successfully performed the first stage of the work and presented the results to the RWMC at the 49th plenary meeting. The RWMC approved the developed methodology, its publication and extension of the mandate for the next two-year period. For the second stage, the RWMC agreed on the following EGIRM objectives:

- expanding the methodology to cover all national radioactive waste and spent fuel inventory data with relevance to all management strategies based on the disposal routes; testing the methodology on several national inventories; and proposing the methodology to the joint “Status and Trends” project for the presentation of radioactive waste and spent fuel data in publications;
- promoting the methodology among potential implementers (EC, IAEA and others) and supporting implementation when requested;
- providing proposals on standardising – where possible and when it has high added value – the national data reported to the joint “Status and Trends” Project (and, where possible, to other international programmes) and using the developed methodology to provide better application of the methodology as a tool for presenting data;
- investigating possible methods to provide improved quality and flexibility of data to address the requirements of potential implementers, and developing relevant recommendations;
- studying the potential for a web-based version of the methodology and the presentation scheme so as to facilitate and unify the reporting process and further data aggregation of the presented data.

In 1994, the IAEA published a safety guide on the classification of radioactive waste (SS 111-G-1.1) to guide member states in developing their waste classification. In 2008, SS 111-G-1.1 was superseded and a new safety guide, GSG-1 (IAEA, 2009), was issued. GSG-1 sets out a general scheme for classifying radioactive waste primarily based on considerations of long-term safety, i.e. on the minimum appropriate disposal method. However, at this point most countries already had their own waste classification systems, which varied widely between countries.

To collect and disseminate radioactive waste management information, the IAEA developed a Net-Enabled Waste Management Database (NEWMDB) for member countries to report their radioactive waste management data, including inventories, on a regular basis. To input the radioactive waste data into the NEWMDB, the IAEA proposed a country-specific Waste Classification Translation Matrix and a method to translate the radioactive waste inventory data from the various national radioactive waste classes into IAEA radioactive waste classes. Translation into a common classification scheme was necessary in order to aggregate the national inventory data in the NEWMDB.

In 2011, the Council of the European Union adopted the Directive 2011/70/EURATOM (EU, 2011) establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste, which requires, *inter alia*, all EU member states to have a national radioactive waste classification scheme. The first round of reporting under Directive 2011/70/EURATOM was not performed by EU member states until after the first stage of work (i.e. September 2016) of the NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) had been completed, and thus the EGIRM only took into consideration the relevant publicly available outcomes of this reporting activity at the second stage of its work.

There is a wide variety of national radioactive waste classification schemes that have been established worldwide. While many of these schemes were developed using the IAEA's waste classification scheme (i.e. GSG-1, SS 111-G-1.1) as a reference, only a few member countries have fully adopted the actual IAEA scheme (GSG-1) in their most recent waste classifications, for a variety of reasons. For example, some countries use long-established systems with the infrastructure and regulations developed around it, and thus they are hesitant about ensuring the logistical difficulties that would be encountered when switching to a new system.

The classification of radioactive waste proposed in the GSG-1 "is based primarily on the long-term safety consideration, and thus, by implication, disposal of the waste" (IAEA, 2009). While the safety goals and objectives are the same, the implementation of the long-term safety for radioactive waste management may be very different among member countries as it depends on various parameters which are often unique to each country and even to each repository. These differences may lead to significant uncertainties when comparing inventories among countries, if the comparisons are based only on long-term safety considerations. It has become apparent that a combination of two approaches – translation to the IAEA international scheme to compare inventories from the point of view of minimum safety requirements, and a presentation in technical terms of the management strategy and disposal routes – can provide a more complete view and comparability of national inventories.

A consensus was reached on the need to develop a method so as to transpose, as well as possible, national radioactive waste inventory data to a common radioactive waste presenting format so that the data of countries could be compared. The method developed would focus on the technical aspects of the disposal methods to facilitate inventory comparisons. Another intention was to try to combine into one scheme both the spent fuel and radioactive waste inventories and the management strategies in place in these countries. It was pointed out that such a method should have no influence on existing radioactive waste classification schemes or spent fuel/radioactive waste management strategies but should simply allow for a direct comparison and understanding of the different spent fuel/radioactive waste management practices.

It should be noted that radioactive waste classification, along with the qualitative and quantitative criteria used, significantly depend on the current and/or planned short-term and long-term strategy of radioactive waste management in a given country and on its nuclear infrastructure and regulatory practices.

Chapter 2. The methodology

2.1. The presentation table

Initially, the expert group had developed a presentation table in a format to present radioactive waste produced during the reprocessing of spent fuel separately. At a later date, when the Status and Trends Project group decided not to request the distribution of the waste amounts by their origin, the NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) took advantage of this additional opportunity to optimise the table format.

Table 1. The presentation table

Spent fuel (SF) and radioactive waste (RW) inventory presentation												
Country: _____												
Date of inventorying: _____												
SF/RW types (in national terms)	Disposal option not yet decided	SF reprocessing/ service		Disposal in:								
		home	abroad	UF-1		UF-2		NSF-1		NSF-2		Others
A	B	C ₁	C ₂	D ₁	D ₂	E ₁	E ₂	F ₁	F ₂	G ₁	G ₂	→
1. SF												
1.1. NPP												
1.2. Other reactors												
2. HLW, [m ³]												
3. class, [m ³]												
4. class, [m ³]												
Equivalence with IAEA GSG-1	2.											
	3.											
	4.											
FOOTNOTES												
1)	<i>The text of a footnote</i>											
2)												
3)												

A detailed description of the presentation table and its cells is given in the previous EGIRM report (NEA, 2017), and therefore in this report only a brief overview is presented.

Column (A) lists all national types of spent fuel and radioactive waste classes (lines 1.1 and 1.2 – for spent fuel from nuclear power plants (NPPs) and other reactors respectively; lines 2, 3, 4 and further (when necessary) – for national radioactive waste classes).

The sub-section, entitled “Equivalence with the IAEA’s GSG-1 classification (type),” shows where the respective waste classes (as in GSG-1) fall within the presentation scheme matrix. To present the correlation between the national class of radioactive waste (which is added in a cell of the main section) and the corresponding GSG-1 class, the EGIRM recommends that the methodology user input:

- in the main section, the radioactive waste volume (m³) in the row of the national class and under the column that is in accordance with the management option practised for this radioactive waste;
- in the sub-section, under the column used in the previous step, the corresponding GSG-1 class designation (e.g. high-level waste [HLW], intermediate-level waste [ILW] or very low-level waste [VLLW]) in the cell, in which the row has the same number as in the previous step. (In its simplest form, it would be a 100% correlation. When national classes do not line up precisely with GSG-1 definitions, fractional amounts can be used – e.g. 40% ILW, 60% low-level waste [LLW]).

Thus, a user should input figures in the main table and designations of radioactive waste classes in the sub-section. For example, as shown in Table 2, the user is going to present 100 m³ of radioactive waste of A class (in terms of the national classification) which is to be disposed of in a UF-2 facility. “A” class corresponds to 100% to ILW (GSG-1). For this, the user will input:

- “A class” in the row number 3 of the main table under column A;
- 100 in row number 3 of the main table under the column E1;
- “ILW” in row number 3 of the sub-section under the column E1 (see Table 2).

Table 2. Correlation with GSG-1

Spent fuel (SF) and radioactive waste (RW) inventory presentation												
Country: _____												
Date of inventorying: _____												
SF/RW types (in national terms)	Disposal option not yet decided	SF reprocessing/ service		Disposal in:								
		home	abroad	UF-1		UF-2		NSF-1		NSF-2		Others
A	B	C1	C2	D1	D2	E1	E2	F1	F2	G1	G2	→
1. SF												
1.1. NPP												
1.2. Other reactors												
2. HLW, [m ³]												
3. A class, [m ³]						100						
4. ... class, [m ³]												
Equivalence with IAEA GSG-1	2.											
	3.					ILW						
	4.											

Column (B) presents data for spent fuel or radioactive waste for which there is no currently decided disposal route. Examples include spent fuel/radioactive waste placed into a long-term storage facility and awaiting a decision, spent fuel/radioactive waste collected because of past activities. The definition of “decided” is left open for individual countries to define. However, for better comparability, it is recommended that “decided” would mean a government announcement had been made concerning future plans and strategies for spent fuel/radioactive waste management. For EU member states, this should be in line with the “policy” decisions as reported in the framework of the Directive 2011/70/EURATOM.

Important: spent fuel or radioactive waste currently stored and awaiting reprocessing/disposal should not be put into Column (B) since its disposal route is decided.

Column (C) presents data for all spent fuel to be reprocessed, including spent fuel, sent abroad (for reprocessing or another service) following the provisions of an international agreement. Column (C1) is for all spent fuel to be reprocessed in the home country, including spent fuel “imported” from the “user” countries. Column (C2) is for spent fuel currently sent for reprocessing or another service abroad.

Radioactive waste obtained after reprocessing of imported spent fuel and waiting to be sent back to the “user” country can be presented by the “servicer” country in relevant cells of Column (C1). A “user” country can present radioactive waste after reprocessing abroad currently stored abroad in the relevant cells of Column (C2).

Column (D) is for direct disposal spent fuel, HLW and other radioactive waste for which a decision has been made that it should be disposed of in a UF-1 facility. Column (D1) is provided to present data for spent fuel/radioactive waste waiting for disposal in the UF-1. Column (D2) is to present data for spent fuel/radioactive waste that has already been disposed of in the UF-1 facilities. Since there are no operating UF-1 type facilities, this sub-column is greyed-out and should not be used until a UF-1 type facility is implemented.

Column (E) is to present data for all radioactive waste to be disposed of and already disposed of in a UF-2. Column (E1) is for radioactive waste to be disposed of in a UF-2 type facility. Column (E2) is for radioactive waste that has already been disposed of in a UF-2 type facility.

Column (F) is to present data for radioactive waste to be disposed of and already disposed of in an NSF-1 type facility. Column (F1) is for radioactive waste to be disposed of in an NSF-1-type facility. Column (F2) is for radioactive waste already disposed of in the NSF-1 type facilities.

Column (G) is to present data for radioactive waste to be disposed of and already disposed of in an NSF-2 type facility. Column (G1) is for radioactive waste to be disposed of in an NSF-2 type facility. Column (G2) is for radioactive waste already disposed of in NSF-2 type facilities.

It is recommended to present data for spent fuel/radioactive waste (each class) in the cells as total values and, when necessary, give relevant explanations and components of the sum in a footnote. It can be a sum of spent fuel/radioactive waste exported/imported to/from different countries (e.g. France – Fr, UK – UK, Italy – It), radioactive waste in different physical status (conditioned, unconditioned, liquid, solid), or in different basis of volume (e.g. as stored, as conditioned, as disposed). Under Table 1, the area for footnotes is presented as a table. However, a user is free to choose another format. For better comparability of data among countries, the expert group recommends filling in the table as much as possible with “as disposed” radioactive waste volumes. The definition of “as disposed” is left open for the country to decide based on their regulatory definitions and practices. However, it would typically represent the external volume of the waste package (“disposal module”) that is actually placed into the disposal facility.

2.2. Flowcharts for completing the presentation table

The EGIRM developed flowcharts presenting the process of completing the presentation table in a step-by-step manner. These were introduced in previous EGIRM reports. However, experts found some possibilities for improvement, and in Annex A, the updated flowcharts are given in order to facilitate the process of filling in the table. In the Annex, flowcharts are given for spent fuel from NPPs, and spent fuel from other reactors and radioactive waste. In each of the flowcharts, “+” means “yes”, while “-” means “no”.

2.3. Examples

At the testing stage, the EGIRM has collected completed tables from EGIRM member countries and several additional volunteer countries. The process of testing is continuing. Examples of national inventories are presented in the EGIRM format in Annex B.

2.4. Contribution to the Status and Trends Project

The EGIRM presented the completed methodology to the Status and Trends Project joint group in June 2017 and proposed the methodology for use in national profiles. The joint group recognised that the presentation format was of benefit for the project, and it will thus include the methodology. The Status and Trends group decided: “In the National profiles the reporting tables were simplified. One new table was introduced to provide an overview (EGIRM).” The Status and Trends reports are published periodically every three years (the same cycle as the national reports to Joint Convention and the European Directive), and for the second publication, the EGIRM table will be introduced in the template of the country profile. The EGIRM will continue collecting country comments, proposals and observations so as to improve the explanatory part of the methodology, and make it clearer and unambiguous.

Chapter 3. Addressing uncertainties in the comparison of inventory data

Because of the differences in national waste classification schemes and interpretations on how waste volumes are calculated and reported, there can be considerable uncertainty when comparing or aggregating data from different countries.

The differences in waste classification schemes can be addressed by converting national schemes to a common reference scheme. Similar to other international projects, the International Atomic Energy Agency (IAEA) GSG-1 classification scheme was chosen as the baseline, and a “translation matrix” is provided as part of the NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) presentation table. The translation matrix is the same one used in the IAEA Net-Enabled Waste Management Database (NEWMDB).

The second main source of uncertainty relates to calculation and reporting of waste volumes, especially when radioactive waste does not yet exist in final form and the endpoint (disposal route) has not been implemented. It is always important to state which volume is being reported (as-generated, as-stored, as-conditioned, as-disposed) to be able to understand the volume figures correctly.

In this phase of the work, which is presented in greater detail in Annex D, the EGIRM focused on radioactive waste volume recalculations and aimed to consider the factors and parameters that influence volume changes during the radioactive waste treatment and conditioning. Various treatment, conditioning and packaging methods will change the volume of waste from its original “as generated” form to its final “as disposed” form. Some treatment methods will reduce volume (e.g. compaction, incineration, drying of liquids.), while others will increase volume (e.g. cementation/grouting of solids). Packaging will always increase the volume. When multiple layers of packaging are used (e.g. drums in a thick-walled disposal module) the overall volume increase can be quite significant.

The EGIRM, therefore, decided to collect and analyse information on radioactive waste treatment technologies and methods that are applied for radioactive waste management (RWM). The group identified the actual factors of the radioactive waste volume reduction or increment that each step applies to the initial volume of radioactive waste, following radioactive waste in a logical procession from the initial to final form.

The targeted audience that would benefit the most from this work, and from the discussion in the international RWM community, are countries that have not yet fully developed or implemented a management strategy (strategy for a given part of radioactive waste). This information would also be useful to those countries that have only partially implemented their RWM systems, leaving room for optimisation in some steps, or for management of some radioactive waste that are still stored and waiting for treatment/conditioning in a non-fully conditioned state.

In addition, this work can be somewhat useful for a broad audience including non-specialists, providing a clear illustrative explanation of the radioactive waste volume related terms used in the RWM community. The report presents all stages of radioactive waste volume evolution and the meaning of all volume parameters and values.

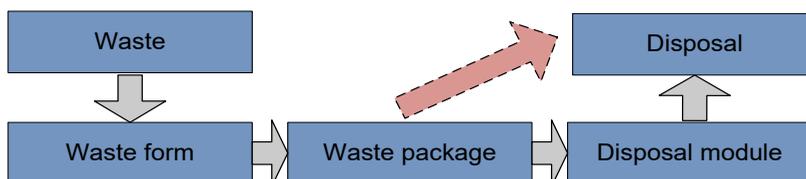
3.1. Volumes of radioactive waste and radioactive waste packages: Definitions

An understanding of the radioactive waste volume and its evolution throughout the lifetime of radioactive waste is of particular importance for the objectives of the EGIRM, and it should be analysed adequately. The EGIRM intention is to get as clear a picture as possible of the spent fuel/radioactive waste volumes stored and disposed of. As a first step, it is necessary to provide standard definitions concerning the radioactive waste volume evolution during the stages of treatment, conditioning, packaging and disposal. For the objectives of the EGIRM, experts have developed definitions based on the three key states or steps for radioactive waste:

- Waste form (WF) stage when radioactive waste, or the spent fuel, is in the form in which it will be disposed of, including any stabilisers but excluding the waste container. This is also sometimes referred to as the “net waste volume”.
- Waste package (WP) stage when the ensemble of one or more waste forms is together with their waste container that must be suitable for handling and storage, and may allow transport and disposal of if waste acceptance criteria (WAC), are addressed. The waste package volume is the external volume of the outermost container.
- Disposal module (DM) stage when the ensemble of one or more waste packages is together with their disposal container (over pack) and optional buffer; may be suitable for handling, transport, storage and must be suitable for disposal without further conditioning. In some cases, the disposal module may also be the waste package. The “as disposed” volume is the external volume of the disposal module.

This arrangement is proposed to give a clear interpretation of the concept of “as disposed”, and it makes it easier to follow the volume evolution of radioactive waste in the treatment and conditioning processes. The term “as disposed” can be attributed either to the disposal module or the waste package; whichever is the final form considered in the disposal facility design. The important thing is to clarify what RWM stage is referred to the moment when radioactive waste volumes can be considered “as disposed”. The “as stored” volume is the volume of the waste as it occupies a storage facility. It can be the same as the waste form volume, the waste package volume, the disposal volume or some other value, depending on the design of the storage facility and how the wastes are stored.

Figure 1. Sequence of radioactive waste volume evolution



* It is necessary to stress the fact that the waste package can be disposed of directly if the repository WAC are met. Also, it should be made clear that the term “conditioning” and “conditioned waste” refers to the final form of waste that is ready for disposal. The precise requirements of conditioning will depend on the WAC for the repository, and may range from simple packaging to sophisticated immobilisation processes. It may involve multiple steps, with resulting interim products, or a single step.

Developing the “as disposed” volume concept, the EGIRM worked closely with the group of the RepMet initiative (NEA Radioactive Waste Management Committee [RWMC]/IGSC) and incorporated a large part of the developed terminology. Detailed information on terms, definitions, their understanding and especially the content of data libraries is available in the RepMet report (NEA, 2018).

The expert group addressed the main features of volume evolution, following the scheme based on the three above introduced key states: the waste form, the waste package and the disposal module. Figure 1 gives a logical sequence starting from the volume of the raw

radioactive waste and ending with the final volume addressing the WAC established for the radioactive waste repository. This schematisation serves as a backbone for the division of the radioactive waste volume evolution in separate and defined stages, in order to consider each stage in more detail.

Details on the definition of each stage of radioactive waste volume and some examples are given in Annex C.

3.2. Advantages of using “as disposed” volume

A format of data presentation was discussed many times within the EGIRM and the Status and Trends Project, and in both groups, the experts reached the understanding that the most comparable form of data presentation is a that where all radioactive waste is levelised to a final conditioned form (a so-called “as disposed” volume). It is necessary to note that this approach is not absolute, and there are some uncertainties in the presented data, generally due to recalculation of current volumes into “as disposed” when the endpoint is not known for certain:

On the other hand, to provide adequate comparability and easy understanding of national situations in spent fuel/radioactive waste management, this approach gives the following benefits:

- it is internationally recognised that solidified and conditioned radioactive waste is the preferred form for final disposal; presentation of radioactive waste in this form excludes consideration of the mix of radioactive waste in different physical states;
- from the point of view of the final disposal solution, the presentation of radioactive waste in “as disposed” form allows the best comparability of demands in radioactive waste disposal infrastructure or sufficiency of existing facilities;
- recalculation of radioactive waste volume allows for an additional evaluation of the efficiency of technical and logistic parameters of RWM strategy implemented, or to be implemented, in a country.

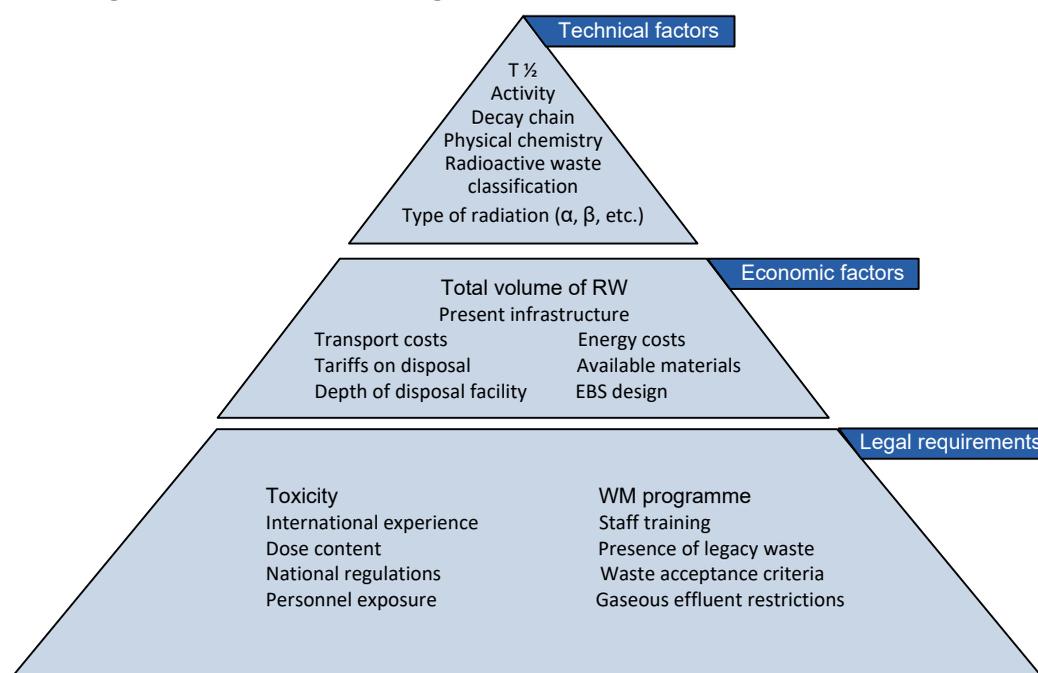
For the purposes of this report, radioactive waste volume should be understood as in “as disposed” form, consisting of the radioactive waste volume itself and additional, sometimes rather considerable, volume for the package, overpack, filling materials, etc. Some countries consider this extra packaging to be part of the “disposal system” and only consider the radioactive waste volume itself in its primary package to be the “as disposed” volume. This second approach is advantageous when a disposal approach has not been selected. The final packaging can be optimised depending on the disposal solution, yet the overall system can be designed around a known quantity of waste. The comparison of radioactive waste volume between countries in “as disposed” form is reasonable and generally more useful for discussion on the technical aspects of RWM, especially its final stage – radioactive waste disposal. When the safety of RWM is the subject of discussion, or a new disposal system is being planned, the volume of waste form inside the package is often more informative. Each time, when presenting the radioactive waste inventory, the user of the methodology can decide what format of data presentation should be applied. This should be explained of course in the national file. The proposed arrangement of different volumes in a unit for disposal is given in Annex C.

3.3. Factors for consideration during recalculation

Firstly, the EGIRM approach is to give structure to the parameters that can affect the volume change with a logical progression from raw waste to final packaging for disposal. The technical factors will depend on the processing, conditioning and packaging technologies that are selected. The available options may be different in different countries, as a result of regulations, the availability of infrastructure, local preferences, economics, etc. The first step in the evaluation is to document the legal requirements and WAC that are presented in the regulatory environment of the country. After this aspect has been considered, the next logical step is to review the available technical options and economic parameters that influence RWM. Some solutions may not be economically viable, even if legally allowed, and this can heavily influence

the decision-making process in the direction of lower costs. This economic analysis is recommended to include all the steps of RWM, from the generation to disposal of waste. It would be short-sighted to choose a particular solution or technology path because of lower costs in a certain phase of the process without consideration of the total cost, including previous, later and final stages that starting a technology choice would impose. Along with the economic factors, the technical factors that can influence a decision of a technology or process also need to be considered. The economic and technical factors are very often closely tied together and cannot be considered in isolation of each other. Here, focus can also be on the finer details of the industrial procedures correlated with a volume change of the radioactive waste. Below, Figure 2 presents the arrangement of factors considered or parameters influencing the radioactive waste volume evolution. The shape of the diagram does not convey the importance of the different group categories. It displays a logical progression and a series of essential prerequisites for each group. Other details are given in Annex D.

Figure 2. Factors influencing the evolution of radioactive waste volume



Source: OECD/NEA.

3.4. Recommendations on recalculation of radioactive waste volumes

After all previous analyses of influencing factors is completed and the chosen technology is cleared, the recalculation of waste volumes should begin. In earlier chapters, the evolution of radioactive waste volume from raw to final forms is described, in general terms. In the annexes, more common examples of volume change factors in the treatment and conditioning steps are provided as a reference. As previously mentioned, values given in the diagrams should be used carefully as indicative ranges since the radioactive waste volume evolution can be alternately increasing and decreasing. Merely multiplying all minimal or maximal values of volume factors will not result in the right approach for recalculations. The best approach is a combination of the recommended method with technical knowledge, and ideally, experience. Also, it is recommended that the results of some qualification tests, or of RD&D tests that mimic the considered waste treatment or conditioning process, be performed and closely taken into account. The values given by the group can be used for the planning of such experimental work. This approach can be beneficial for the study of radioactive waste volume evolution, for

example, in the cementation process. For the vitrification process, the recalculation can be made based on physicochemical calculations of the material balance. The glass density, in this case, is an essential parameter for the final volume forecast.

In the case of vitrification, the recalculation can be performed more simply.

To define the final volume of glass, one should select all of the elements in the radioactive waste content that are able to be incorporated into the chemical composition of the glass. The amount or the percentage of these elements in the glass composition is usually within specified boundaries. The choice of the glass type is defined based on the radioactive waste chemical composition, and the composition of typical glasses is well-known.

It is also necessary to note that the final glass composition is formed of the oxides of metallic (Na, K, Al, Fe, Ni, etc.) and non-metallic elements (Si, P, B, etc.). Generally, components such as $NxOy$, Cl, S, organic and water are almost fully eliminated from the initial radioactive waste. Cl and S can be incorporated into the glass matrix rather to only a limited extent (1-5% weight). It was found through experimentation that the recalculation of the radioactive waste content that considered the glass composition to be a set of oxides provides acceptable estimates for the glass composition.

When evaluating the ratio of radioactive waste– glass formers, it should be verified that the total amounts of all main components fit into the boundaries given for the relevant type of glass.

In general, the volume recalculation for vitrification can be described as follows:

- find the remaining part of the radioactive waste after the thermal destruction of chemical compositions and removal of water and gases as – $n_i = c_i \cdot V / M_i$, where n_i – number of moles of element “i” ions; c_i – concentration of element “i” in radioactive waste (e.g. g/l or kg/m³); V – radioactive waste volume unit (e.g. m³); M_i – molar mass of an element “i” (g/mole or kg/kmole);
- define the amount of glass-former to be added based on a comparison of the total content of the “i” element in glass and the amount remaining after radioactive waste treatment;
- define the weight of glass obtained from 1 m³ of initial radioactive waste as – $M = \sum_{i=1}^n k_i \cdot M_i$, where k_i – the number of moles of an element “i” oxide ($k_i = \frac{n_i}{x}$, where x is an index in the formula of oxide of given element E_xO_y); M_i – the molar mass of “i” element oxide (e.g. g/mole or kg/kmole);
- define the volume of glass based on the knowledge of glass density as $V_g = M / \rho$, where V_g – glass volume (l or m³), M – the mass of glass (g or kg), ρ – the density of glass (g/sm³ or kg/m³).

3.5. Overview of the features of containers developed for radioactive waste/spent fuel disposal

The EGIRM discussed the radioactive waste volume evolution from the raw form to “as disposed”, and recognised the considerable role of the “package factor” in this evolution. Considering the presentation of radioactive waste volume and the variety of its interpretation, the expert group deemed it useful to provide a brief overview of containers used in some NEA countries for spent fuel/radioactive waste management. At the same time, it is necessary to note that this overview is made only to provide some examples of existing types of containers, and it does not claim to be an authoritative source of information for recalculations of radioactive waste volumes. A number of references to relevant documents and reports have been provided in the present report. When making a decision, national experts can use the parameters of widely implemented containers as a reference to tentatively define the final volume of the waste packages and/or disposal modules.

Containers for radioactive waste/spent fuel transportation/storage/disposal can be conditionally divided into the following groups:

1. by purpose:
 - for collection and primary storage (at the site);
 - for transport;
 - for interim storage;
 - for disposal or long-term storage of:
 - low-level waste (LLW);
 - ILW;
 - high-level waste (HLW)/spent fuel.
2. by dimension and weight:
 - overall dimensions defined according to properties of radioactive waste, type of track, economic calculations, technological requirements and limitations, and other factors;
 - defining dimensions considering existing standards and limitations established by the International Organization for Standardization (ISO), which allows for the use of existing tracks and loading mechanisms.

International standard ISO for containers (see a list of references):

 - length 3; 6.1; 9.1; 12.2; 13.7 m;
 - transverse section dimensions 2.44 x 2.59 m;
 - mass of container (gross) (1, 3, 5, 10, 20 t and further).
3. By the radiological protection features during transport (e.g. according to the SSR-6 “Regulations for the Safe Transport of Radioactive Material” [IAEA]).
4. By the radiological protection features in relation to the type of radiation:
 - alpha emitters (biological protection is not required);
 - beta emitters (2-3 mm of steel);
 - gamma emitters (transport categories depending on activity);
 - neutron emitters (transport categories depending on activity);
 - mixed waste (transport categories defined by gamma/neutron activity).
5. By material of containers, e.g.:
 - concrete;
 - heavy concrete with hematite or barite;
 - steel, cast iron;
 - combined material (e.g. steel-concrete);
 - composite materials.
6. By content and physical form of radioactive waste:
 - disused sealed radioactive sources (DSRS) (α , β , γ , neutron) need containers with low or medium internal volume (0.2-2 m³) for storage, transporting and disposal.

The main requirements for the container, its material and package in total can be defined based on the specified function of the container. The role of an engineered barrier and its efficiency depends on the adequate selection of material.

The following factors need to be considered during the selection of material for the container:

- design lifetime and required operational period;
- mechanical features (tensile and compression strength, wear resistance, impact strength, plasticity and others);
- corrosion durability;

- compatibility with radioactive waste, its matrix and the container environment:
 - at treatment and conditioning when the container is used as a technological vessel;
 - during the storage period;
 - after the closure period in the repository;
 - considering the influence of the surrounding media in the long-term period, including disposal;
- weight and volume limitation;
- protection features;
- handling/lifting/stacking features;
- transport requirements (including shock test and fire test);
- for heat emitting waste; thermo-physical properties (thermal conductivity, specific thermal capacity, linear expansion factor, specific emissivity);
- radiation and chemical durability;
- manufacturing feasibility;
- optimal quality for the price.

There are many different types of materials and their combinations, which address the above requirements in varying degrees.

The waste package or disposal module should provide adequate protection against release of radionuclides at all projected conditions of operation. It should also ensure containment in case of interaction with other waste packages/disposal modules.

Specific numeric parameters of containers should be defined, taking into account the features of particular storage and disposal facilities. Finally, the characteristics of a site and the construction of a facility will determine the main parameters of selected containers.

The EGIRM conducted an overview of containers used in different countries for radioactive waste disposal. The results are collected and presented in Annex F.

It is necessary to note that the table in Annex F is given as a reference. It provides a general understanding of the possible forms of containers used in several countries, and it should not be considered as a global overview of containers.

Chapter 4.

Recommendations for standardising the reporting of radioactive waste/spent fuel inventories in international programmes and activities

Since 2014, European Commission (EC) services have been collaborating with the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA) to achieve a more standardised approach to the collection and presentation of data on present and future inventories of spent fuel and radioactive waste. This collaboration has occurred primarily through the “Status and Trends in Spent Fuel and Radioactive Waste Management” project (“Status and Trends Project”), which is a collaborative effort between the three organisations and is supported by other international organisations, such as the World Nuclear Association (WNA). The project involves the preparation of a report – on the same three-year reporting cycle as the Joint Convention and the Directive 2011/70/Euratom – on the global situation regarding the management of spent fuel and radioactive waste in IAEA member states (including EU member states). It includes a summary of inventories of radioactive waste and spent fuel. The IAEA published the first report from this project in January 2018, and work has begun on preparing the second cycle report. In addition, as required by the Directive, in May 2017 the Commission issued its first report on the implementation of the Directive in the EU member states, along with the inventory of spent fuel and radioactive waste on Community territory.

The IAEA, in collaboration with the Commission services, and supported by the NEA, is also developing an integrated reporting tool for spent fuel and radioactive waste inventories, to make the best use of the available information currently being reported by member states. The reporting tool, for which application will be strictly voluntary, will support:

- the use of a standardised data set;
- the facilitation of the reporting process in member countries to:
 - the EC;
 - the Joint Convention;
 - the IAEA;
 - the NEA.
- support consistency of inventory information presented in different international reports.

The standardised dataset has been developed mainly by integrating the data reporting requirements of the Directive (which are legally binding) and those of the Joint Convention. It is thereby hoped that reporting burdens on member countries were reduced in due course, e.g. by decreasing overlap and duplication of reporting activities.

4.1. Review of requirements/recommendations on reporting spent fuel/radioactive waste inventory data within international programmes/activities

The NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) reviewed available international documents that stipulate the requirements or recommendations/guidelines in relation to the format and types of inventory data that are required/recommended for use in national reports to international programmes/activities. Based on the relevant points in these documents, an analysis is provided in the section below.

Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (INFCIRC/546 24 December 1997)

The EGIRM studied the Joint Convention documentation concerning radioactive waste /spent fuel inventory reporting and selected requirements that had to be taken into account as relevant to the reporting of national spent fuel/radioactive waste inventories. The parts of the reports related to the requirements and recommendations in the Joint Convention on the form and content of spent fuel/radioactive waste inventory, as well as descriptions of national management strategies, are given as quotations in Annex E. In addition, requirements and recommendations from all relevant international documents are gathered in Annex E.

The Joint Convention thus requires reporting on spent fuel and radioactive waste inventories, management strategies and a description of the criteria used for the waste categorisation. Also, it requires reporting on the amounts of spent fuel/radioactive waste that is disposed of and to be disposed of. It is necessary to note that the Joint Convention requires reporting on the inventory of radioactive waste that has resulted from past practices. Some EGIRM experts have noted that the requirement on the description of material (radioactive waste) should be included in the national report.

Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management: Guidelines regarding the Form and Structure of National Reports – INFCIRC/604/Rev.2 (07 Sept 2012)

These guidelines give recommendations on what information should be provided in the national reports to the Joint Convention and how it should be presented. There is no requirement or recommendation to report national radioactive waste inventory data transposed into the GSG-1 radioactive waste classification scheme in either document. Countries are free to use their national radioactive waste classification schemes in their reports. Since there was no intent to aggregate the data, there is also no restriction on the reporting units (e.g. mass, volume, number of fuel assemblies) or calculation basis (e.g. as-stored, as-disposed).

Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste

The EGIRM experts studied the European Union Directive 2011/70 EURATOM to find requirements for national reports in sections related to spent fuel/radioactive waste inventory reporting (mostly Article 14). The relevant requirements are listed in Annex E.

The experts took into account this requirement in developing their recommendations.

HLG_p(2014-27)_137. C. “Final Guidelines for MS Reports to the Waste Directive”

These guidelines were developed by the European Nuclear Safety Regulators Group (ENSREG) to assist member states in fulfilling the requirement of Article 14.1 of Council Directive 2011/70/Euratom (hereinafter called the Waste Directive). Their purpose is to guide the member states regarding information and material that may be useful to include in the reports required under Article 14.1 (“National Reports”), and to establish a unified structure for reporting. These guidelines are intended to be read in conjunction with the text of the Directive. It contains recommendations to member states indicating that they are free to follow or not follow these recommendations. The EGIRM considered this document as the recommending guidelines in support of the reporting process, but did not consider any document directly connected to the Directive 2011/70.

In the document, recommendations are given regarding information that may be useful to include when reporting according to Article 14.1, and it is recommended to use a unified structure for national reports. This unified structure could facilitate the process of preparing the summary report to the Council and the European Parliament by the European Commission. The EGIRM has taken into account the ENSREG WG 2 activity on the revision of the document in order to address lessons learnt after the first reporting round.

The Waste Directive clearly recognises disposal as the endpoint of the management of spent fuel, when declared as waste and radioactive waste and the need for member states' policies and programmes to include disposal options.

In the inventory reporting, a distinction should be made between current inventories of radioactive waste and spent fuel, and future forecasts.

The EGIRM reviewed the "Final Guidelines for MS Reports to the Waste Directive" in 2017 when these were valid. Later, in January 2018, the ENSREG reviewed them and issued a new document "Guidelines for Member States reporting on Article 14.1. of Council Directive 2011/70/EURATOM". This document is based on the first guidelines that ENSREG issued in May 2014. ENSREG has taken into account the experience and lessons learnt from the process of preparation of the first National Report (submitted in August 2015). It has also taken into account the report from the European Commission to the Council and the European Parliament on the progress of implementation of Council Directive 2011/70/EURATOM and an inventory of radioactive waste and spent fuel present on the Community's territory, along with prospects in relation to their arisings. Reviewing the guidelines, the EGIRM recognised that they were intended to be read in conjunction with the text of the Directive and had no legal status.

From the point of view of inventory data reporting, the new guideline recommendations have not changed a great deal. However, the main points are nonetheless provided in Annex E.

4.2 The EGIRM approach to the development of recommendations for reporting standardisation

The EGIRM has collected information on the requirements and recommendations for national reports under the Joint Convention and the European Directive. This information was used when the group developed their recommendations on the joint reporting tool.

4.3 The EGIRM recommendations for standardisation

Analysing requirements and recommendations for the reporting of data related to national spent fuel/radioactive waste inventories, management strategies and disposal routes, the EGIRM developed and delivered to the joint initiative their recommendations on the development of a joint reporting tool.

1. The group has found that the clarification and standardisation of terms and definitions has been of value for the objectives of the joint initiative. Thus, the group recommends considering the following:
 - standardised definitions and understanding of the main terms related to radioactive waste/spent fuel management, the strategy of management and disposal route;
 - reporting data in common terms and units even when using national classification schemes can provide more legibility and comparability.

It was recognised that the harmonisation of the terminology and definitions could improve the reporting process. The joint initiative could propose to member countries developed or agreed terms, definitions and recommendations for achieving a better harmonisation at the international level. The EGIRM arrangement of radioactive waste disposal routes has been discussed in the joint initiative and recognised as being valuable for the future reporting tool.

2. The second point discussed by the EGIRM experts was improving the tools for the transfer of national radioactive waste classifications into the international (GSG-1) scheme. In particular, the group recommended considering the following:
 - facilitation of the transfer procedure through the revision of the existing guidance (Net-Enabled Waste Management Database [NEWMDB] method) or development of new guidance;

- avoiding as far as reasonable “approximated” assessments of radioactive waste volumes during the transfer, which can be obtained with the provision of agreed recommendations on the radioactive waste volume interpretation.
3. The EGIRM experts also considered the development and provision to countries supporting methodology for recalculation of “as is” radioactive waste volumes into “as disposed” to provide:
- facilitated and unified (as much as possible) recalculation;
 - fewer “rough estimates” during recalculation.

In the process of developing the joint reporting tool, the EGIRM reviewed the data set to be reported and provided its comments. A recommendation was given to the reporting tool developers on the use of this data set as a base for the elaboration of a web-based reporting tool. Ideally, this web-based tool should be an integral tool for information management, combining features of information collection and presentation, including for waste treatment, conditioning, storage and disposal options. The EGIRM recommended updating the existing NEWMDB database as the most advanced and adaptable means of reaching this objective. The EGIRM experts also suggested the integration of the EGIRM methodology into the developed web tool.

Chapter 5. Forms of the methodology implementation

5.1. The presentation scheme and methodology in the Status and Trends Project

At the meeting of the Status and Trends Project in June 2017, the Nuclear Energy Agency (NEA) presented the completed methodology to the joint working group. Following discussion, the joint group recognised the methodology and presentation table as being useful for the project objectives. The group decided to include the presentation table in the template for the national profile. The methodology will be introduced in the second (current) cycle of the report and expanded in future cycles.

5.2. Web-based tool for presentation of national data (NEA)

The NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) recognised that the best way to support widespread implementation of the methodology was to demonstrate the capabilities of the methodology on real national inventories. Addressing the EGIRM request, the NEA Radioactive Waste Management Committee (RWMC) recommended to member countries at its 50th annual session that they include the completed table in their national reports to international programmes. At the same time, the NEA started the creation of a web-based version of the presentation table and a simplified tool for populating it. With the web application, the process of filling in the table is organised in the form of a dialogue with the user. The NEA intends first to apply the tool within the RWMC web-page so as to test it when presenting the national inventory data of NEA countries. Then, the tested web version of the methodology can be proposed to the interagency group developing the joint reporting tool (revised NEWMDB entitled Spent Fuel and Radioactive Waste Information Tool [SWIFT]) for the integration of this tool. The developed joint reporting tool would thus combine both the collection of national inventory data and the presentation of this data in different formats including the EGIRM table, and it would then be proposed to countries for reporting to the Joint Convention and European Commission.

5.3. Integration of the methodology in the NEWMDB and/or developed joint reporting tool (IAEA, EC and NEA)

The Status and Trends Project team discussed the application of the NEWMDB in support of the project and possibly of reporting to the Joint Convention and European Commission (Directive 2011/70/EURATOM). In this regard, the EGIRM considered it logical to propose the methodology to be integrated into the developed joint reporting tool. The EGIRM experts kept in contact with the team developing this tool, discussing the technical aspects of such an implementation. As evoked in the previous section, the EGIRM will propose the designed and tested web-tool for inventory data presentation to the group developing SWIFT in order to incorporate it into the presentation part of the SWIFT.

Chapter 6. The methodology implementation: Outcomes of the NEA Workshop

6.1. The workshop concept and programme

In February 2018, the Nuclear Energy Agency (NEA) organised a workshop on the “Implementation of the EGIRM methodology for presenting of national radioactive waste and spent fuel management programmes”.

The main objectives of the workshop were as follows:

- support all interested parties in the implementation of the methodology through wide discussion, demonstration and training in its practical application;
- give detailed explanations and examples of how the methodology can be implemented;
- discuss possible ways of implementation and benefits from the implementation;
- convince the spent fuel and radioactive waste management community to use the methodology in their national and international communications ;
- collect feedback from workshop participants and use this feedback to make the implementation process more efficient.

Having developed a methodology to harmonise the presentation of different national radioactive waste/spent fuel management strategies and disposal routes connected to national radioactive waste and spent fuel inventories, the NEA then proposed a broad discussion and practical exercises on the application of the methodology. The EGIRM representatives presented the methodology to workshop participants. Representatives from waste management implementers, regulators, decision makers, responsible government ministries and other interested organisations from member countries were invited to join the workshop. International organisations (the European Commission [EC], the International Atomic Energy Agency [IAEA] and the WNA) that might potentially be interested in the methodology were also invited to participate in the workshop.

Representatives of 18 NEA member countries and the EC, the IAEA and the WNA participated in the workshop. The EGIRM presented in detail the methodology and presentation table over three sessions, with discussions among participants on possible ways of implementation and potential benefits, and then dedicated the fourth session to practical aspects. The workshop participants were invited to test the methodology on five proposed tasks prepared by the EGIRM. It should be noted that all of the tasks were “hypothetical” and all of the data were unrelated to any national radioactive waste management programmes. The exercise examples are provided in Annex F, as well as the completed tables for each exercise.

6.2. Main outcomes of the workshop (conclusions)

At the end of the workshop, the Chair made a summary report where the main conclusions of the workshop were summarised as follows:

- the workshop participants recognised the need for consistency in national data presented at the international level; such a consistency is considered to be clearly justified and of high added value;

- participants found the EGIRM methodology to be workable and a beneficial tool in the context of international dialogue; some benefits might also be obtained at the state level;
- after the test exercises and discussions on the presentation table, participants concluded that:
 - the meaning of “no strategy” (column B) should be explained in a more clear and unambiguous manner to avoid any misunderstanding among the different users;
 - a clear explanation is needed to present the correlation between radioactive waste amounts added in the cells of the table in terms of national classifications and the corresponding radioactive waste class of the international GSG-1 (IAEA) radioactive waste classification.

The EGIRM discussed the collected feedback and updated the explanation in the methodology. The updated table and revised explanation are provided in this report (see Section 2.1). The title of the “No strategy” column was changed to “Disposal option not yet decided”; and a clear explanation is given on how to present a correlation between national and international classes.

Chapter 7. Conclusions

In completing the present report, the Nuclear Energy Agency Expert Group on Waste Inventorizing and Reporting Methodology (EGIRM) has finalised a reporting methodology that will help to ensure the consistency of national radioactive waste inventory data presented in a common scheme and provide the best achievable comparability of data within the framework of international programmes and initiatives. This methodology will allow countries to present all types of radioactive waste and spent fuel in connection with their management strategies and disposal routes.

After having actively participated in and contributed to the joint initiative on the development of a joint standardised tool for national reporting both with regard to the International Atomic Energy Agency (IAEA) Joint Convention on the Safety of Spent Fuel Management and Safety of Radioactive Waste Management Joint Convention and the European Commission Directive 2011/70, the EGIRM has been able to develop a methodology that provides sufficient flexibility to address a wide range of possible, additional presentational needs. It has considered ways to decrease uncertainties in inventory data presentation, and has introduced definitions that will facilitate the understanding of the different radioactive waste management stages and their associated volumes. Based on previous considerations and the analysis of technologies, it has also developed recommendations on factors to be accounted for during recalculation and on the recalculations themselves.

The expert group is continuing to support the implementation of the methodology through the demonstration of its capacities at international events and by assisting countries in completing the presentation table with their inventory data, notably through a web-based assistance tool. This tool organises the process for filling in the presentation table in the form of a dialogue between the interface and the methodology user in order to standardise the population of the table and use it for the presentation of national inventory. The participating members of the EGIRM are eager to continue supporting the NEA Radioactive Waste Management Committee (RWMC) in its future activities related to the management of information, data and knowledge.

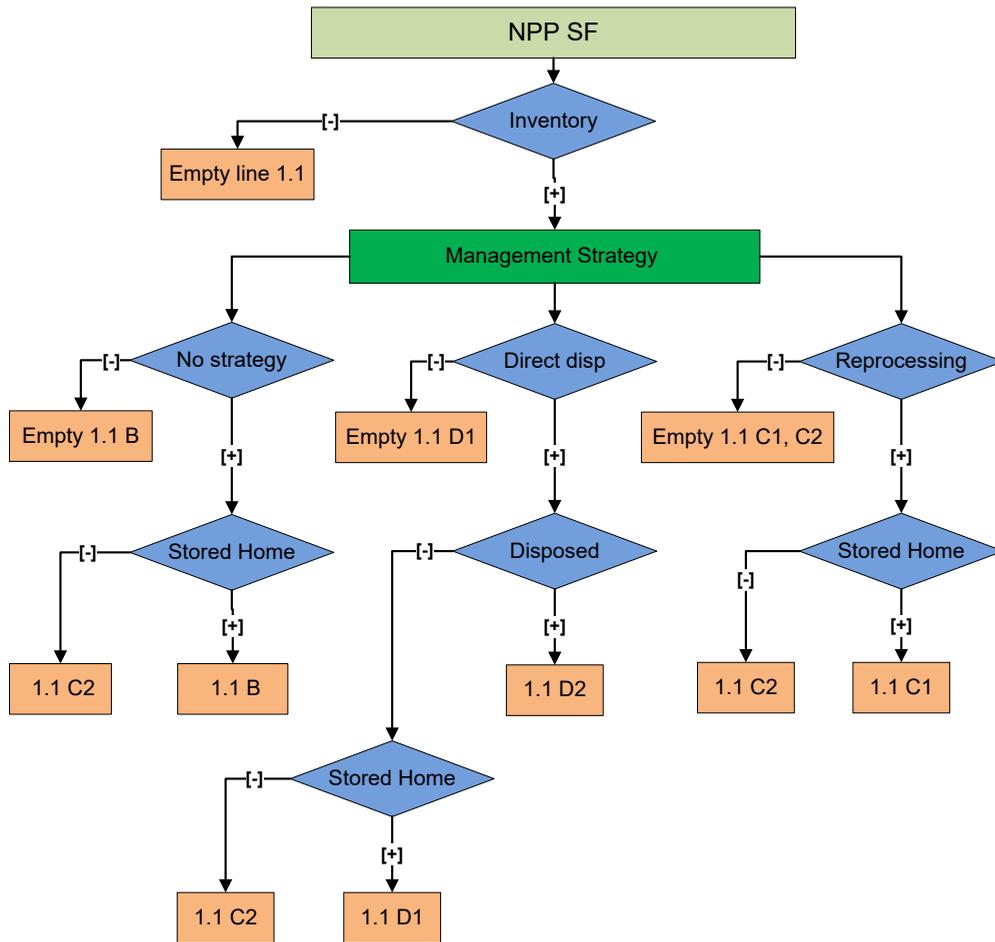
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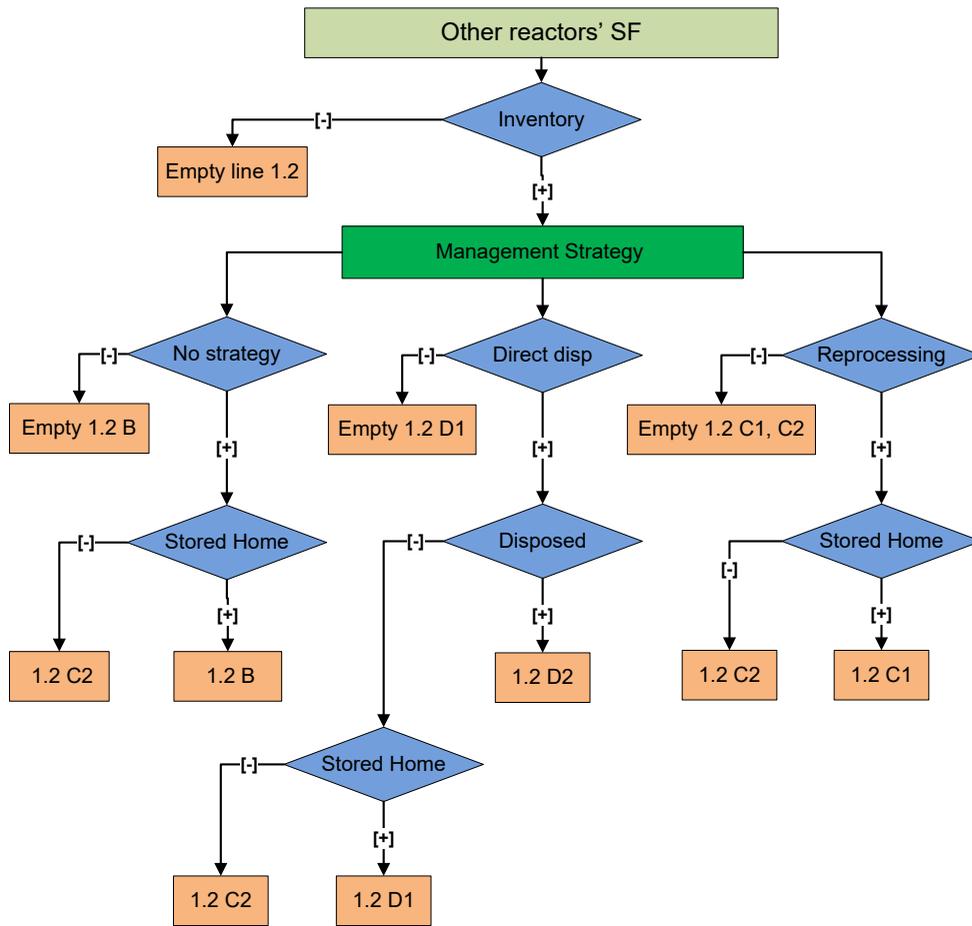
Annex A. Flowcharts for completion of the presentation table

Figure A.1. The flowchart for the completion of the table with data on NPPs' spent fuel



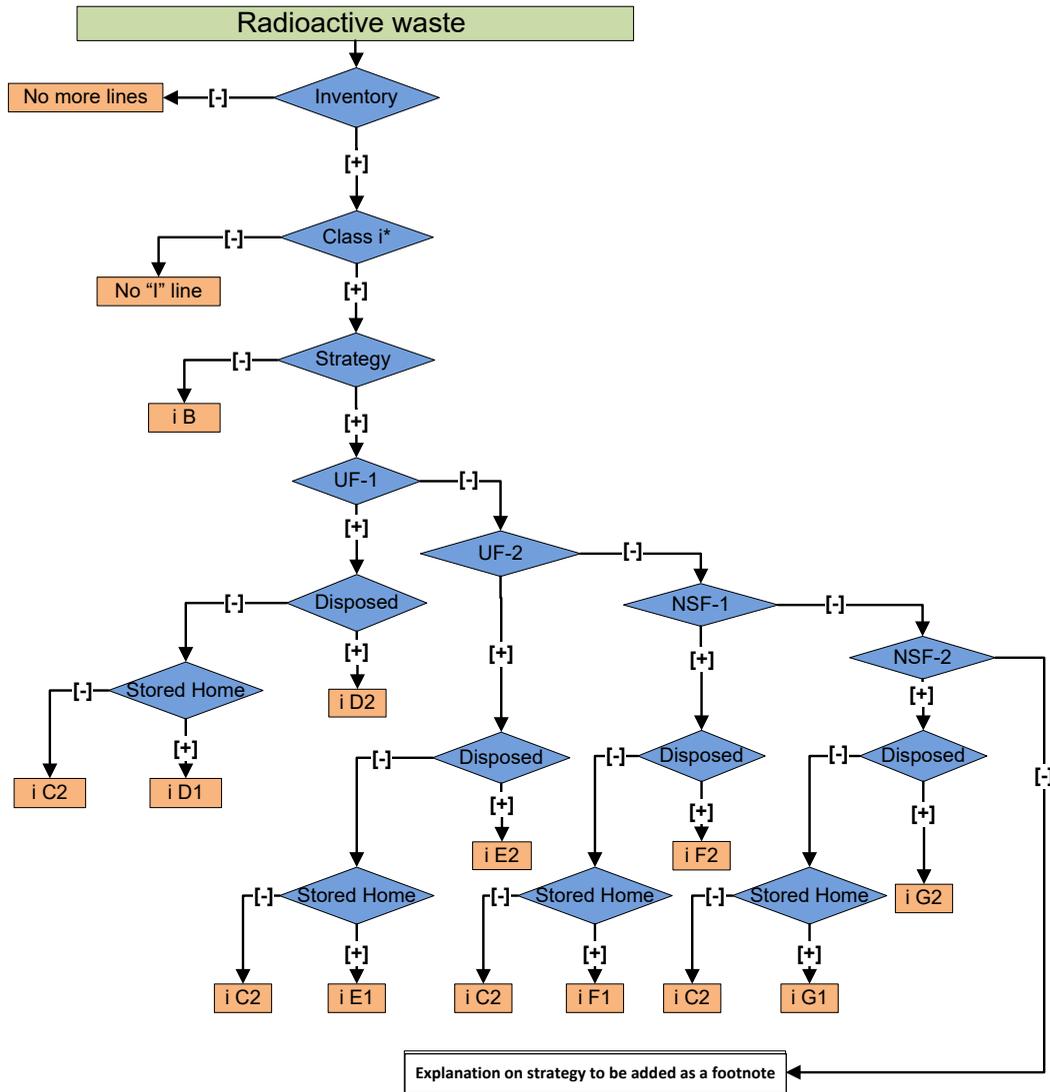
Source: OECD/NEA.

Figure A.2. The flowchart for completion of the table with data on other reactors' spent fuel



Source: OECD/NEA.

Figure A.3. The flowchart for completion of the table with data on radioactive waste



Important: National radioactive waste classes or types, disposal of which is out of the scope of this scheme, and for which additional columns are provided in the table, are not considered in this diagram. These radioactive waste volumes should be put in relevant cells in additional columns (rows). * i = 2,3,4... – serial numbers of lines in accordance with the number of national radioactive waste classes and disused sealed radioactive source (DSRS) types; in each cell, "Class i" should be replaced with national waste classes or DSRS types.
 Source: OECD/NEA.

Annex B. Examples of the national spent fuel/radioactive waste inventory data presentation

Spent fuel (SF) and radioactive waste (RW) inventory presentation													
Country: Italy _____													
Date of inventorying: 31.12.2013													
SF/RW types (in national terms)	Disposal route not yet decided	SF reprocessing/ service		Disposal of in:									
		home	abroad	UF-1		UF-2		NSF-1		NSF-2		SDL	SDS
(A)	(B)	(C1)	(C2)	(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)
1. SF													
1.1. NPP		27.7	1 848.25 ¹⁾										
1.2. Other reactors	2.74												
2. HLW, [m ³]			40 ²⁾										
3. ILW, [m ³]	5 540 ⁴⁾		34 ³⁾										
4. LLW, [m ³]								31 986 ⁵⁾					23
5. VLLW, [m ³]								3 870 ⁶⁾					
Equivalence with IAEA GSG-1 classification	2.		HLW										
	3.	ILW	ILW										
	4.							LLW					LLW
	5.							VLLW					

- 1) 1 848.25 = 963.2 tHM (UK) before 1978 without radioactive waste (RW) return + 678 tHM (UK) in 1978-2005 with RW return + 207.75 tHM (Fr).
- 2) 40 = 20 m³ vitrified high-level waste (HLW) (UK) (substitution of intermediate-level waste (ILW)-low-level waste (LLW) with only HLW is in discussion) + 20 m³ vitrified HLW (Fr) – volumes refer to canisters.
- 3) ILW to be returned from (Fr).
- 4) 5 540 = 2 360 m³ (estimated final volume of the conditioned liquid ILW from pilot reprocessing plants [EUREX and ITREC, still to be conditioned]) + 3 180 m³ (estimated final volumes after conditioning).
- 5) 31 986 = 6 371 m³ (estimated final volume of conditioned LLW from pilot reprocessing plants [EUREX and ITREC, mainly still to be conditioned]) + 25 255 m³ (estimated final volumes after conditioning).
- 6) Estimated final volumes after conditioning.

Spent fuel (SF) and radioactive waste (RW) inventory presentation													
Country: Germany _____													
Date of inventorying: 31.12.2013													
SF/RW types (in national terms)	Disposal route not yet decided	SF reprocessing/ service		Disposal of in:									
		home	abroad	UF-1		UF-2		NSF-1		NSF-2			
(A)	(B)	(C1)	(C2)	(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)		
1. SF													
1.1. NPP				8 216									
1.2. Other reactors				9.5									
2. RW with HG, [m ³]			859 ³⁾	569 ²⁾									
3. RW with negligible HG [m ³]						120.000 ³⁾	83 683 ⁴⁾						
Equivalence with IAEA GSG-1 classification	2.		HLW/ILW	HLW									
	3.					LLW/ILW	LLW/ILW						

UF-1: future HLW disposal.

UF-2: Konrad; Morsleben; Asse.

- 1) HLW stored in the UK and France and to be sent back 764 m³ (Fr) + 103 m³ (UK).
- 2) 569 = 554 m³ after NPP spent fuel (SF) reprocessing (volume given "as is" – 180 l canisters) + 15 m³ after other reactors' SF reprocessing.
- 3) RW to be disposed in UF-2 "Konrad".
- 4) 83 683 = 36 753 m³ (disposed in UF-2 "Morsleben") + 46 930 m³ (disposed in UF-2 "Asse") (in future, retrieval from "Asse" is planned and will increase the "to be disposed of" volume).

Annex C. Radioactive waste volume stages, description and examples

Waste form

- A waste form is commonly obtained as a result of the pre-treatment, treatment and conditioning of radioactive waste; the input volume can be either raw waste, partially treated waste, or even degraded old waste packages that have lost their container integrity and must be dealt with again.
- In this phase, in general, the greatest of volume changes to the initial radioactive waste is reached, normally being volume reductions during the pre-treatment and treatment processes, followed by volume increases during the conditioning steps. Consequently, in this phase, there is an increase of radioactivity concentration due to the decrease in the net volume of the waste. There may also be a change in the radionuclide profile (e.g. thermal treatments may result in the release of some or all of the volatile radionuclides).
- In this state, the waste usually cannot be stored for prolonged periods or disposed of because it lacks a container. Handling and transport are also limited to cases, generally inside the facility.

Sometimes, no treatment is planned or performed for some types of radioactive waste (e.g. very low-level waste [VLLW]), and the waste form is the same as raw radioactive waste, and therefore there is no need for recalculation. Volume increase after treatment and conditioning is also possible, for example, cementation of liquids without previous concentration, non-pre-treated ion exchange resins or uncompressed solid radioactive waste), and this possibility should be accounted.

Also, there are examples of long-term storage and/or disposal of untreated or unpackaged radioactive waste. For instance, high-level waste (HLW) that is stored in tanks in the United States and Russia, liquid low-level waste (LLW) and intermediate-level waste (ILW) being injected into confined aquifers or discharged in the past into the sea, radioactive waste immobilised in storage vaults, VLLW stored and disposed of without package. The need for recalculation for such situations should be assessed on a case-by-case basis.

Waste package

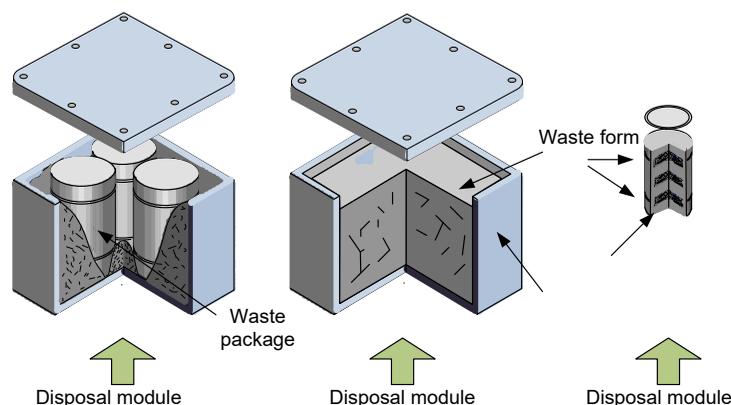
- This stage is associated with packaging and final conditioning. In general, the waste package is ready for final disposal after this step.
- There usually is a small to moderate increase in the volume due to the added volume of the container in which the waste form is placed. The degree of volume increase depends on the characteristics of the container (e.g. dimensions and wall thickness), the efficiency of packing (i.e. amount of void space that may need to be filled with buffer/backfill materials).
- At this stage, the waste package can be safely handled and transported, sometimes with the aid of a transport container to fulfil specific transport requirements. The waste package must be ready, by definition, for storage, and can also be disposed of if the waste acceptance criteria (WAC) of the repository are met at this stage.

A significant increase in the waste form volume can result due to the use of a thick-walled container as the package or when the filling occupies a great deal of space in the waste package.

Disposal module

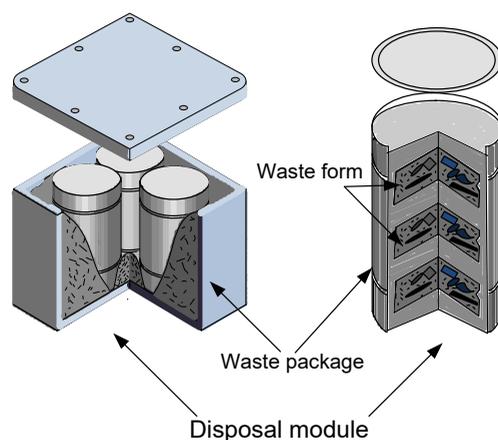
- When there is a need to additionally address the containment, radionuclide retention, heat dissipation or other issues, one or more waste packages can be incorporated into a disposal module, which can then be safely disposed of in the repository.
- In this stage, there is often a rather noticeable increase in the volume, due to the presence of multiple layers (of buffer and/or filler), and the additional disposal container.
- At this point, the final volume for the waste unit is reached, and it is possible to give an integral factor for the volume change, from raw radioactive waste to the actual volume needed to be considered by the repository.

Figure C.1. Illustration of key states of radioactive waste



Source: OECD/NEA.

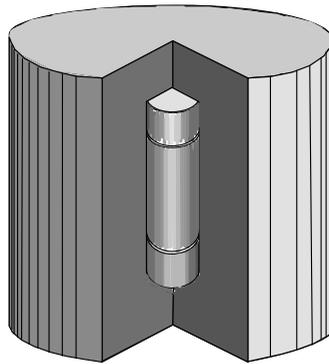
Figure C.2. Waste form, waste package and disposal module



Source: OECD/NEA.

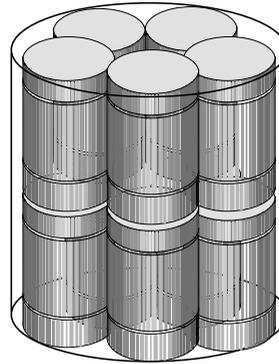
The illustration of the difference between radioactive waste volumes presented as the waste form and as the conditioned (“as disposed”) form is shown on an example of containers used for radioactive waste disposal.

Figure C.3. Example of radioactive waste volumes difference depending on the type of package



Pipe component in drum
 0.05 m^3

Drum volume: 0.2 m^3



Ten drums in ten-drum overpack
 $10 * 0.2 = 2.0 \text{ m}^3$

volume: 4.5 m^3

Source: OECD/NEA.

Annex D. Factors influencing the evolution of radioactive waste volume

Legal requirements and regulations

The first aspect to be encountered in defining a radioactive waste management (RWM) strategy, concerning the evolution of radioactive waste volume is the legal and regulatory framework of the country concerned. Whenever radioactive waste has to cross national borders during its lifetime, legal aspects have to be also considered for all countries involved in the process. These considerations apply to both treatment processes and transport. Minor aspects may also be relevant between parts of the same country if laws or local practices differ from state (region) to state. While the nuclear law is generally uniform throughout a country, occupational, environmental and economic aspects may vary, so particular attention to internal regulation may be necessary in some cases.

Some of the more important aspects have been analysed in greater detail in relation to radioactive waste volume evolution. Building expertise in the legal environment allows for more informed decisions when considering financial, economic and technical parameters of RWM. Some recommendations are more focused on certain aspects of the legal and regulatory framework, divided into two sub-categories: restriction and requirements. Restrictions are applied to radioactive waste and for each stage, but there are different limitations on certain parameters and features of radioactive waste. The requirements are focused on the practices related to the RWM strategy, whether is it a treatment facility or for transport.

Table D.1. Examples of legal factors influencing the evolution in the volume of radioactive waste

Restrictions	Requirements
<ul style="list-style-type: none"> • general regulations; • toxicity and radio-toxicity; • activity concentrations; • dose rates; • stability of waste package or disposal module; • waste acceptance criteria; • repository design preferences; • technology availability. 	<ul style="list-style-type: none"> • nuclear, radiation protection and environmental legislation; • effluents emissions; • transport routes; • safety (security) features; • transport containers; • facility location; • disposal route; • repository location.

One of the first decisions to make in determining an effective RWM strategy is to specify the level of the waste treatment and processing according to existing legal and regulatory requirements starting from almost zero level of treatment (no conditioning or just simple packaging), to increasing levels of processing (pre-treatment, treatment and conditioning). When making this decision, regulation and nuclear, radiation protection and environmental law should be reviewed in detail. Some procedures and practices may be mandatory or strongly advised while others are optional, which may provide some freedom in decision making.

For instance, two important factors are the legal requirements concerning the radio-toxicity and toxicity levels of the waste. It is necessary to note, that radioactive waste is primarily an industrial waste with a variety of hazardous properties (including non-radioactive hazards) that need to be considered since they greatly influence the array of possibilities in the waste management strategy. For example, radioactive waste could have other hazardous properties

(like conveying chemically carcinogenic particles) that impose limits to the degree of human involvement during the treatment processes; aiming for the best practice in relation to volume evolution may not be possible when considering the legal aspects, for example in occupational safety. In some cases, these requirements may impose certain practices to better protect the environment, public and the involved working personnel, which may lead to an increase of the waste volume or an increase in the complexity of the treatment technologies. Also, in some cases, it may be counterproductive to process the waste and reduce its volume if the increased concentration of the radioactivity results in a change of the radioactive waste class to a higher one and all the implications that derive from that.

Treatment in general and volume reduction in particular, not only affect the waste and its parameters but also generate secondary waste and effluents; mainly with facility maintenance as well as treatment of gaseous and liquid effluents (which themselves can become wastes as well). The radioactive waste processing facility needs to be continuously monitored to ensure that concentrations of toxic and radioactive elements in the effluent streams are kept under the regulatory limits. The limits imposed on gaseous effluents of treatment facilities by the regulatory bodies, for example, in an incinerating plant for radioactive waste, are crucial for evaluating the correct waste management strategy. Sometimes, very stringent regulations on the allowed releases of radioactive nuclides and/or chemical species may impose additional levels of complexity (such as high efficiency particulate filters, water and air scrubbers or other technologies) that may render certain waste treatment methods and processes uneconomic or even technically unfeasible.

The radiation dose, both to workers and the general public, is obviously a critical aspect of any consideration regarding RWM strategies, and waste volume recalculation is not any different. Waste treatment facilities have to address strict regulations, on various aspects, from the location of the site to the total public dose, usually expressed in man-Sieverts per year. There are two types of doses to be considered – “real” doses in the short term from the operation of a facility and calculated hypothetical future doses from a closed disposal facility. This often results in trade-off – increased real doses in the short term from increased radioactive waste treatment in an attempt to reduce hypothetical long-term doses.

Particular attention should be given to the location, as it may be easier to speed up licensing and construction times to choose a location within or adjacent to an already existing nuclear installation. However, such a location may restrict the available dose limit for the new facility if the existing facilities already make a significant contribution to calculated local public doses. When legal considerations are fully addressed, then the decision of the location can be further based on economic and logistical aspects.

Another important aspect for security, economic and legal considerations, in particular for countries with larger inventories of waste or with widely distributed nuclear facilities, is the transport of radioactive waste. The logistics behind a large waste management network are very complicated. The quantities, types and forms of materials transported between different locations need to be considered, especially when waste producers, treatment, storage and disposal facilities are in different locations. The main driving force in this scenario is financial, because of the high costs due to transport of radioactive waste. For example, is it more cost effective to have local treatment at each waste producing site, or to transport all waste to regional or central facilities for treatment? How many regional facilities are optimal in a country with large land area? How far away from the treatment facility should the disposal facility be? The first and foremost issue recommended for analysis is the regulation related to the radioactive waste transport – what kinds of waste can be transported, how must it be packaged, what forms are acceptable, what transport modes and routes must be taken, what are the size and/or weight limits, etc. In some cases, the most economically beneficial method of transport (e.g. road, rail or water) may not be available because of national regulation, or in other cases, even if it is allowed; the requirements are too stringent, compared to other scenarios, rendering that particular route too costly to be viable. The NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) recommends that considerations for all involved countries should be made if national borders of several countries are crossed during the radioactive waste transport.

Waste acceptance criteria (WAC), when established, can be a key element that the EGIRM suggests considering in any waste management strategy. WAC can be defined on a national level (the general part) or an operational level, and the specific part differs from facility to facility. The final destination for radioactive waste is definitive disposal and WAC have to be met in order for the waste to be disposed of in that facility. In addition, when the waste is in interim or extended storage, WAC have to be also considered for storage facilities, and not only repositories. Transport requirements may also influence the WAC, and in some cases may be more stringent than the pure disposal requirements (e.g. surface dose rates or waste package sizes/weights for transport over public roads may be more restrictive than limits acceptable at a repository).

WAC also needs to be considered for any treatment facility, since not all facilities can accept all types of waste. They can be called different things, but in principle, some types of acceptance requirements are always used to establish the physical, chemical and radiological limits for treatment in a given process. If treatment, storage and disposal facilities are already constructed or operated, there is usually not much room for decision making, as the WAC are already well-defined, and the waste has to match such criteria. Treatment resulting in volume changes may not be feasible if there is the impossibility to meet the requirements given by the WAC, should some parameters change and become incompatible during the treatment. In some cases, though, these facilities have not yet been implemented, which gives the opportunity to design the system and define WAC to suit the entire range of waste that requires treatment. This is an opportunity for all stakeholders to set limits and ranges on crucial parameters from a joint discussion, involving waste producers, treatment and conditioning facility operators, storage facility operators, and final repository operators. In this situation, a country has an opportunity to apply the holistic approach making decisions to the development of the national RWM system.

Economic factors

After a clear understanding of the legal basis is obtained, it is recommended to consider economic factors that can influence the technology choice. Volume reduction and volume evolution, in general, are also driven by economic considerations, always keeping in mind the safety and stability of the waste, so when choosing a path for a waste management strategy that includes waste volume reduction, the economic benefits of such practices should outweigh the additional costs and complexity introduced in the processes. The economic benefits in waste volume reduction are all related to reduced volume: less space is needed for storage and disposal, less transport is required, and at a psychological level the total volume of waste is one of the topics for the public that has the most impact. However, for economic viability from the radioactive waste producer's point of view, the *total* cost of processing should be less than the *incremental* cost of storage and disposal space saved. It is false economy to compare the cost of processing against the average cost of transport, storage and disposal. It is only the incremental cost (i.e. the cost to transport or provide space for an extra cubic metre, which may be far less than average cost per cubic metre of the saved space or transport) that will be saved by any reduction in the volume.

The following topics are discussed further below. However, it should be noted that this is not a full list of potentially relevant topics. All factors need to be considered in conjunction with each other – they cannot be considered each in isolation since they can all influence each other in different directions:

- total volume of waste;
- yearly production rate;
- available infrastructure;
- energy and primary resources;
- labour costs;
- storage and disposal route;
- transport costs;
- tariffs and financing structure;
- safety features;

- waste classification boundaries.

One of the most important aspects to be considered during the economic evaluation of a volume reduction practice is the total volume of waste that has to be treated, and the yearly production rates of radioactive waste. It is probably not reasonable to undergo an additional process for volume reduction if the process is of high cost and the total quantity of radioactive waste is very low, and there are no expectations for this quantity to rise in the near future. On the other hand, even if the present quantity of radioactive waste is rather low, but there are planned activities that will produce an increase in the production rates of radioactive waste, for example, a new nuclear power plant (NPP) under construction, or an older NPP starting decommissioning, would, in this case, be economically viable to plan and invest in volume reduction early on. Countries that can benefit the most from volume reduction processes are obviously those that have large amounts of radioactive waste waiting for a solution. Nuclear activities and a waste management strategy require extensive planning, often at a national level, so it is possible to rely on future prediction to make estimates of the economic viability of volume reduction treatment with a reasonable degree of confidence.

When evaluating costs for radioactive waste treatment strategies, the current level of development of the infrastructure in the surrounding area should be considered. This is very technology and country dependent, as different treatment technologies require different types of facilities, mostly not interchangeable, and different countries have different preferences and capabilities. There are some types of technologies that are well developed and where the industrial process is already used in other sectors and is very well understood. One should consider what technologies are already present and developed in each country, as this may significantly lower the prices of new installations, maintenance and staffing. In other cases, as when there are no “bridge” technologies or solutions which one can take example from when designing new waste treatment facilities, one has to decide whether to go with the more cost-effective solution, like a simple technology but with a limited use and limited results, or if to go with a more expensive solution, which allows greater volume reduction, or achieves better technical results for a wider array of waste typologies. This decision cannot be made in isolation. The downstream effects (such as required storage and disposal capacity and its cost) also need to be considered.

When considering technologies and waste management strategies in general, other than the installation and maintenance costs of a facility, a crucial parameter is energy consumption. Some processes are very high in energy consumption (such as plasma torch incineration), thus requiring a detailed analysis of the energy consumption as part of the operating cost estimate. If the facility is situated in a remote area or is not connected to a sufficient power source on the main electrical grid, expenses due to grid installation or upgrades must also be considered.

Labour costs can influence a choice of technology. Where labour costs are high, labour intensive processes, such as sorting, monitoring, manual decontamination, etc., as well as those processes which require them upstream or downstream, may become less cost effective. This part of the economic evaluation is very dependent on the local labour market, and legal requirements and its impact will be different in different countries, or even in different regions of the same country. Labour intensive processes may also result in higher worker doses.

Probably the most important economic parameter to take into consideration is the cost of the disposal and storage facilities. The whole idea of waste volume reduction is to lower the cost for disposal and storage sites, by requiring smaller facilities or in fewer number. This consideration is heavily dependent on the type of disposal facility (storage facilities behave more uniformly in this sense), as costs for such instalments depend on the total volume of waste they can accept, the depth of the facility (strongly influenced by the class of waste it will dispose), and the level of the engineered barriers present. For example, deep geological repositories, deep geological repositories (DGRs), may have a higher average cost per cubic metre disposed of than a near-surface facility, or for a borehole configuration. However, what is usually important in the cost calculation is not the average cost, but the incremental cost, i.e. how much does it cost to provide the space for an extra cubic metre. In the case of a DGR, this can be relatively low, since a high proportion of the cost is “fixed”, whether you dispose of a hundred cubic metres or a thousand cubic metres (e.g. the cost of the infrastructure, such as shafts, hoist, ventilation). The impact of this will be governed by how fees are charged and allocated in a given country and who

controls them. For a near-surface disposal facility, the average and incremental costs may be very similar due to modular construction and lower initial development costs.

The degree of implementation of the engineered barrier system (EBS) also contributes to the total costs of the storage or disposal facility and determines what class or type of waste can be disposed (for example heat-generating waste requires specific considerations in repository design). It is more convenient to reduce volumes of waste that will be disposed in facilities where the incremental cost per cubic metre is very high, rather than to focus on volume reduction for waste that can be disposed in facilities with low incremental costs. For countries without existing treatment or disposal facilities, this trade-off should be carefully considered before deciding on a disposal or treatment facility design. Investing in a disposal facility with low incremental costs rather than a treatment facility with high volume reduction may be a more cost-effective solution. In the case of high volume reduction, care must be taken to remain in a specified waste class when treating waste to be disposed, especially in a near-surface facility; if volume reduction causes an increase in the radiological class of treated waste, this could force disposal in a different and more expensive facility, negating the benefits of disposing a lower volume of waste by the increased costs of disposal per cubic metre.

Transport costs usually do not make up the largest part of spending, but they must be considered. Particular attention must be given to larger countries with distributed facilities throughout their territory. Waste volume reduction, in this case, can provide even more benefits when the treatment occurs at the waste generating facility, reducing the volume needed to be transported or shipped, thus lowering costs. However, if all wastes need to be transported to a centralised facility, this transport benefit will not be realised but the overall cost of building one central processing facility (or several regional ones) may be less than building separate facilities at each waste generating site. If waste producing facilities or repositories are placed in isolated areas, this has to be taken into consideration when deciding the site of treatment facilities. An optimal solution should aim for a well-connected location that keeps as low as possible the total travel distance per cubic metre.

Fees and tariffs also have to be included in an economic evaluation of a waste management strategy. Other than the actual costs for the implementation and operation of disposal and storage facilities, there could be additional costs due to tariffs, taxes or other fees, imposed by the government of the hosting country and/or local area. These additional expenditures, how they are calculated and who pays them, should be carefully analysed as they could be seen as a tax on the disposed waste and, may be relevant to the total costs. There could be cases where a volume reduction process is even more beneficial and economically viable due to high tariffs to dispose of such waste if they are based on volume or mass. In some countries, such as the United States, disposal fees and tariffs are based on a combination of volume, mass and activity, so reducing volume may not necessarily reduce the tariff, and in some cases, can actually increase it.

Radionuclide retention and safety features are paramount in a waste management strategy. It should not be forgotten that the waste package, or the disposal module unit, acts as the first barrier to radionuclide release and to prevent water and solvents from reaching the radioactive material. In a repository, there are normally a series of barriers that prevent unwanted processes from happening, and the operators must demonstrate through a safety assessment that the required level of safety and confidence is achieved. How and through which mechanisms this is achieved is normally left in part to the repository operator's design decisions. Together with the second barrier, the buffer and/or backfill, the waste package forms the main part of the EBS. The third barrier is the geological formations (geosphere) in which the repository is situated. All three barriers have their specific roles and safety functions, which vary between design concepts and from site to site. Some of the roles and safety functions may overlap, providing defence in depth. When considering waste volume evolution, there could be cases where a highly engineered external EBS (i.e. other than the waste package) or high-quality geosphere allows for a less engineered waste package (e.g. thinner waste package walls, lower requirements on conditioning or higher waste load factors), thus reducing the total volume of disposed waste. It is possible if the external EBS or the geological formation provide enough shielding and isolation to meet the safety requirements of the whole site. On the other hand, there could be cases where higher levels of external EBS could be too costly or otherwise not practical, making it more economical to shift more of the isolation and safety requirements directly on to the waste package or disposal

module, even if this means ending up with increasing volumes of waste, due to the thicker package walls or lower waste load factors.

Technical factors

After the all previously discussed considerations have been made, the technical aspects of radioactive waste volume evolution should be reviewed. Note that some of the technical factors will influence the economic choice and vice versa. It is therefore important to have a good understanding of the technical parameters of the waste before making a final economic choice. General recommendations on the selection of a particular strategy or process for the treatment and conditioning are given in this section. More details concerning the treatment technology selection and their main characteristics can be found in the publications cited in the references section of this report. In this section, the general group of parameters and aspects is discussed; it influences technical decisions on a general level, valid for a wide range of technologies. The review focused specifically on a set of technologies that are widely implemented, such as cementation and vitrification, starting with cementation. Methods like incineration, evaporation, sublimation, etc. are considered to be pre-treatment technologies because they generally do not produce final waste forms and require further treatment or conditioning prior to disposal. First, the EGIRM experts recommend reviewing the following waste parameters:

- the half-life of the main nuclides;
- the total volume of waste;
- total activity and activity concentration;
- radiation type, intensity and decay chain;
- physico-chemical properties of the raw waste and required properties of the final waste form;
- radioactive waste class both the raw waste and the final waste product (in accordance with national radioactive waste classification);
- potentially disruptive disturbing (limiting implementation) components (i.e. ones that may interfere with or otherwise limit the implementation of a particular technology);
- other less critical factors.

The half-life ($t_{1/2}$) period of radionuclides decay is one of the key parameters to define and classify radioactive waste. Very often countries divide radioactive waste classes primarily by this parameter, and the International Atomic Energy Agency (IAEA) GSG-1 classification is no exception. Evaluation of the volume evolution, should take into consideration the main radionuclides present and their half-life periods (i.e. how long they will need to be contained for). Usually, for radionuclide of shortest lifespans (very short-lived waste, VSLW), the best option is decay storage where the waste is kept in storage long enough to reduce its radioactivity, hence allowing for a controlled release. This is only convenient for certain types of waste, and in this case, it is usually not worth to undergo any additional process to reduce the volume of waste since the storage period is very short. The only exception to this may be in the case of high volumes of waste where it is not practical to construct larger decay storage facilities. On the other hand, the longer the half-life, the more beneficial it may be to process the waste with the aim of reducing its volume, facilitating the handling and allowing for more efficient use of storage and disposal facilities. A difficulty often encountered is when there is a mix of radionuclides with different half-lives and different radiation types. It is generally not possible to separate individual radionuclides from a waste stream. However, in this case, it may be beneficial to segregate wastes or waste streams based on the half-life of their major radionuclides, in order to be able to proceed with the practices previously mentioned. Typical radioactive waste from NPP contains a wide mix of long-lived and short-lived radionuclides. Research and medical facilities, on the other hand, may have waste streams with single nuclides or a very limited range of nuclides, which are easier to keep separated. Disused sealed sources are typically composed of a single radionuclide.

The total activity and/or activity concentration is a crucial parameter for calculations, as it determines the radiological class that the waste fits in. The total activity usually does not change a great deal during the volume changes, what really changes is the concentration of radioactivity in the considered material. Some exceptions are possible, for instance, volatile species (e.g. tritium, carbon-14, iodine and other gaseous nuclides) which may be released from the wastes during thermal treatment. The general intention is to concentrate and reduce waste volume, but mostly up to the limit of the radiological class the waste started in: generally, the waste producer prefers not to increase to higher waste class as this brings numerous problems with it (especially for low-level waste [LLW] and intermediate-level waste [ILW]). For high-level waste (HLW), increasing the activity concentration is limited by other parameters such as heat load, not the radiological classification. Distribution of the radionuclides in the waste form is also essential. When the activity is concentrated in a small and well-defined area of a portion of the waste, (which could be a part of the waste, the superficial layer or a particular phase of the mix) then volume reduction may be even more advisable, and depending on the nature of the distribution the right process to proceed can be found. There could also be the case where the waste is in a liquid or semi-fluid state, and the activity is distributed evenly in the waste form. In this case, there are plenty of solutions to separate the relevant radionuclides from the rest of the waste form (mainly by reducing its water content). However, some of the processes can be applied only for small-scale implementation.

Radiation type, along with decay times, is an important parameter to define and classify radioactive waste. For waste classification, in general, the distinction is made between alpha-emitters and other types of radioactivity. Alpha-radiation poses a greater hazard to human health in terms of ingestion and inhalation risk, while even if beta and gamma radiation are very different from each other, they can be treated as equivalent in most cases. Gamma (or X-ray) emission often accompanies other types of radiation. Decay chains must also be evaluated in detail as there is a need to evaluate the chemical and radiological behaviour of each of the possible daughter elements (e.g. they may form voids or other disruptions in the structure of a waste matrix), and how they could interact with the waste matrix and the environment in general. For example, certain decay products may have a very different solubility in the waste matrix than their parent element, and this should be considered in the safety case of the waste treatment process. It could be the case that a higher volume reduction coefficient is not achievable because of the different solubility of decay products in the waste matrix (for example concrete), and it is necessary to reduce the load factor or increase the thickness of the concrete buffer. Also, the determination of the contact dose for the exposed personnel is important, in which radiation type plays a fundamental role; alpha-emitters do not have the penetration to reach outside the waste package and for this reason contribute less than gamma or beta emitters in the total external dose. However, alpha-emitters will contribute significantly to the internal dose of the waste package and may cause damage to the matrix over time if present in too high concentration.

The physio-chemical properties relevant to waste volume recalculation will be covered more in detail later in the review of the more widely implemented treatment technologies. First, the physical state of the waste should be defined: solid, liquid or maybe in a viscous paste (gaseous radioactive waste is out of the scope of this review as it does not apply well to a volume evolution discussion). Liquid waste can be more easily moved and treated, but it should then be solidified in order to be stored for long periods or disposed. (For liquids, a crucial task is to find an immobilisation matrix that is compatible). Solid waste, on the other hand, may need to be either chopped into small parts or incinerated in order to receive proper treatment. Usually, cement is the first option for an immobilisation matrix, as it is a well-studied and relatively cheap material. But there are many cases where ordinary Portland cement (OPC) does not meet the requirements of standards, and so either additives have to be considered, or a completely different immobilisation matrix, such as polymers, ceramics or glassy products considered. The following physicochemical and structural properties are recommended for consideration:

- compressive strength of waste form (should be higher than certain thresholds, as many types of waste package also have structural requirements);
- leachate resistance of waste form (to water and other solvents), ageing resistance (depending on the expected lifetime of the waste package and/or disposal module);

- corrosion resistance (for the waste canister and for all corrodible material inside the waste canister);
- waste form thermal expansion coefficients;
- tendency of waste form to chemical and biochemical degradation;
- impact resistance of waste form (in case of accidents or unplanned events);
- radionuclide sorption (for the main radionuclides present in the waste form);
- gas generation (of both flammable, non-flammable and explosive gases).

The relative importance of each of these will depend on the required properties of the waste form with respect to the safety case for storage or disposal.

Radioactive waste classification is a subject that spans over multiple areas of interest. There are some key issues relative to technological aspects. All technologies that apply separation of the different radionuclides species and/or phases inside the waste can be reviewed. The possibility to sort an initial volume of waste containing mixed radionuclides into separate volumes of wastes each with more defined radiological classes and properties may be of crucial benefit to the whole treatment processes. When such a separation is not possible or is only partially effective, the subsequent treatments may suffer from the fact that a higher volume of waste will be categorised and treated due to a small percentage of longer-lived radionuclides, with a more toxic nature, or that are just harder to handle. Unfortunately, this is normally the case, especially for NPP wastes where the separation of radionuclides from the wide variety of waste types is a complex and costly process. The ultimate treatment in terms of volume change is possible after the complete separation when each type of waste or radionuclide is managed separately.

With respect to potentially disruptive agents, the reference is to a category of chemical compounds and physical elements that are the part of waste and are deleterious to a particular treatment process or to a particular compound or material that is introduced in such processes. In some cases, there are specified limits under which these substances are not considered harmful. Sometimes there are no limits, and the reasonably lowest concentration is needed. For these cases, in order to pursue the best conditions possible with reasonable costs, an optimisation of the technological process can be performed. Disruptive agents can also undermine important safety features of the waste form or the waste package like corrosion resistance, structural integrity or radionuclide solubility. They influence certain properties of the immobilising agent or provoke secondary release or contamination (e.g. chelating compounds). Considering waste volume changes, one should pay close attention to these compounds or elements and to their changes in concentration during volume evolution. In some cases, treatment processes may create disruptive agents in situ (e.g. degradation products from cellulose) and/or disproportionately increase the concentration of certain disruptive agents even if volume reduction coefficients are lower than achievable for waste without such agents. Each case should be analysed individually since some chemical compounds reach dangerous concentrations only in certain geographical areas or process conditions, being dependent on the natural materials used during treatment. For example, if cement is used as a stabilising matrix then in some areas the building materials, like gravel, sand or clay, could introduce high concentrations of unwanted compounds, but in other areas, this could not be a problem because of the different chemical composition of the materials.

Waste management practices connected with volume changes

Earlier, a general overview of the factors influencing the waste volume recalculation has been done following the logical scheme presented previously. In this chapter, more in detail analysis has been made concerning the more widely implemented technologies.

Radioactive waste processing technologies have been studied many times, and relevant publications provide the detailed technical information about them; the recent publications are listed in the report. In this work, experts limited the discussion to only widely implemented

approaches and used them as examples of how to approach the volume recalculation based on knowledge of technology.

Cementation technology is the most used method for radioactive waste conditioning, one of the more versatile and well known. It is also the most practical method where price and efficiency are acceptable and attractive for implementer of any size. However, it is necessary to note that cementation technology cannot normally be considered as a volume reducing method; more often the final waste volume will exceed the initial one.

Vitrification technology is implemented mainly in countries which have the strategy of spent fuel reprocessing. Some of them have decided against spent fuel reprocessing in favour of direct disposal, some countries continue or plan to reprocess spent fuel and accordingly employ vitrification technology. Vitrification is first considered to be a method of liquid HLW treatment. However, some countries (e.g. Russia, Korea and the United States) have studied the application of vitrification for the treatment of ILW and even LLW (both liquid and solid).

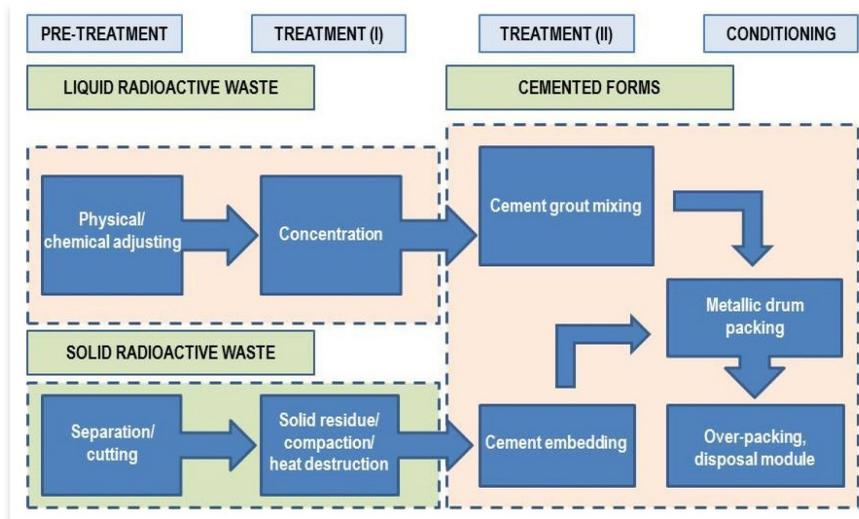
For both cementation and vitrification, EGIRM experts followed the volume evolution from the starting raw volumes all the way to the final disposal volume, specifying important steps in between.

This part and the rest of the analysis is intended not to derive a cumulative multiplication factor, but to provide the user with the key insight on the most important issues in waste volume recalculation.

Cementation

A visual overview of the technological steps involved is given in Figure D.1 as well as how they relate to each other. The radioactive waste physical state (solid or liquid) should be accounted for first as a factor defining the waste volume evolution. For clarity, the waste treatment itself can be divided into three main stages: pre-treatment, treatment, and conditioning. Treatment can be further sub-divided into two sub-stages. This division is representative for the purposes of this discussion and may vary, sometimes with other definitions of treatment and conditioning in national and international literature. It is only used here to give a logical order of the process; it does not relegate a given practice to one or the other stage and should be considered accordingly.

Figure D.1. The main stages of the cementation process



Source: OECD/NEA.

Physical and chemical adjustment

The adjusting of chemical or physical waste properties can be a pre-treatment step in waste processing. It is essential for liquid waste, but it can also be done for solid waste, such as solid residues and certain types of sludge. This pre-treatment can be applied for preparation for further treatment of large and smaller liquid waste streams. Sometimes, chemical adjustment is not necessary, and the waste can go directly to treatment or conditioning, but often chemical adjustment is implemented, at least to a lower degree. Adjustment can be applied to both aqueous and organic liquids, with separate considerations to be made for each waste stream. It is normally important to separate as fully as achievable liquid waste when it is a mix of organic and aqueous solutions and to keep them separated for the rest of the treatment processes. Sometimes, a high-shear mixing to emulsify or completely amalgamate the phases can be a more cost and technically effective solution. However, the ability of the matrix to immobilise both liquids should be evaluated.

There are a few aspects that are more frequently targeted during this process: pH, chemically aggressive agents, flocculating agents, dispersed fine particles, and in general, any component that could interfere with the cement mixing, pouring, setting and hardening. These values should be within certain ranges in order to allow the use of OPC or other less expensive and better-known cement typologies. Without implementing these procedures, the use of special cements may be necessary with consequent costs increasing when waste parameters do not address the requirements. It is also an economic decision what to select – using more intensive chemical adjustment now allows the use of cheaper cementitious materials later, or skipping this phase requires the use of special or significantly modified cements instead. Filtration can be an adjusting method should the liquid solution be contaminated with solid particles of different size. Techniques exist to execute filtration depending on various factors. However, in most cases, it is more technologically and cost effective not to filter and treat the liquid and particulate as a single waste stream rather than dealing with each phase separately.

Adequate (high) pH levels are necessary for cement setting and hardening as these processes require high alkalinity in the waste form which should be properly accounted. In general, pH adjustment is often made for highly acidic waste streams. It is usually technically simple and a not very expensive procedure, time-consuming or requiring a very complex infrastructure.

The pH also influences the solubility of certain compounds and molecules that might cause problems if they precipitate in later stages or are not soluble enough in the liquid cement slurry. Regulating of the pH of liquid waste usually does not cause volume variations, but it is essential to make feasible further treatment procedures that operate optimally in the limited range of the pH values.

Removal of flocculating agents can be also made at this point or at least their activity can be severely mitigated. The restrictions for flocculating agents' concentration vary according to the cement type used for conditioning and mainly related to the ratio of water to cement. Anyway, flocculating agent content should be kept at a minimum since their removal is an easy and standard task, which can be done through filtration, chemical precipitation or absorption. For any solids removed from liquid waste, separate treatment and conditioning will be required and may result in an additional volume increase. Therefore, a decision on the separation should be carefully considered at the overall process optimisation stage.

Sorting, segregation and shredding

Sorting and segregation is a commonly used pre-treatment step for solid waste as it does not require very complex equipment or facilities but can be of a high technical benefit. This practice is somewhat useful for waste from decommissioning sites as it is usually of various size, shape, activity content and typology. However, it is generally labour intensive, and its effectiveness will depend on the cost of labour. It can also be dose-intensive for workers if it is not designed properly. The level of sorting and the cost of associated technological operations should also be considered in the overall economics. There are cases when waste is already very well segregated and characterised, and it is not necessary to apply further sorting and segregation. Sorting is

especially rewarding when there are parts of the waste with higher activity levels and concentration, and by removing these parts, the bulk radioactive waste volume can be shifted to a lower radiological class, while the separated parts of greater activity will have their volume significantly reduced. In general, this operation facilitates all the subsequent treatments and procedures. There is, of course, no real volume change in the waste flow after sorting, but a part of the waste stream can be categorised as a lower category at this stage, even as exempt waste, and, thus, excluded from the total volume of the radioactive waste.

Shredding, or cutting, is a pre-treatment operation with potentially very high volume-reduction factors. It is viable when the waste has a significant percentage of voids or empty spaces that heavily impact on the total envelope volume of the batch. In this case, shredding can compact the waste, even if a compaction treatment is applied further on for better results. Solid radioactive waste which is considered to be a brittle and non-uniform material is well suited for shredding and cutting. The hardness and the structure of the material are very important as it determines the machinery and the procedure to use for each type of waste. The percentage of void space is also crucial, because it may not be reasonable to cut and shred a waste stream with a very low percentage of voids. However, as a pre-treatment for other processes, such as incineration and other technologies, it may provide some benefit, even for softer wastes such as plastics and cellulose. Once the material is shredded to pieces of the desired size, it can be further processed, or it can be directly sent to cement embedding. The other option of high void waste treatment (> 50%) is to consider the possibility to fill only the void spaces with cement to obtain the waste form (e.g. discarded equipment such as tanks, heat exchangers.) It can be an option that eliminates the need of shredding and allows for good space economisation; the downside is there a need to use very fluid cement mixes, achieved either with high water to cement ratios or with the use of fluidisers and superplasticisers. Using these additives, a balance should be provided between sufficient impregnation rates and the structural properties of the resulting waste form. Sometimes, the use of superplasticisers is problematic because of the complications it brings with other parameters, like the radionuclide leach rate. Also, in this case, the cement mixture could be made from liquid radioactive waste, providing further optimisation of space usage.

Concentration

Usually, liquid radioactive waste treatment procedures start with evaporation and/or concentration with some techniques that increase the concentration of contaminants in liquid waste streams. At this stage, the highest volume reduction factors of the whole process can be reached due to the nature of the treatment. Water is extracted and, if it complies with the release criteria, can then be released freely in the environment. Possibly the main exception to this is tritiated wastes. The tritium concentration in the separated “clean” water will be the same as in the original liquid. These practices can reach volume reduction factors of about 2 – 1 000, depending on the starting scenario and the processed radioactive waste features. A summary of the different technologies used for concentrating liquid waste is given in Table D.2.

Possibly the most important parameter for evaporation and concentration practices is the presence of salts in the liquid waste. Volume reduction increases the salt concentration of the solution. Unlimited volume reduction is impossible, and the practical limit would be defined depending on diverse considerations, such as solubility limits for individual chemical species. In any case, for liquid waste, a certain amount of water has to be maintained in order to hydrate the cement. The actual quantity depends on the required water-cement ratio, and it is not reasonable to evaporate or concentrate the liquid waste stream beyond this point. The water to cement ratios vary and have a wide range of applicable values, depending on what type of cement is used and what additives are introduced and what properties the cement waste form is required to have. For some limiting values of salt concentration, it depends on the nature of the salts, but for a general understanding, a limit of 700 g/l can be given as a reference, while the more technically proved value is considered to be around 500-600 g/l. In this way, the maximum salt concentration in the cemented waste form is around 20-25% by weight; one can achieve higher values, but then other factors become too much of an issue to handle. The presence of salts in the cement matrix is cumbersome for mainly two reasons: it hinders the hydration of the cement and so lowers the mechanical strength of the matrix; and it may also

lead to the increased radionuclide solubility in the pore water, favouring higher leach rates. After concentration of the waste is completed, further treatment of either a liquid or a solid waste stream can be started, depending on the level of evaporation. When radioactive waste is concentrated but still liquid waste, it can be followed by liquid cement mixing, and while radioactive waste is a solid residue of various nature, the process is followed by solid residue treatments and then with cement encapsulation.

Table D.2. Methods for concentrating radioactive waste

Technology	Features	Disadvantages
Evaporation	<ul style="list-style-type: none"> well known and implemented technology; high reduction factor; suitable for the majority of; radionuclides; very high decontamination. 	<ul style="list-style-type: none"> implies complex process limitations: scaling, foaming; corrosion, high volatility of some radionuclides; high operation and capital costs; not suited for highly tritiated wastes.
Solvent extraction	<ul style="list-style-type: none"> very high flexibility; allows for the recovery, removal, or recycle of treated radionuclides. 	<ul style="list-style-type: none"> results in large quantities of organic material present in the aqueous raffinate solution; generates different types of; secondary waste (aqueous and organic); complex chemistry, especially for mixtures of nuclides.
Organic ion exchange	<ul style="list-style-type: none"> good resulting mechanical properties; re-generable; positive outcomes with low salt concentrations. 	<ul style="list-style-type: none"> limited physical and; chemical stability; high resins cost; immobilisation difficulty.
Inorganic ion exchange	<ul style="list-style-type: none"> improved physical and chemical stability compared to organic ion exchangers; allows for a fairly easy immobilisation. 	<ul style="list-style-type: none"> susceptible to high salt concentrations; regeneration and recycling may be complicated; high costs.
Reverse osmosis	<ul style="list-style-type: none"> high decontamination factors; cheap; removal of dissolved salts; easily scalable. 	<ul style="list-style-type: none"> non-back washable, subject to blockage; very high-pressure system; limitations imposed by osmotic pressure
Membrane filtration	<ul style="list-style-type: none"> very flexible on particle size; high separation factors; easily scalable. 	<ul style="list-style-type: none"> complex installation; varying costs; requires pressurised equipment.

Compaction

Compaction is one of the more common treatment methods for volume reduction of solid radioactive waste. It often requires a certain degree of pre-treatment in order to operate at the maximum efficiency, but some types of waste can be compacted just as is. Compaction requires that the waste be of a size to fit into the container. Therefore, the most common pre-treatment is shredding or segmentation of larger waste items. It is one of the most useful technologies in volume reduction, and it can be divided into two subclasses: low-force (about 5 tonnes) and high-force compaction (up to about 5 000 tonnes). The main differences are explained in Table D.3.

To summarise the table, and to give a general picture to the user, we can say that low-force compaction is cheaper, faster to install but with limited volume reduction factors, while high-force compaction is costlier to install, maintain and train qualified personnel, but produces higher quality waste forms and with greater volume reduction factors. Usually, there is also a minimum quantity of waste production or stored waste in order for high-force compaction to be economically viable. The most common procedure is to collect the waste in 200-l drums and then to compact the drums into compressed metallic “pucks” as they are commonly called. The resulting pucks can then be placed into larger over-packs, such as 400-l drums, where they will be conditioned and packaged. The high pressing force will also tend to squeeze out any residual

liquid or moisture in the wastes (even seemingly “dry” wastes will contain some moisture) and will require collection, further treatment and conditioning before disposal.

Table D.3. Main features and advantages of radioactive waste compaction methods

Technology	Features	Disadvantages
Low-force compaction	<ul style="list-style-type: none"> • low investment costs; • the possibility of mobile units; • low maintenance; • easy to operate; • waste is compacted into drums or other containers. 	<ul style="list-style-type: none"> • relatively low volume reduction (3-5); • more dependent on pre-treatment; • feasible only for “soft” radioactive waste.
High-force compaction	<ul style="list-style-type: none"> • good reduction factors, up to 100; • the high quality of resulting waste form; • feasible also for “harder” radioactive waste streams; • the original waste container is crushed as part of the process. 	<ul style="list-style-type: none"> • high investment costs; • costly maintenance is required; • liquids are squeezed out of wastes and require separate treatment; • secondary container (overpack) is required to contain the crushed primary containers.

The final volume reduction really depends on the strength and hardness of the waste, the presence of voids and the level of pre-treatment it has received; for example, ashes and rubble, being more compact from the start have a lower reduction factor than scrap metal or glass, which is more brittle and presents more void spaces. There is also the option to add filling material before compaction, and while this will lower the reduction factor (this buffer/filler material should ideally only fill in the voids that would not be compressed anyway), it will also produce a more mechanically stable waste form.

While very hard to achieve perfect compaction with no voids left, it could also be counterproductive for those waste typologies that exhibit a certain level of gas production. This could be due to corrosion of metal products, to hydrolysis of residual water, or to biodegradation of the organic compounds present, so almost all of the common waste streams could be affected. If there is evidence of gas producing mechanism inside the waste and there is no release mechanism for the gas to escape, the best option is to leave a certain percentage of voids inside the waste form in order to lower the maximum possible pressure inside the container. This feature is easily countered by installing a vent in the waste package, or by not welding the seal of the waste package, thus allowing for some of the gas to slowly migrate out.

“Spring-back” is also an issue with some types of wastes, e.g. some plastics and rubbers. These wastes may not be mechanically stable in the compacted form and may try to return to their uncompacted shape. This may cause some degree of re-expansion in the compacted waste form, creating uneven puck shapes (for high-force compaction) or (bulging containers for low-force compaction).

Another added benefit of this treatment method is the minimisation of secondary waste produced during compaction. However, any secondary waste that is produced (e.g. air filters, liquids from high-force compaction, waste from equipment maintenance) has to be both treated and taken into consideration.

Thermal destruction

With thermal destruction, waste undergoes combustion or oxidation in order to alter its composition and, in particular, for the purposes of this study, to reduce its volume. Standard techniques are, for example, incineration and pyrolysis. The initial focus is on radioactive waste incineration in this report. First of all, this treatment method can be applied to liquid, solid and a mix of both radioactive waste streams in the required proportions. Liquid waste (contaminated oil, organic solvents, etc.) can be burnt simultaneously with solid waste in a

weight ratio solid/liquid depending on the calorific value of each waste type (a typical weight ratio solid/liquid is 3 to 1). A moderate amount pre-treatment is sometimes required; the most important is separation for solid radioactive waste to separate combustible and non-combustible radioactive waste. Also, some concentration methods for liquid combustible waste can be applied to achieve the required calorific value for optimal combustion. Shredding of some materials may also help to promote efficient radioactive waste loading and combustion. It is possible to incorporate some amount of non-combustible radioactive waste in the incineration process, but this should typically not be higher than 5% of weight. (Some high-temperature incineration processes, such as plasma processing, can handle much higher levels of non-combustibles in the waste. In these cases the non-combustibles are melted and incorporated into a slag-like or glass-like residue).

Radioactive waste incineration is normally organised as a continuous process. Hence, its throughput calculated in kg/h or kg/d should be sized to match the production rates of radioactive waste to maximise efficiency. Batch wise incineration processes can also be used for smaller volume requirements. Both mobile and fixed facilities are a common solution, depending on the volumes to be treated. The volume reduction factors are very high and can reach factors of up to 120 (sometimes even higher for liquid wastes) in optimal conditions. The added benefits of this practice are the destruction of organic compounds that may also contain biologically hazardous waste, and the production of residual ashes that are incombustible, with good thermal properties and very stable. There are some limits on the activity concentrations that should not be exceeded, in particular with alpha bearing waste the maximum advised concentration should be lower than 10^{10} Bq/m³ (as a mean value). There are also very strict requirements and precautions made on the gaseous effluents produced during combustion; excess air has to be carefully adjusted, based on the state of the waste (solid, liquid or a mix), and the calorific content. After leaving the combustion chamber, the gases go through an afterburner and a series of wet and/or dry scrubbers, passing as a final stage through a high efficiency particulate air filter, specifically designed to capture any residual particle or radioactive compounds. The filters and scrubbers also serve to retain and separate toxic molecules and do not focus solely on radioactivity control. The residual ash (from the primary chamber, secondary chamber and filtration systems) is collected and sent to either a cementation facility where it will be cemented, possibly also with other waste streams, sent to a compacting array, in order to further reduce its volume, or directly packaged, usually in metallic drums or concrete containers. While they are generally recycled for some time, the scrubber solutions or dry compounds, as well as spent filters, will eventually also require further treatment and/or conditioning as radioactive wastes.

Conditioning

EGIRM experts have analysed radioactive waste conditioning practices focuses on the use of cement matrixes. Cement is the most used immobilisation agent in relation to the radioactive waste management due to its well-known behaviour in almost all possible case scenarios, its good mechanical properties and its comparably low cost. There are numerous different types of cement and concrete recipes, each suited for a range of applications. However, it is necessary to differentiate the conditioning processes based on the level of homogeneity of the resulting waste form and waste package. The conditioning process can be obtained with a homogeneous waste form matrix (with liquid or fine particulate waste dispersed in the matrix) or as a heterogeneous waste form matrix (if larger pieces of solid waste are loaded and encapsulated in the cement matrix). It can be a combination of both if the solid waste is embedded by a cement mix that prepared with liquid waste. When selecting the process to implement, the solution is relatively simple for only one type of waste stream, either solid or liquid. However, in case of conditioning of both liquid and solid waste, there is the option to mix both waste streams to produce only one final waste package. This option is usually preferred as it is the solution with the best-combined results for solid and liquid waste streams. Finding the optimal balance between the two waste streams is an ongoing policy challenge.

Liquid cement mixing

Apart from the nature of the waste conditioned, there also is a series of possible mixing technologies that define the quality of the conditioned waste form, the final total volume, and the total volume of waste treated per unit of time and consequently the cost of the operation. One of the possibilities is batch style in-container mixing. Waste and the cement grout are loaded into the container and then are mixed and left to harden. This can be done either with a normal stirring technique, where the stirrer is removed after the operation or with a lost-stirrer technique where the stirrer is left inside the container with the waste. The main difference between these two methods is the more complicated facility requirements to clean the stirrer and to ensure the minimum possible contamination. The main issue with the lost-stirrer operation is the added volume inside the waste container that lowers the overhaul loading factor and introduces additional cost. There are also other mixing techniques where a dedicated external mixing vessel is used for this purpose, and after the waste and the cement grout have been properly agitated and homogenised they are poured in the waste container to harden. As with the removable stirring option, cleaning systems need to be provided for the mixing vessel. In both cases, the cleaning water can be retained for use in the next batch of waste, thus reducing the amount of secondary waste.

Loading factor

One of the crucial parameters in volume recalculation during conditioning is the loading factor. There are two forms of loading factor that need to be considered. The first is the percentage of waste that is disposed of in each container (with the rest being cement, water and various additives). The second is how full the container is filled. Normally, the container is not filled to 100% as some space is left to accommodate uncertainties during loading and for minor expansions during settling and curing of the cement matrix; this “lost” volume is usually no more than 5%. Also, we must consider each of the physical objects that are inserted into the waste container to handle the waste, like metal grills to hold in place the compacted waste, void space around solid objects being encapsulated, or the stirrer used in lost-stirring techniques; each of these objects takes up space and reduce the loading factor.

The loading factors also vary greatly between different types of cement and for different waste streams. OPC, which is the most used cement in waste conditioning, can allow loading factors of around 50% for most of the more stable and inert waste streams, like metallic fragments (compacted or not), and residual ashes. The problem can arise for OPC with ion exchange resins and concentrated salines, both of which may have higher than normal concentration of borate or other salts due to the interaction with the primary circuit where boric acid is present. In this case, the load factor can be as low as 20% in order to reduce the effects of the boric acid, which is a cement retardant, and the hydration heat during solidification, which can cause cracking phenomena, hindering the radionuclide retention capabilities of the cement matrix. Ion exchange resins will also contract and expand during dehydration and hydration cycles that occur during cement curing and subsequent re-wetting. This will also cause the cement matrix to crack, which will allow more water to enter and thus further propagate the cracking. For minimisation of these problems and allowing for good load factors, the use of different types of cement, like calcium sulphur-aluminate cement (SAC) has been investigated and demonstrated. This type of cement has proven to provide increased waste loadings, minimise leach rates of nuclides and improve the mechanical properties of the matrix. Hydration heat and thermal cracking may be avoided with the addition of zeolite in the cement mix, without compromising compressive strength and leaching rates. With these arrangements load factors of up to 70-75% can be reached with SAC matrices.

Cement embedding

The other possible conditioning method is cement embedding, where the waste is physically encapsulated in the cement matrix, immobilising the radioactive content. The most common problems in this case are not the chemical stability of the waste-cement mix, but the proper encapsulation of the various solid particles suspended in the waste matrix. In this case, the

main parameter is the linear rate of impregnation, which determines how well and how quickly the grout can cover and embed all the crevices and niches present in the waste, which is a very heterogeneous material, with a different shape, sizes and void spaces scattered throughout it. In addition, not all the space inside the container is considered available for certain waste streams (e.g. cement embedding of larger objects or compacted metallic “pucks”), as a several centimetre-thick annulus or frame is left for cement only, to strengthen the compressive and tensile properties and to reduce any possible radionuclide release.

One way to optimise the impregnation is to increase the water to cement ratio, but this has also very evident consequences, like lower than average compressive strength and reduced radionuclide retention. The need for highly impregnated cement is tied to the void presence in the waste; if it has gone through pre-treatment and treatment procedures like sorting, shredding and compaction, all of which should have reduced the void content, then the produced waste form will be much easier to condition with cement embedding techniques. If the void spaces are all open and linked, then cement can flow into the spaces more readily. If the voids are isolated inside an object, then no amount of fluidity will reach them unless holes are purposely created in the shell of the object.

On the other hand, if it was considered not necessary, or not economically viable, to implement a high degree of treatment before conditioning, then it is necessary to find a different solution. The advantage of this kind of waste management strategy is the possibility to have very little increase in volume during conditioning: this is possible since solid waste that has not been treated, has a void content in the range of 40-50% and with highly impregnated cement this volume can be used to condition the waste, hypothetically not increasing the volume of the final product in comparison with the initial poured volume of waste products. In order to do this, linear impregnation rates higher than 4-8 cm/min are needed. Such a rate can be achieved by the use of certain additives -- in particular with a class of polymers belonging to the poly-hexa-methylene-guanidine class (PHMG). With the use of PHMG polymers flowability of the cement grout is increased (around 1.5 times), loading factors are almost doubled, durability and frost resistance is increased, and leaching rates and mixing times are greatly reduced. However, the use of such polymers and super-plasticisers is often limited in higher radioactivity wastes due to their low radiation stability and the formation of chelating agents when they break down under the effects of radiation.

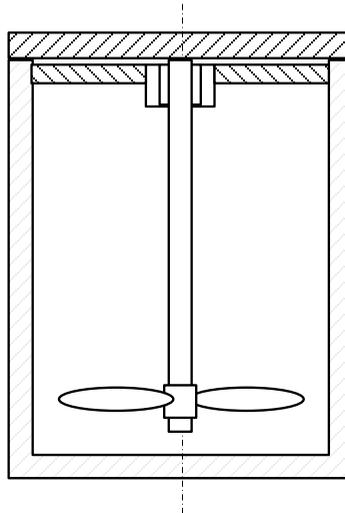
These kinds of considerations have been verified not just for ash residues, but also for different waste streams. Other waste streams that have been tested are mixed solid radioactive waste (SRW) (90% vol – scrap metals, 10% vol – soils), and coarse-grained SRW (construction material fragments). Different waste containers have also been considered: 200 L steel drums, 4 m³ cubic concrete containers, and direct disposal in near-surface repository chambers of 160 m³ (8 800 x 4 200 x 4 400 mm).

Packaging

The last phase of waste management prior to disposal is packaging. This is already partially decided in the conditioning phase, as the waste form with the immobilisation agent, cement, in this case, is conditioned inside the waste container. Hence, it is important to have a clear understanding of the requirements and advantages of each different container family and packaging option. Additionally, the waste package can be inserted into other containers and can be furtherly conditioned with buffer/filler materials, and in this case, the different options in choosing the optimal solution for the disposal module are not so tightly fixed on the waste package, even if there is a strong relation between the two. The packaging phase introduces, in any case, an increase in volume, and even if the volume of the waste inside the waste form does not change the effective volume now occupied by the waste package and/or the disposal module has to be taken into consideration. It is not beneficial to consider volume evolution only in terms of volume reducing practices, and not to consider those phases where there is a certain increase in volume, such as conditioning and packaging, since this hinders the ability to compare waste volume data and technical expertise between countries. There are multiple volumes that have to be taken into consideration when referring to waste packages and disposal modules:

- **Usable storage (or free) volume:** this is the volume available for the waste form, taking into consideration each of the internal accessories (holding grills or mixing rods) present inside the waste container taking up physical space. It also considers that part of the volume that is deliberately left empty, for handling reasons, or if the waste form expands over time.

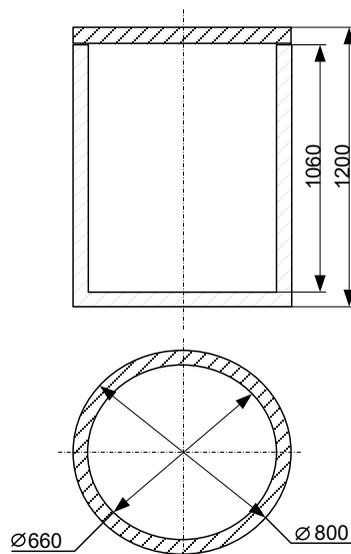
Figure D.2. Cross-section of a waste package with internal mixing gear



Source: OECD/NEA.

- **Internal volume:** this volume comprises the free volume, plus the volume of any internal accessories and void volumes, and describes the total volume inside the waste container, once the lid is sealed. It is always higher than the free volume.
- **External volume:** this is the volume of the waste package or disposal module calculated as the maximum external dimension of the object, this also takes into consideration the space taken up by external features, such as the container wall thickness, handles and vents.

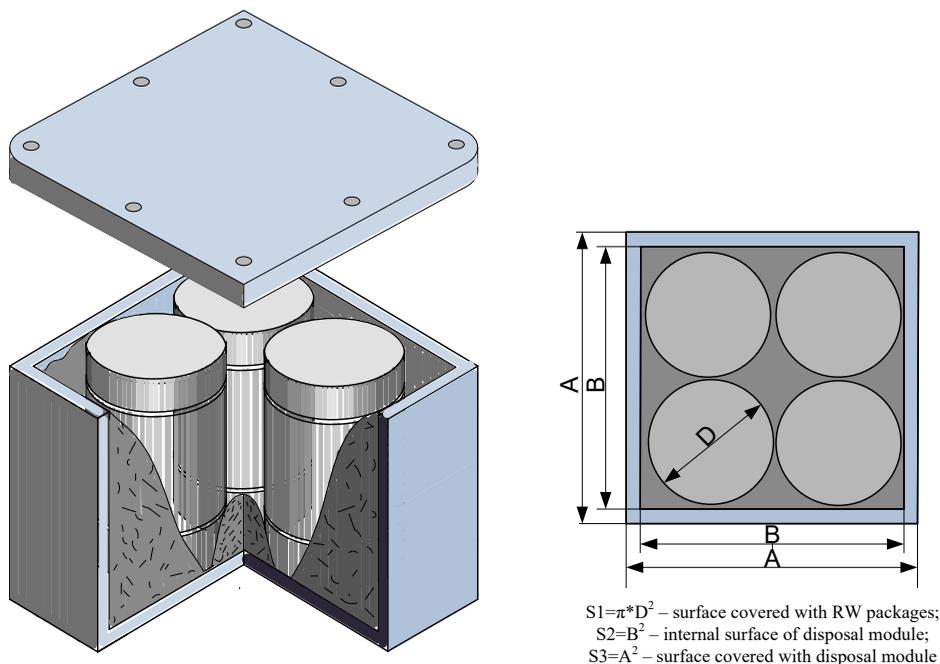
Figure D.3. Cross-section of a waste container, outlining internal and external dimensions



Source: OECD/NEA.

- **Envelope volume:** this volume is defined as the parallelepiped that envelopes the waste package or the disposal module, and it is different from the external volume if the object has a cylindrical shape, so it generally of interest for waste packages, as disposal modules tend to be rectangular. This volume is very important for the final storage or disposal volume required for multiple units, as it defines the volume lost when stacking objects with a cylindrical shape. Note that the volume increase from stacking cylindrical objects can be minimised by offsetting adjacent rows or tiers so that upper layers are nested in the curvature of the layer beneath it.

Figure D.4. Expanded view of a disposal module, showing the difference between external volume and envelope volume



Source: OECD/NEA.

It is important to state that the volume indicated with a certain waste container (like the 200 L drum, or the 400L container), does not necessarily always refer to one of these volumes. In some occasions, it is just a traditional or descriptive name which does not correspond to any of the above-mentioned volumes, and it is given just as a code of reference. In addition, there is no international standard for sizing of radioactive waste containers. Thus, the volume and dimensions of a “standard 200 L drum” in one country is likely to be different than those of another country.

The volume increase factor related to packaging can be calculated as the ratio of the external volume (or envelope volume depending on the shape of the container), and the usable volume and this will always be a positive number, greater than 1. There is a multitude of containers that have been used for waste packaging, and it is impossible to make an estimate of the volume increase for a general package. On the other hand, it can be stated what parameters are relevant in the packaging phase and are related to the final volume of the waste package or disposal module. The EGIRM specified the major four parameters involved in these considerations:

- shape;
- shielding (including wall thicknesses) and handling (e.g. embedded forklift pockets or lifting lugs);

- weight limits;
- stability and durability.

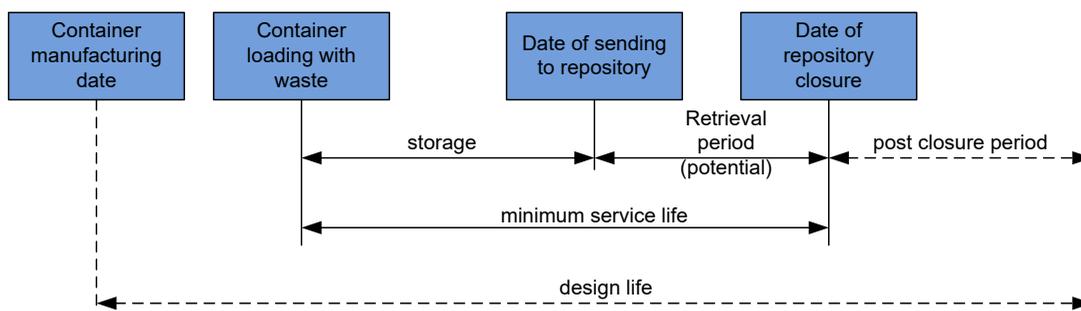
The shape of the container is obviously a key parameter in determining the volume increase, with thick-walled cylindrical containers having the greatest increase, and thin-walled cubic or rectangular containers having the lowest volume increase, for the same usable storage volume available. The shape also determines whether there is a need for additional external features, such as handles or forklift pockets, required for the movement and the handling of the waste. In most cases, there is a difference between the maximum external dimension of a package (height and diameter, or height, width and depth) and the average dimension, and this is to give a space for handling cranes and machines to grip the unit with more ease.

Another parameter is the radiological shielding of the container, needed to achieve a safety status and acceptable external doses, related to transport, handling and storage scenarios. These shielding requirements can be achieved either with thicker container walls, usually made of steel or concrete, materials with good shielding properties, or by restricting the usable storage volume available inside the internal volume, creating an internal spacing where the waste cannot be placed, and which is usually occupied by the immobilisation agent (e.g. cement grout). The higher the activity of the waste, and the higher the linear penetration of the radiation, the bigger the volume increase due to shielding requirements.

The total weight of the waste package or disposal module can be a limiting factor in some cases. This can be more of an issue when treating very dense materials inside the waste form or with a large-sized waste package or disposal module, since if the maximum weight of the unit is surpassed then the loading factor has to be reduced, and the same waste form volume has to be distributed between more waste packages or disposal modules.

Stability and durability of the container are also relevant in this calculation. The longer the waste package is expected to maintain its structural properties, the thicker and more engineered the container walls are going to be, resulting in a more significant volume increase. A typical container life cycle is shown in Figure D.5. Non-heat-emitting waste packages are usually assumed to fail between 50 and 500 years after production, and in this consideration, it has to be included the time spent in storage and not only disposal. For heat-generating wastes (HLW or spent fuel), the container design life may be many thousands of years.

Figure D5. Typical container lifecycle



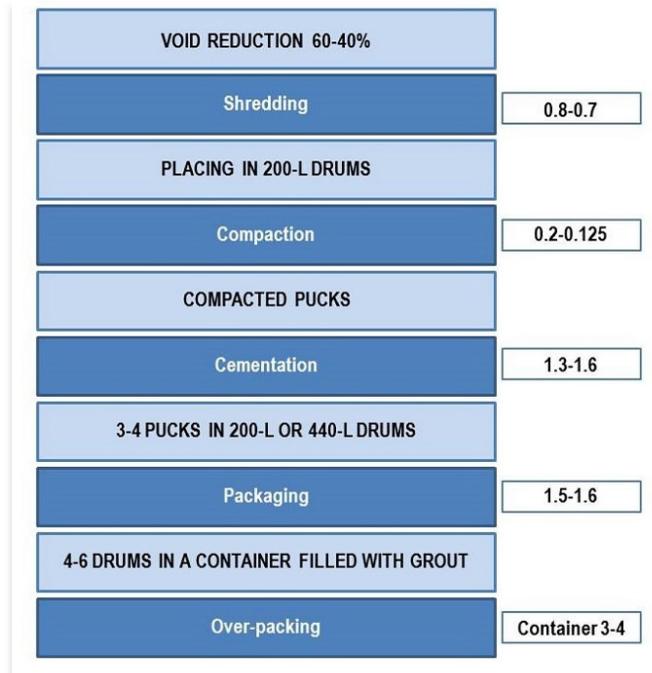
Source: OECD/NEA.

Volume evolution

After consideration of all common and implemented scenarios in a waste management programme centred on cementation of radioactive waste and having followed and analysed the parameters involved during each phase of the RWM, it is possible to give an overall picture of the volume evolution of some common waste streams that have been introduced previously. It is necessary to note that all of the factors cited in volume recalculation have to be taken as indicative and very general since they are too dependent on the exact starting waste

composition and the particular process involved to be completely accurate for a general approach. In this report, a range is given for these coefficients as an indication of what values one should expect when going through a similar process. Values less than 1 indicate a reduction in volume, while values greater than 1 indicate a volume increase.

Figure D.6. Cementation of solid radioactive waste



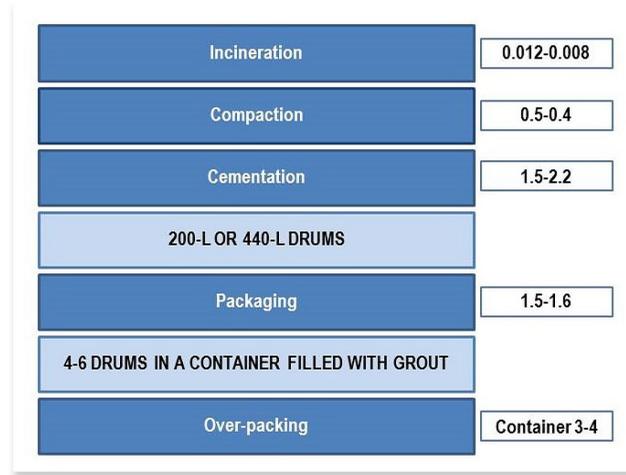
Source: OECD/NEA.

As a first example, a metallic waste stream has been considered. The stream can be first shredded and compacted into 220 L drums; then these will be embedded with cement grout and packaged into 440 L drums.

It is necessary to note: the minimal and maximal values are obtained with not just multiplying of all lower and all upper values. These are just approximated values. Recalculating, one should take into account that not only reducing or increasing of radioactive waste volume is possible from the stage to stage. Very often the positive and negative volume changes can change each other stage by stage.

Ashes after incineration

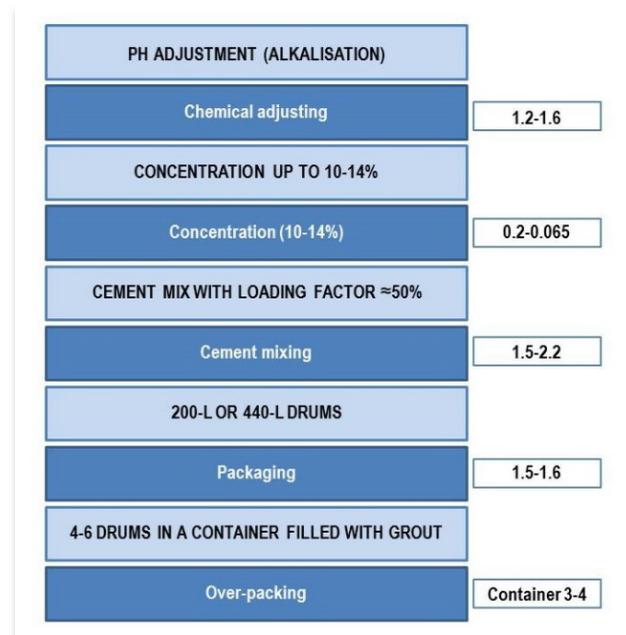
As a second example, a solid waste stream is considered, going first through an incineration process. The resulting ashes are being compressed and cemented into an OPC mix. The resulting waste form is packaged inside a 440L drum, and as previously the volume evolution is also given for the emplacement inside a disposal module.

Figure D.7. Cementation of burnable radioactive waste

Source: OECD/NEA.

Liquid waste stream

This example is a widely used process – the cementation of a liquid waste stream, which should be firstly concentrated and subsequently mixed, poured and embedded with cement grout, and finally packaged. The waste packages are usually placed into a concrete container, just like in the preceding example, considered as a disposal module, ready for storage or disposal. As previously the various stages for that the volume evolution is calculated are stated: from the initial volume to the waste package volume, or from the initial volume to the disposal module volume.

Figure D.8. Cementation of liquid radioactive waste

Source: OECD/NEA.

Vitrification of radioactive waste

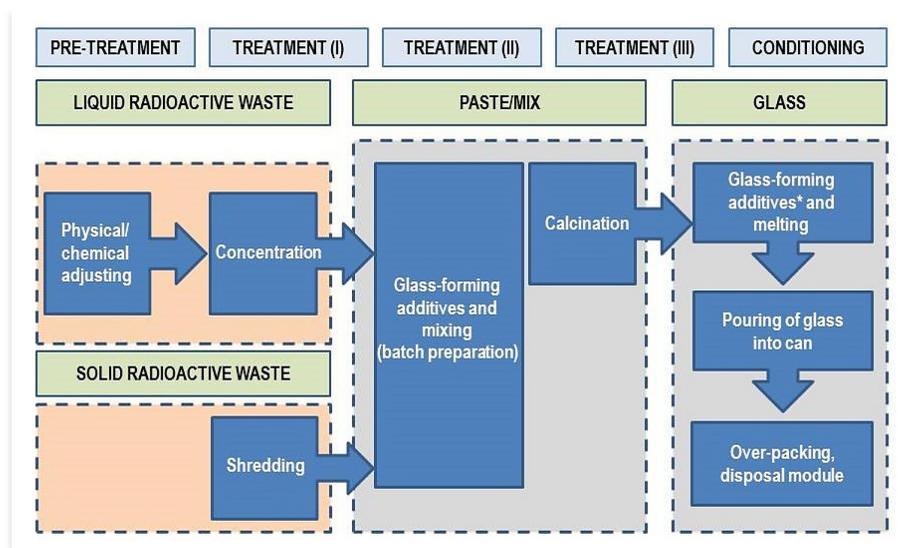
Waste vitrification technology has been successfully implemented in several countries for more than 40 years. By the year 2000 vitrification facilities had produced about 10 000 t of glass with immobilised radionuclides, and at the end of 2013, there was about 22 000 m³ of vitrified HLW stored worldwide. First, this technology is the most suitable for the treatment of liquid HLW obtained from spent fuel reprocessing. Borosilicate glass is the accepted type of matrix in most applications although other formulations are also used, e.g. phosphate glasses are used to immobilise some HLW in Russia. The high durability of glass ensures a high degree of environmental protection. Radioactive waste vitrification gives high waste volume reduction. Although vitrification requires a high initial investment and operating costs, the overall cost of vitrified radioactive waste is usually lower than alternative options when the subsequent transport and disposal costs is accounted for. However, the application of the method for ILW and even LLW (both solid and liquid) has been considered in some countries as well (Russia, Korea, the United States, etc.). Sometimes, even vitrification of LLW can be an alternative to cementation considering further management in the long-term perspective. Vitrification is the waste treatment method providing the following considerable benefits:

- high chemical durability of glass to leaching effect of water;
- thermal and radiation durability of glass;
- high mechanical strength;
- ability to immobilise a wide range of radionuclides independently of their electronegativity;
- high capacity to include radionuclides;
- well-developed technological principals; etc.

In this section, the technology of waste vitrification and all the factors that are relevant for volume reduction of waste are examined. A visual overview of the technological steps of the vitrification method and how they relate to each other is provided. Vitrification technology comprises several stages, starting with the evaporation of excess water from liquid radioactive waste, followed by paste preparation, calcination, glass melting, and ending with pouring and cooling of vitrified waste blocks with some small amounts of secondary waste.

Two types of nuclear waste glass preparation processes are currently used: one-stage vitrification and two-stage vitrification. In the one-stage vitrification process, glass-forming additives are mixed with concentrated liquid wastes, and so a glass-forming batch is formed (often in the form of a paste). This paste is then fed into the melter where further water evaporation occurs, followed by calcination and glass melting, which both occur directly in the melter. In a two-stage vitrification process, the waste is first calcined, and after calcination, the required glass-forming additives (usually as a glass frit) together with the calcine are fed into the melter.

For better clarity, the process has been conditionally divided into several stages, pre-treatment, treatment (divided into three sub-stages), and conditioning. This division is used to give a logical order to the process; it is not intended to relegate a particular practice to one or the other stage and should be considered accordingly. Sometimes the calcination is technically separated from the melting, in other cases, the calcination and melting can be realised in one apparatus (one stage). Anyway, the melting process is usually preceded by calcination. Also, the sludge preparation can be realised in a specific stage, or both the waste and glass formers can be loaded into the melter.

Figure D.9. Stages of the vitrification process

Source: OECD/NEA.

Chemical adjustment

Prior adjusting of waste chemical properties is the most common pre-treatment step of waste vitrification. The need to adjust the chemical composition is justified because the subsequent steps will concentrate salts and may impact on sludge preparation and further glass melting. The main adjusted parameter is the pH, and the required value depends on the type of glass obtained in the process (boro-silicate, phosphate or other glass). pH regulation may be required to prevent the precipitation of mineral salts (CaCO_3 and MgCO_3) during vaporisation. The presence of surfactants in the initial liquid radioactive waste can require the application of different de-foamers for the evaporation stage when it is used.

Regulation of the pH of liquid waste usually does not cause significant volume variations, but it provides the feasibility of further sludge preparation and loading into the melter for subsequent optimal operation in the limited range of pH values.

It is necessary to note that not all types of radioactive waste are initially suitable for vitrification. Sometimes, special fluxes added at different stages of the vitrification process or process parameters or procedures changes could solve problems. For example:

- radioactive waste may contain large proportions of refractory elements that sometimes limit the waste loading;
- radioactive waste may require higher processing temperatures which result in volatilisation and loss of semi-volatile contaminants;
- radioactive waste chemical composition may cause problems in controlling the viscosity or conductivity of the melt;
- radioactive waste chemical composition may cause the glass crystallisation (e.g. spinels) or the forming of separate phase (e.g. “yellow phase” in phosphate glass) which can hinder processing;
- radioactive waste may have a highly variable composition that can affect processing (averaging by blending is needed).

Sometimes, additives can influence the final product volume, but the contribution to total volume evolution is usually not significant. On the other hand, it is necessary to note that the economic effect of using additives can be rather negative for the treatment cost. Typical situations and types of additives for various types of glass are described in a number of

publications (see references). It is important to note, there are developed technological processes for several types of radioactive waste, and glass-forming additions and additives are recommended for the provision of effective treatment. These developed recipes can be considered for the selection of a suitable glass matrix and a recipe for radioactive waste to be treated. The compositions of glasses are developed for technically and economically viable preparation and melting, for excluding phase separation or uncontrolled crystallisation of glass, and for acceptable chemical durability, e.g. leaching resistance in contact with ground waters at disposal sites. High waste loadings and high chemical durability can be achieved, for example, in borosilicate and aluminium-phosphate glasses.

Chemical adjustment can be applied not only at the pre-treatment stage. The chemical composition of radioactive waste can be corrected at the stage of sludge preparation or the melting stage depending on physicochemical properties of the processed material at the given stage.

Two main glass families are currently used for nuclear waste immobilisation: borosilicate and phosphate. Moreover, such glasses immobilise large quantities of actinides well. For example, borosilicate glasses can accommodate up to 7.2 mass% of PuO₂. Phosphate glasses can accommodate large amounts of aluminium oxides. However, in contrast to borosilicate melts, molten phosphate glasses are highly corrosive to refractory linings. Some liquid waste streams contain sulphate and chloride ions, which limits the waste oxide content to 5-10% wt because of the low sulphate and chloride solubility (~1%) in silicate and borosilicate melts. Thus, LILW vitrification becomes inefficient. Excess sulphate-chloride phases segregate as a separate phase floating on the melt surface because of the immiscibility of silicate and sulphate (chloride) melts. The same phenomenon occurs for molybdate- and chromate containing waste vitrification, where the separate phase is coloured and named "yellow phase". The content of these components in radioactive waste can stipulate the glass type selection.

Concentration

There are two main options for implementing the vitrification process: concentration before vitrification and concentration directly in the melting zone. In both cases, all water will eventually be removed from the radioactive waste during the process by evaporation. From the point of view of the liquid radioactive waste volume evolution, the concentration stage is one of the biggest contributors to the total volume change. When the concentration is applied separately and followed with sludge preparation, the depth of concentration is usually limited by the level of dynamic viscosity of the obtained sludge. It should be low enough to allow transport to the melter through pipes from the sludge mixer. It is necessary to know the dynamic viscosity of the sludge at the temperature of mix preparation to define the concentration factor. It can be determined by the experimental study. In the case when the evaporation of water is performed just in the melter, this parameter is not of importance. Also, sometimes, concentration can be performed to the point of obtaining dry salts which consequently are directed to the melter (previous calcination). For all these options, the radioactive waste volume decrease can be found based on a value calculated from saline density at a given salt concentration. Depending on the technological scheme of the treatment, the radioactive waste volume can be reduced by a factor 2-3 overall for concentrated liquids and up to 100 for more dilute liquids.

Preparation of the batch with glass-forming additions

A number of glass compositions were developed for immobilisation of liquid waste containing sodium nitrate. Various boron-containing minerals, as well as sandstone, were tested as glass-forming additives. Datolite CaBSiO₄(OH) was found to be a suitable fluxing agent. Loam and bentonite clays were also used as glass-forming additives. Up to 50% of either loam clay or bentonite in the batch was substituted for sandstone. This substitution increases the chemical durability of glass and, moreover, such batches containing 20 to 25% wt. of water form homogeneous pastes, which are stable for long times without segregation and are transportable in pipes over long distances. Sodium nitrate is the major component of both institutional liquid

LLW and ILW and NPP operational wastes from RBMK (channel type uranium-graphite) reactors. NPP wastes from water-water energetic reactors (WWER) contain boron, although the major components of this waste are sodium nitrate and sodium tetrahydroxyborate NaB(OH)_4 . Thus, there is no need to add boron-containing additives to vitrify WWER waste. Silica, loam, or bentonite clay or their mixtures are suitable as glass-forming additives. WWER waste glasses are in the $\text{Na}_2\text{O}-(\text{Al}_2\text{O}_3)-\text{B}_2\text{O}_3-\text{SiO}_2$ system for which glass-forming regions are well known. Long-term tests of vitrified LILW have been carried out in a shallow ground experimental repository since 1987. These show a low and diminishing leaching rate of radionuclides. Boron-free aluminosilicate glasses in the $\text{Na}_2\text{OCaO}-\text{Al}_2\text{O}_3-\text{SiO}_2$ system for immobilisation of institutional and RBMK wastes were produced from waste, sandstone, and loam clay (or bentonite).

Calcination

Calcination is also an integral part of the vitrification process where the thermal destruction of chemical combinations happens. Calcination is one of the more significant contributors to the volume evolution. In this phase, nitrates are decomposed and emitted as gaseous N_xO_y . All organic compounds are also broken down and emitted as CO_x , C_xH_y and similar gases. The calcination stage can be organised separately in special apparatus, or it can be conducted directly in the working space of the melter. From the point of view of the volume evolution, it is necessary to note that after calcination the radioactive waste chemical composition is mostly present as oxides. Depending on the content of nitrates and other volatile compounds, the volume of radioactive waste can decrease by up to 20%.

Melting

Two types of melters are most frequently used at waste vitrification plants: Joule-heated ceramic melters (JHCM) and induction-heated melters which can either be hot (IHC) or cold crucible melters (CCM). Melting of nuclear waste glasses needs to be performed efficiently at low temperatures (less than about 1200°C) because of the volatility of some fission products, notably Cs and Ru, at higher temperatures, thus avoiding excess radionuclide volatilisation and maintaining viscosities below $10 \text{ Pa}\cdot\text{s}$ to ensure high throughput and controlled pouring into canisters. A more fluid glass is preferred to minimise blending problems. However, higher fluidity is associated with higher temperatures and higher losses of volatile radionuclides (Cs, Ru, Tc).

Phase separation during melting is most important for waste streams containing glass-immiscible constituents. However, these can be immobilised in the form of isolated and phase separated dispersions (in glass composites).

During melting, some fluxes or additives can be added to adjust physical properties of the melt. For CCM technology, for example, the concentration of sodium is of importance as it defines electroconductivity of the melt. The required value of electroconductivity is usually defined for a given electromagnetic field generator and melter geometry. For the Joule-heated process, the melt electroconductivity is also the defining parameter. However, other values are required. When heat is delivered through the melter wall (the wall is heated), the electroconductivity is not a crucial parameter. However, the content of sodium is also important to provide a required viscosity of the melt. For each given facility, the required parameters and their values can differ, and the amount and content of fluxes vary as well. As stated earlier, fluxes do not change the final volume considerably, and this contribution can easily be accounted for during the recalculation.

Glass discharging, packaging and disposal module forming

The melted waste glass is poured into containers (canisters) typically made of stainless steel when immobilising HLW or carbon steel for vitrified LLW or ILW to allow for handling and storage of the waste glass. These may or may not be slowly cooled in an annealing furnace to avoid accumulation of mechanical stresses in the glass.

After glass discharging, some change can occur in the glass due to its solidification. This change is usually not significant and depends on the properties of the given glass. Usually, glass is discharged into canisters (about 200 L). After that, depending on the process, canisters with glass cool down and solidify. Subsequently, canisters can be put into an outer capsule awaiting disposal. Disposal of vitrified radioactive waste can be done in a disposal container which increases the final volume. It also can be done in the capsules or original canisters, and a thick-walled container is often used for transport. The choice of final disposal packaging configuration will affect the final disposal volume to varying degrees.

Volume evolution

After consideration of all the main scenarios in a waste management programme based on vitrification of radioactive waste and analysis of the parameters involved during each phase of the vitrification process, an overall picture of the volume evolution of some waste streams that were discussed previously emerges. It is necessary to emphasise, that all the factors cited in volume recalculation have to be taken as indicative and very general since they are too dependent on the exact starting waste composition and the particular process involved to be completely accurate for a general approach. A range of the volume change coefficients is given below, as an indication of what values to be expected when going through a similar process.

As an example, a solid waste stream, to be first shredded and compacted into pellets; then mixed with glass frit and put into the melter (e.g. Korean approach).

Figure D.10 Example of solid radioactive waste vitrification

Shredding	0.6-0.4
Pellet forming	0.2-0.125
Vitrification	0.3-0.1
Discharging	200-L can 1.0-1.2
Over-packing	Container 1.0-1.2

Source: OECD/NEA.

Figure D.11 example for liquid LLW vitrification

Chemical adjusting	1.0-1.2
Concentration	0.5-0.01
Sludge preparation	1.4-2.0
Melting	0.3-0.7
Discharging	Container 20-L 1.1-1.3
Over-packing	Container 3m ² 1.2-1.6

Source: OECD/NEA.

Annex E. Review of requirements and recommendations regarding the reporting of national inventories to international bodies, initiatives and programmes

Joint Convention on the Safety of Spent Fuel Management and on the safety of Radioactive Waste Management (INFCIRC/546 24 December 1997)

“Each Contracting Party shall in due course take the appropriate steps to review:

(ii) the results of past practices in order to determine whether any intervention is needed for reasons of radiation protection bearing in mind that the reduction in detriment resulting from the reduction in dose should be sufficient to justify the harm and the costs, including the social costs, of the intervention.”

“For each Contracting Party, the report shall also address its:

- (i) spent fuel management policy;
- (ii) spent fuel management practices;
- (iii) radioactive waste management policy;
- (iv) radioactive waste management practices;
- (v) criteria used to define and categorise radioactive waste.

“This report shall also include:

- (ii) an inventory of spent fuel that is subject to this Convention and that is being held in storage and of that which has been disposed of. This inventory shall contain a description of the material and, if available, give information on its mass and its total activity;
- (iv) an inventory of radioactive waste that is subject to this Convention that:
 - (a) is being held in storage at radioactive waste management and nuclear fuel cycle facilities;
 - (b) has been disposed of; or
 - (c) has resulted from past practices.

This inventory shall contain a description of the material and other appropriate information available, such as volume or mass, activity and specific radionuclides”.

Guidelines regarding the Form and Structure of National Reports – INFCIRC/604/Rev.2 (07 Sept 2012)

“...

each Contracting Party may submit a report with the form, length and structure it believes necessary in order to describe the measures taken to implement its obligations under the Convention;”

“the need for effective and efficient review makes it desirable that reports be in as similar a format as possible, to aid comparison;”

“Contracting Parties are encouraged to report in Système International (SI) units.”

“It should include a statement outlining the national policy for spent fuel management and a description of national practices pertaining to spent fuel management, together with a statement outlining the national policy for radioactive waste management and a description of national practices pertaining to radioactive waste management. It should also specify the criteria used to define and categorise radioactive waste.”

“In this section, the position of the Contracting Party as regards the following matters should be stated clearly:

- (a) whether the Contracting Party has declared reprocessing to be part of spent fuel management, pursuant to Article 3(1);
- (b) whether any waste that contains only naturally occurring radioactive material and does not originate from the nuclear fuel cycle has been declared as radioactive waste for the purposes of the Convention, Article 3(2), and, if so, where this radioactive waste appears in the inventory; and
- (c) whether any spent fuel or radioactive waste within military or defence programmes has been declared as spent fuel or radioactive waste for the purposes of the Convention, pursuant to Article 3(3).”

“Contracting Parties are encouraged to use clearly defined waste categories when reporting inventories”.

“Contracting Parties are encouraged to report here on their experiences concerning transboundary movements.”

“- national strategy for the management of disused sealed sources, ...”

Council Directive 2011/70/EURATOM of 19 July 2011 establishing a Community framework for responsible and safe management of spent fuel and radioactive waste

“Where radioactive waste or spent fuel is shipped for processing or reprocessing to a Member state or a third country, the ultimate responsibility for the safe and responsible disposal of those materials, including any waste as a by-product, shall remain with the Member State or third country from which the radioactive material was shipped”.

“Member States shall submit a report to the Commission on the implementation of this Directive for the first time by 23 August 2015, and every 3 years thereafter, taking advantage of the review and reporting under the Joint Convention”.

“On the basis of the Member States’ reports, the Commission shall submit to the European Parliament and the Council the following:

- (a) a report on progress made with the implementation of this Directive; and
- (b) an inventory of radioactive waste and spent fuel present in the Community’s territory and the future prospects EN 2.8.2011 Official Journal of the European Union L 199/55”.

Also, in article 12, concerning national spent fuel/radioactive waste programme content the next is required:

“an inventory of all spent fuel and radioactive waste and estimates for future quantities, including those from decommissioning, clearly indicating the location and amount of the radioactive waste and spent fuel in accordance with appropriate classification of the radioactive waste”.

HLG_p(2014-27)_137. C. “Final Guidelines for MS Reports to the Waste Directive”

“It is proposed that the International Atomic Energy Agency (IAEA) Classification of 2009 (IAEA Safety Guide “Classification of Radioactive Waste” GSG-1, 2009) is used as the basis for the unified system. Guidance on how to transform national classification schemes into the unified system is provided in Appendix 2”.

“Note: It is suggested that for the purposes of clarity that Member States provide the Commission with information on how they have translated their national classification systems into the unified system. It should be noted that for the purposes of notifying the national programme, Member States should use their national classification systems”.

“On the basis of these reports, the Commission will report to the European Parliament and the Council on the spent fuel and radioactive waste inventory in the EU and future prospects”.

“Therefore, it will be necessary to focus the inventory reporting on the main waste categories in direct relationship with long-term management by or towards disposal, and to clearly indicate in the inventory reporting the relationship between the main waste categories and the disposal routes.

Inventory reporting by MS should also be based on common rules such as units and reference dates, in order to obtain homogeneous and easily comparable and interpretable information”.

“In the reporting on the radioactive waste and spent fuel inventory MS should clearly indicate for each of the main waste categories the considered, planned or operational disposal routes”.

“Member States have also national policy and management flexibility in terms of separate disposal facilities for each waste category or combined disposal facilities for more than one waste category. A MS might plan to develop one single disposal facility for all its radioactive waste. If radioactive waste from two or more waste categories is (or is planned to be) co-disposed in one disposal system, this should be indicated in the reporting either by providing a waste volume for the two or more combined categories (e.g. low-level waste [LLW] + intermediate-level waste [ILW] or ILW + high-level waste [HLW]) linked to one disposal system or by providing waste volume estimations for each of the waste categories that are or will be routed to the same disposal system”.

“As ‘exempt waste’ has not the regulatory status of radioactive waste, it is excluded from the Waste Directive. In addition, for very short-lived waste – VSLW – no inventory reporting is required, given the temporary radioactive character of the waste”.

“Waste inventories are reported as rounded value waste volumes (m³) for each of the waste categories very low-level waste (VLLW), LLW, ILW and HLW.”

“The following information should be provided:

- Waste volumes (m³) by category disposed of in (an) operational or closed disposal facility(y)(ies). For operational disposal facilities the existing total capacity (m³) should also be given.
- Member States which have exported waste for disposal should indicate the quantities concerned (volumes in m³ by category) and countries of destination and refer to the export agreement(s).
- Waste volumes (m³) of conditioned waste by category stored in storage facilities. For unconditioned waste in storage, if possible, some indication of the final conditioned volume should be provided.
- This should include waste currently stored abroad, subject to return, e.g. HLW from reprocessing.
- MS holding foreign waste for return should indicate quantities and destinations (EU or non-EU countries)”.
- “The following information should be provided:
- Quantities of spent fuel disposed of (tHM), number of assemblies and type – boiling water reactor (BWR), pressurised water reactor [PWR], CANDU, MOX, spent fuel from research reactor.
- If shipped to another MS or outside EU for disposal (or reprocessing without return of waste): quantities in tHM, number of assemblies and type, countr(y)(ies) of destination, and reference to the export agreement(s).

- Quantities of spent fuel in storage (tHM), number of assemblies and type (BWR, PWR, CANDU, MOX, spent fuel from research reactor, ... by store type (dry cask, vault, pond) and locations (number of locations could be acceptable if this is seen as sensitive).

For spent fuel from research reactors, the quantities of fuel subject to a 'return' agreement should be indicated.

MS should indicate separately the quantities and location of spent fuel stored abroad awaiting reprocessing. Correspondingly MS holding foreign fuel should indicate the quantities stored from EU and non-EU countries".

Guidelines for Member States reporting on Article 14.1 of Council Directive 2011/70/Euratom (January 2018)

"Member States can deliver the inventory and the future prospects according their national classification system. In order to make the information comparable on the Community's territory, Member States are recommended to provide a table to translate their national inventory into the IAEA classification system (IAEA Safety Guide "Classification of Radioactive Waste" GSG-1, 2009) or directly report their inventory according to IAEA classification system".

"Inventory reporting by Member States should clearly indicate the units on which the inventory is based and the reference dates. Member States can use the latest inventory reported under Joint Convention or update".

"The inventory should comprise all radioactive waste (nuclear facilities operation and ongoing decommissioning, medical, research and industrial applications, remediation, ...) present in the Member States territory".

"MS are expected to report the following information:

- Waste volumes (m³) by category disposed of in each operational or closed disposal facilit(y)(ies). Type of disposal is expected to be reported (Borehole, Landfill, Near Surface Disposal, Geological, Deep Geological). For operational disposal facilities the existing total capacity (m³).

Member States which have exported waste for disposal are expected to indicate the quantities concerned (volumes in m³ by category) and countries of destination, and refer to the export agreement(s).

- Waste volumes (m³) of conditioned waste by category stored in storage facilities. For unconditioned waste in storage, if possible, indication of the final conditioned volume is expected to be provided.
- For disused sealed sources declared as radioactive waste it has to be stated if they are included in the different waste categories, if not they should be reported separately.
- Member States holding foreign waste for return are expected to indicate quantities and destinations (EU or non-EU countries)".

"MS are expected to report the following information:

- Spent fuel disposed of (tHM, number of assemblies and type – BWR, PWR, CANDU, MOX, research reactor, ...);

If shipped to another Member State or outside EU for disposal (or reprocessing without return of waste): quantities in tHM, number of assemblies and type, countr(y)(ies) of destination.

- Spent fuel in storage (tHM, number of assemblies and type [BWR, PWR, CANDU, MOX, spent fuel from research reactor, ...] by store type (dry cask, vault, pool) and locations (aggregation of locations could be acceptable if this is seen as sensitive).

For spent fuel from research reactors the quantities of fuel subject to a "return" agreement is expected to be indicated.

Member States are to report separately the quantities and location of spent fuel sent abroad for reprocessing. Correspondingly Member States holding foreign fuel are expected to indicate the quantities stored from EU and non-EU countries”.

“Member States are expected to report the total volumes (m³) (as disposal volumes¹, if possible) of radioactive waste by category for storage and/or disposal for relevant time horizons (e.g. 2030, 2050, ...) for:

- operational waste including new build (lifetime assumptions to be given);
- decommissioning waste (decommissioning assumptions to be given);
- remediation;
- waste (including disused sealed sources categorised as radioactive waste) from medical, research and industrial application.

This information is expected to include waste currently stored abroad, subject to return, treatment of radioactive waste abroad or intermediate-level waste (ILW)/high-level waste (HLW) from reprocessing”.

“Member State are expected to report the following information:

Total expected quantities (tHM, number of assemblies) of spent fuel from reactors under operation, including from research reactors if return is not envisaged, and from new build (lifetime assumptions to be given) for relevant time horizons (e.g. 2030, 2050, etc.)”.

1. “Disposal volume” as the total volume taken up by the waste package for disposal.

Annex F. Overview of containers for radioactive waste disposal

Table. F.1. Containers for radioactive waste used in some countries

Capacity/type of container	Dimensions		Radioactive waste class	Remarks
	D * H (mm)	H * L * W (mm)		
Metallic barrels (m³)				
0.2	(570-620) * (775-880)	-	LLW (VLLW)	Many countries
0.225	(590-620) * (880-850)	-	LLW (VLLW)	Many countries
0.38	710 * 1 100	Carbon steel	LLW (VLLW)	Italy
0.4	770 * 1 100	-	LLW (VLLW)	Many countries
0.44	791 * 1 100	Stainless steel	LLW-ILW	Italy
Metallic container (steel) (m³)				
2.5	-	850 * 1 700 * 1 700	LLW (VLLW)	France
2.6	-	1 250 * 1 650 * 1 250	LLW	Italy
5.0	-	1 700 * 1 700 * 1 700	LLW (VLLW)	France
5.2	-	1 250 * 2 500 * 1 650	LLW-ILW	Italy
10.0	-	850 * 3 400 * 3 400	LLW (VLLW)	France
20.0	-	1 700 * 3 400 * 3 400	LLW (VLLW)	France
6.0	-	2 000 * 2 400 * 2 200	LLW (VLLW)	UK
12.0	-	2 500 * 4 000 * 2 400	LLW (VLLW)	UK
10	-	2 990 * 2 440 * 1 340	LLW (VLLW)	Sweden
20	-	6 060 * 2 500 * 1 340	LLW (VLLW)	Sweden
40	-	6 060 * 250 * 2 600	LLW (VLLW)	Sweden
3.8 (type I)	-	1 450 * 1 600 * 1 700	NHE RW (VLLW, LLW, ILW)	Germany
4.6 (type II)	-	1 700 * 1 600 * 1 700	NHE RW	Germany
8.7 (type III)	-	1 700 * 3 000 * 1 700	NHE RW	Germany
7.4 (type IV)	-	1 450 * 3 000 * 1 700	NHE RW	Germany
10.9 (type V)	-	1 700 * 3 200 * 2 000	NHE RW	Germany
5.4 (type VI)	-	1 700 * 1 600 * 2 000	NHE RW	Germany
3.1 (KMZ)	-	1 650 * 1 650 * 1 375	LLW (ILW)	Russia
1.36 (KRAD-1.36)	-	1 280 * 1 280 * 900	LLW (ILW)	Russia
3.0 (KRAD-3.0)	-	2 620 * 1 430 * 1 080	LLW (ILW)	Russia
0.2 (KO-1340)	560 * 910	-	LLW (ILW)	Russia
3.7	-	1 600 * 1 600 * 1 600	(ILW)	Japan
3 (3 m ³ resin liner)	1 630 * 1 800	-	ILW/LLW	Canada
2.3 (B25 box)	-	1 300 * 1 840 * 1 120	LLW	Canada
1.8 (AIBN)	-	1 400 * 1 320 * 1 320	LLW	Canada
2.5 (NPB47)	-	1 190 * 1 960 * 1 320	LLW	Canada
1.2 (DBIN 6 * 0.2 m ³ drums)	-	1 030 * 1 960 * 1 320	LLW	Canada
Cast iron containers (m³)				
0.7 (type I)	900 * 1 150	-	NHE RW	Germany
1.3 (type II)	1 060 * 1 500	-	NHE RW	Germany
1.0 (type III)	1 000 * 1 240	-	NHE RW	Germany
Steel Mould	-	1 200 * 1 200 * 1 200	LLW, ILW	Sweden

Notes: D – diameter; H – height; L – length; W – width; HLW – high-level waste; intermediate-level waste – ILW; Low-level waste – LLW; radioactive waste – RW; very low-level waste – VLLW; capacity/volume – internal (useful) volume of containers.

Table F.1. Containers for radioactive waste used in some countries (cont'd)

Capacity/type of container	Dimensions		RW class	Remarks
	D * H (mm)	H * L * W (mm)		
Armoured concrete containers (m³)				
2.0	1 400 * 1 300	-	ILW	France
4.2	-	1 260 * 2 110-1 560	ILW	France
3.0	-	1 200 * 1 720 * 1 720	ILW	UK
3.5	1 400 * 2 200		LLW-ILW	Italy
12.0	-	2 200 * 4 000 * 2 400	ILW	UK
1.2 (type I)	1 600 * 1 370	-	NHE RW	Germany
1.4 (type II)	1 060 * 1 500	-	NHE RW	Germany
3.1 (type III)	1 000 * 1 240	-	NHE RW	Germany
Concrete mould	-	1 200 * 1 200 * 1 200	LLW, ILW	Sweden
Concrete tank	-	3 300 * 1 300 * 2 300	ILW	Sweden
1.5 (NZK-150-1.5P)	-	1 650 * 1 650 * 1 375	LLW, ILW	Russia
1.9 (NZK Radon)	-	1 650 * 1 650 * 1 340	LLW, ILW	Russia
0.8 (RWC-PT/CT)		2 250 * 1 850 * 1 850	ILW	Canada
2.7 (RWC-EF)		1 920 * 3 350 * 1 700	ILW	Canada
Other materials (m³)				
1.5 (RTK – resin tank, high-density polyethylene)		1 680 * 1 240 * 1 240	LLW	Canada
Cans and capsules for glass (m³)				
0.54 (capsule)	620 * 3 000	-	HLW	Russia
0.15 (overpack)	500 * 1 700	-	HLW	Japan
"Super containers" (m³)				
(ONDRAF container)	... * 4 500	-	HLW	Belgium

Notes: D – diameter; H – height; L – length; W – width; HLW – high-level waste; intermediate-level waste – ILW; low-level waste – LLW; radioactive waste – RW; very low-level waste – VLLW; capacity/volume – internal (useful) volume of containers.

In countries where the strategy of spent fuel management is direct disposal, the containers are used for the spent fuel transportation, storage and sometimes disposal. Below, some examples are given to illustrate existing technical solutions (see Table F.2). It is necessary to note that the design of the spent fuel container depends on the type of fuel, type of reactor and the purpose of the container development. In the case of spent fuel references to existing examples hardly could be used.

Table F.2. Spent fuel containers (examples) (tHM, N of bundles/assemblies)

Capacity/type of container	Dimensions		RW class	Remarks
	D * H (mm)	H * L * W (mm)		
48 bundles (0.96 tHM CANDU)	600 * 2 500	-	SF	Canada
12 BWR bundles or 4 pressurised water reactor bundles but the capacity depends on burnout of bundles and heat generation	1 050 * 4 835	-	SF	Sweden, Finland

Notes: D – diameter; H – height; L – length; W – width; SF – spent fuel; RW – radioactive waste.

Annex G. The NEA/EGIRM workshop; practical exercises (case tasks and the presentation of results)

The workshop participants were invited to break out into five groups. Groups were given the time to study, discuss and fill in the tables.² Each group received five tasks prepared by the NEA Expert Group on Waste Inventorying and Reporting Methodology (EGIRM) (see below) as hypothetical national situations in spent fuel/radioactive waste management.

Note: the data in the tasks proposed by the EGIRM have no relevance to real country inventories, management strategies and disposal routes; the tasks were developed only for the objectives of this session.

In the tasks, the EGIRM provided brief descriptions of the national situations in relation to spent fuel/radioactive waste management. Each task was presented in a unique format; different possible situations in countries are described, and groups were proposed to interpret the proposed inventories in the methodology format.

Country A

Country A has an advanced nuclear programme. Below is the inventory at the end of 2015.

Nuclear power plant spent fuel

The national management strategy for spent fuel is direct disposal.

The country started a deep geological repository (DGR) development programme about ten years ago. Currently, a siting process is going on. The DGR implementation is planned for 2035.

Waiting for disposal, the spent fuel is stored in several storages. In three storages at the reactor sites, at the inventory revision date, nuclear power plant (NPP) operators had:

- NPP “A” – 748 [tHM] – from pressurised water reactor (PWR) type;
- NPP “B” – 623 [tHM] – from PWR-reactor type;
- NPP “C” – 496 [tHM] – from boiling water reactor (BWR) type.

Also, the centralised storage facility “AA” was built and started the operation five years ago. By the inventory revision date, NPP operators had shipped a total 3 261 [tHM] to the facility “AA”, consisting of 1 230 [tHM] of BWR spent fuel with the rest being PWR spent fuel.

Research reactor spent fuel

In the country A, an enhanced nuclear research programme has been realised from the initial stage of nuclear energy implementation. At the time of inventory revision, there were four research reactors shut down and two research reactors in operation. The spent fuel is removed from three of the shutdown reactors and stored in storages at the reactor sites:

- 3 [tHM] at the reactor N1;
- 1.5 [tHM] at the reactor N2;

2. Completed tables for each case are attached (presentation schemes A-E).

- 2.2 [tHM] at the reactor N3.

A fourth research reactor is still with fuel 1.6 [tHM] but not operated.

Two operating research reactors have produced 14 [tHM] of spent fuel that has been shipped to the storage “AA”.

Radioactive waste inventory

Country A has three classes of radioactive waste:

- **Class 1** – defined by the concentration of long-lived (≥ 30 y) alpha nuclides more than 400 Bq/g;
- **Class 2** – defined by the concentration of long-lived alpha nuclides less than 400 Bq/g;
- **Class 3** – defined by the concentration of long-lived alpha nuclides less than 10 Bq/g and short-lived beta-gamma nuclides less than tenfold exemption level.

The planned disposal route for Class 1 is in the DGR together with spent fuel. The country has 28 300 m³ of Class 1 waste (recalculated into “as disposed” volume) in the central storage facility “X”.

Class 2 radioactive waste is stored at three storages:

- storage 1 with 23 000 m³ [as disposed] of radioactive waste; this waste is to be disposed of in the existing near-surface facility “Y”; 4 800 m³ of Class 2 radioactive waste has already been disposed of there;
- storage 2 with 5 000 m³ [as stored], which is a mixed waste with toxic chemical compounds; for the moment, the country has not developed any disposal route for it;
- storage 3 with 1 200 m³ [as disposed] which is decided to be disposed in the DGR “Z”.

For Class 3 radioactive waste, the country policy is landfilling type disposal. There are 1 300 000 m³ of this type of waste stored in two sites, with 210 000 m³ already disposed in the landfilling sites.

The national inventory presented in the EGIRM format is given in Table 1.

Country B

Country B has no commercial nuclear programme. Below is the inventory at the end of 2015.

Research reactor spent fuel

In country B, a nuclear research programme has been performed in recent years. The country B has a contract with country C on the reprocessing of spent fuel. The resulting high-level waste (HLW) will be sent back to country B in the form of vitrified radioactive waste.

At the time of inventory revision, there were two research reactors in operation. The collected amount of spent fuel stored at the reactor site storages waiting for the shipment to country C:

- 17 [tHM] at the reactor N1;
- 2.3 [tHM] at the reactor N2.

38.3 [tHM] have already been shipped to country C, and according to the report from there, 6 m³ of HLW is stored waiting to be sent back. As stated in the contract, this corresponds to 5 [tHM] reprocessed.

Radioactive waste inventory

Country B has three classes of radioactive waste:

- **HLW** – defined in accordance with the International Atomic Energy Agency (IAEA) GSG-1 classification.
- **Intermediate-level waste (ILW)** – defined in accordance with the GSG-1 classification.
- **Low-level waste (LLW)** – defined in accordance with the GSG-1 classification. However, very low-level waste (VLLW) class is not separated from LLW. The lower boundary of LLW is aligned with the exemption level.

The disposal route for HLW considered in country B is in a DGR. However, the DGR development seems rather expensive for the country. Thus, the decision is to store waiting for an international solution for a DGR. For the interim storage, country B has constructed a storage facility for HLW.

23 m³ of ILW (raw volume) are also collected in the storage facility. The decision on the disposal route is postponed due to the reason mentioned above.

Country B has 16 870 m³ (not recalculated into “as disposed” volume) of LLW stored in several sites. Disposal in a well-engineered near-surface facility of the “El Cabril” type design is planned for this LLW. A further 60 m³ of LLW (recalculated into “as disposed” volume) is also stored waiting for disposal in the engineered near-surface facility.

An additional 7 800 m³ of LLW is already disposed in a simple near-surface facility of landfill type design.

Separately, country B considers the disposal of disused sealed radioactive sources (DSRS) and has a disposal facility for IAEA Category 3 and 4. In this facility, 46 500 pieces of DSRS have already been disposed at a depth of about 45 m. An additional 23 150 pieces have been collected and are stored awaiting disposal.

The national inventory presented in the EGIRM format is given in Table 2.

Country C

Country C has an advanced nuclear programme. Below is the inventory at the end of 2015.

NPP spent fuel

The national management strategy for NPPs’ spent fuel is reprocessing.

Spent fuel is stored in the centralised storage facility “BB” waiting for the reprocessing. To date, NPP operators have transported 16 000 [tHM] of NPPs’ spent fuel to this storage:

- 9 000 [tHM] from PWR type reactors;
- 7 000 [tHM] from light water graphite reactor type.

Also, country C provides a service to other countries for spent fuel reprocessing and has imported 38.3 [tHM] of spent fuel from country B; 5 [tHM] are already reprocessed, and 6 m³ of HLW are waiting to be sent back. Additionally, 45 [tHM] of spent fuel have been imported from country K for reprocessing.

Spent fuel from other reactors

The national strategy is reprocessing for all spent fuels.

In the country C, a nuclear research programme has been performed for many years. At the time of inventory revision, there were three research reactors in operation. The spent fuel, 142 [tHM], is stored in the storage facility “BB”. Also, country C has a nuclear-powered ice-breaker, and spent fuel (56 [tHM] from it is also stored in “BB”.

Radioactive waste inventory

The country C has three classes of radioactive waste:

- **HLW** – defined in accordance with IAEA Safety Series No. 111-G-1.1.
- **LILW** – defined in accordance with IAEA Safety Series No. 111-G-1.1; the country C provided the radioactive waste classification translation matrix to the IAEA Net-Enabled Waste Management Database (NEWMDB) where the rate LLW-ILW is specified as – 80-20%.
- **VLLW** – defined in accordance with IAEA Safety Series No. 111-G-1.1.

The chosen disposal route for HLW is in a DGR. The country has 3 200 m³ of vitrified HLW in the storage facility of reprocessing plant “Z”.

For the LILW, country C has a strategy of disposal in the underground facility “Y”, which is designed for non-heat emitting waste.

At the time of inventory revision, country C has 45 000 m³ of LILW to be disposed in the underground facility “Y” and has already disposed 26 000 m³.

For VLLW, country C has chosen the disposal in a near-surface facility with minimally sufficient engineering barriers. There are 257 000 m³ of this type of waste are stored in several sites awaiting disposal, and 156 000 m³ are already disposed in this near-surface facility.

In the past, country C engaged in the practice of sea dumping and disposed 26 000 m³ of liquid waste (LILW) and 5 600 m³ of solid LILW in this way.

The national inventory presented in the EGIRM format is given in Table 3.

Country D

Country D has an advanced nuclear programme. Below is the inventory at the end of 2015.

NPP spent fuel

The national management strategy for NPPs’ spent fuel is reprocessing abroad for PWR spent fuel, and not decided for the BWR spent fuel.

PWR spent fuel is stored in the at the reactor storage facilities awaiting for the reprocessing. In these storages, NPP operators have collected spent fuel:

- 8 200 [tHM] from PWR type reactors;
- 4 500 [tHM] from BWR type reactors.

Country D receives reprocessing services from the country F by an international agreement and, to date, has sent 2 300 [tHM] of PWR spent fuel to country F. 1 500 [tHM] are already reprocessed, and 800 m³ of vitrified HLW are waiting to be sent back.

Other reactors’ spent fuel

The state strategy is reprocessing abroad.

At the time of inventory revision, 8 [tHM] of spent fuel is stored in the at-reactor storage facility. 12 [tHM] have already been sent to country F for the reprocessing.

Radioactive waste inventory

Country D has four classes of radioactive waste:

- **Class A** – radioactive waste with heat emission over 2 kW/m³;
- **Class B** – radioactive waste with specific activity $1 * 10^3 - 1 * 10^6$ Bq/g alpha, $1 * 10^4 - 1 * 10^7$ Bq/g beta and $1 * 10^2 - 1 * 10^5$ trans-uranium;

- **Class C** – radioactive waste with specific activity $1 \cdot 10^2 - 1 \cdot 10^3$ Bq/g alpha, $1 \cdot 10^3 - 1 \cdot 10^4$ Bq/g beta and 10 – 100 Bq/g trans-uranium;
- **Class D** – radioactive waste with specific activity less than 100 Bq/g – alpha, $1 \cdot 10^3$ Bq/g – beta and 10 Bq/g trans-uranium.

The planned disposal route for class A is in a DGR. At the time of inventory revision, all HLW resulting from spent fuel reprocessing is stored in country F, awaiting eventual return (see above).

For class B, country D has a strategy of disposal mainly in an engineered underground facility. 23 500 m³ of ILW will be disposed in the planned DGR together with class A. An additional 650 m³ of class B are considered suitable for disposal in the existing well-engineered near-surface facility “W” along with the class C radioactive waste.

At the time of inventory revision, country D has 87 000 m³ of class C to be disposed in the near-surface “W” facility and has already disposed 34 000 m³ there.

For class D radioactive waste, the country has built a near-surface disposal facility “L” with minimally sufficient engineered barriers. 45 3000 m³ of class D waste are stored in several sites, and 13 4000 m³ are already disposed in the facility “L”.

The national inventory presented in the EGIRM format is given in Table 4.

Country E

Country F has no commercial nuclear programme. Below is the inventory at the end of 2015.

Other reactors' spent fuel

The national strategy is reprocessing abroad.

At the time of inventory revision, 8 [tHM] of spent fuel are stored in the at-reactor storage facility. 12 [tHM] of spent fuel have been sent to country G for reprocessing.

Radioactive waste inventory

Country E has four classes of radioactive waste:

- **HLW** – radioactive waste with heat emission over 3 kW/m³;
- **ILW** – radioactive waste with specific activity more than 400 Bq/g alpha, and $1 \cdot 10^4 - 1 \cdot 10^7$ Bq/g beta;
- **LLW** – radioactive waste with specific activity less than 400 Bq/g alpha, and $1 \cdot 10^3 - 1 \cdot 10^4$ Bq/g beta;
- **VLLW** – radioactive waste with specific activity more than exempt levels – up to $10 \cdot$ exempt level – alpha and $1 \cdot 10^3$ Bq/g – beta.

The disposal route for HLW is anticipated to be in a DGR. At the time of inventory revision, all HLW (6 m³) after spent fuel reprocessing were stored in country G. According to the agreement between countries E and G; this HLW will be stored until the country E implements a DGR.

For ILW, country E has a strategy of disposal in a near-surface facility with an advanced system of engineering barriers “ZZ”. 6 500 m³ (as disposed volume) of ILW will be disposed in this facility.

Also, country E has 230 m³ of liquid ILW, and there is currently no decided solution for its management. There is also a disposal module for LLW (with barriers sufficient for the safety provision) at the “ZZ” site. At the time of the inventory revision, country E has 23 400 m³ of LLW to be disposed in the “ZZ” and has already disposed 11 500 m³ there.

For VLLW, the country plans the disposal at the place of storage (in situ disposal). According to the disposal concept, VLLW storage facility will be transformed into a disposal facility by means creation of additional safety barriers. 78 600 m³ of VLLW of waste are currently and stored in this place.

The national inventory presented in the EGIRM format is given in Table 5.

Table 1. Country A

Spent fuel (SF) and radioactive waste (RW) inventory presentation												Additional routes								
Country: <u> A </u>			Date of inventorying: <u> 31.12.2015 </u>																	
Type of SF or RW (A)	Disposal route not yet defined (B)	Reprocessing of SF/RW (including international service)		Disposal in																
		(C1)	(C2)	UF-1		UF-2		NSF-1		NSF-2		BH-1		BH-2		BH-3		SDL	SDS	
				(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)	(I1)	(I2)	(J1)	(J2)	(K1)	(K2)	
1. SF																				
1.1. NPP SF, [tHM]				5 128 ¹																
1.2. Other reactors SF, [tHM]				20.7 ²																
2. Class 1 [m ³]				28 300																
3. Class 2 [m ³]	5 000 ³			1 200				23 000	4 800											
4. Class 3 [m ³]										1 300 000	210 000									
Equivalence with IAEA GSG-1 classification (type)	2.			ILW																
	3.	LLW		LLW				LLW	LLW											
	4.									VLLW	VLLW									

- 1) 5 128 tHM = 748 (PWR in NPP A) + 623 (PWR in NPP B) + 2 031 (PWR in "AA") + 496 (BWR in NPP C) + 1 230 (BWR in "AA").
 2) 20.7 tHM = 3 (N1) + 1.5 (N2) + 2.2 (N3) + 14 (in "AA"); 1.6 tHM (not included as is still in reactor).
 3) 5 000 m³ – as is (not conditioned).

Table 2. Country B

Spent fuel and radioactive waste inventory presentation												Additional routes								
Country: <u> B </u>			Date of inventorying: <u> 31.12.2015 </u>																	
Type of SF or RW (A)	Disposal route not yet defined (B)	Reprocessing of SF/RW (including international service)		Disposal in																
		(C1)	(C2)	UF-1		UF-2		NSF-1		NSF-2		BH-1		BH-2		BH-3		SDL	SDS	
				(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)	(I1)	(I2)	(J1)	(J2)	(K1)	(K2)	
1. SF																				
1.1. NPP SF, [tHM]																				
1.2. Other reactors SF, [tHM]				19.3 ¹	33.3 ²															
2. HLW [m ³]				6 ³																
3. ILW [m ³]				23 ⁶				16 930 ⁴												
4. LLW [m ³]										7 800										
5. DSRS [pieces]												23 150 ⁵	46 500 ⁵							
Equivalence with IAEA GSG-1 classification (type)	2.			HLW																
	3.			ILW																
	4.							ILW		LLW/VLLW										
	5.																			

- 1) 19.3 tHM = 17 (N1) + 2.3 (N2).
 2) 33.3 tHM (in country C).
 3) 6 m³ (in country C).
 4) 16 930 m³ = 16 870 (as is) + 60 (as disposed).
 5) DSRS of 3rd and 4th categories.
 6) As is (not conditioned).

Table 3. Country C

Spent fuel and radioactive waste inventory presentation													Additional routes						
Country: <u> C </u>			Date of inventorying: <u> 31.12.2015 </u>																
Type of SF or RW	Disposal route not yet defined	Reprocessing of SF/RW (including international service)		Disposal in															
				UF-1		UF-2		NSF-1		NSF-2									
(A)	(B)	(C1)	(C2)	(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)	(I1)	(I2)	(J1)	(J2)	(K1)	(K2)
1. SF																			
1.1. NPP SF, [tHM]		16 078.3 ¹																	
1.2. Other reactors SF, [tHM]		198 ³																	
2. HLW [m ³]		6 ²		3 200															
3. LILW [m ³]						45 000 ⁴	26 000												
4. VLLW [m ³]										257 000	156 000							26 000	5 600
Equivalence with IAEA GSG-1 classification (type)	2.			HLW															
	3.					80% LLW/20%ILW	80% LLW/20%ILW											80% LLW/20%ILW	80% LLW/20%ILW
	4.									VLLW	VLLW								
	5.																		

1) 16 078.3 tHM = 9 000 (PWR) + 7 000 (LWGR) + 33.3 (from country B) + 45 (from country K).

2) 6 m³ to be sent to country B.

3) 198 tHM = 142 (research reactor) + 56 (transport reactor).

4) Status is not clear (as is/as disposed).

Table 4. Country D

Spent fuel and radioactive waste inventory presentation													Additional routes						
Country: <u> D </u>			Date of inventorying: <u> 31.12.2015 </u>																
Type of SF or RW	Disposal route not yet defined	Reprocessing of SF/RW (including international service)		Disposal in															
				UF-1		UF-2		NSF-1		NSF-2									
(A)	(B)	(C1)	(C2)	(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)	(I1)	(I2)	(J1)	(J2)	(K1)	(K2)
1. SF																			
1.1. NPP SF, [tHM]	4 500	8 200	800 ¹																
1.2. Other reactors SF, [tHM]		8	12 ³																
2. Class A [m ³]			800 ²																
3. Class B [m ³]				23 500 ⁴				650 ⁴											
4. Class C [m ³]								87 000 ⁴	34 000										
5. Class D [m ³]										453 000	134 000								
Equivalence with IAEA GSG-1 classification (type)	2.		HLW																
	3.			ILW				ILW											
	4.							LLW	LLW										
	5.									VLLW	VLLW								

1) 800 tHM remaining, stored in country F (1 500 tHM reprocessed).

2) 800 m³ stored in country F.

3) 12 tHM sent to country F.

4) Status is not clear (as is/as disposed).

Table 5. Country E

Spent fuel and radioactive waste inventory presentation												Additional routes							
Country: <u>E</u> Date of inventorying: <u>31.12.2015</u>																			
Type of SF or RW	Disposal route not yet defined	Reprocessing of SF/ RW (including international service)		Disposal in															
				UF-1		UF-2		NSF-1		NSF-2								BH-1	
(A)	(B)	(C1)	(C2)	(D1)	(D2)	(E1)	(E2)	(F1)	(F2)	(G1)	(G2)	(H1)	(H2)	(I1)	(I2)	(J1)	(J2)	(K1)	(K2)
1. SF																			
1.1. NPP SF, [tHM]																			
1.2. Other reactors SF, [tHM]		8	12 ³																
2. HLW [m ³]			6 ²																
3. ILW [m ³]	230 ³							6 500											
4. LLW [m ³]								23 400 ⁴	11 500										
5. VLLW [m ³]										78 600									
Equivalence with IAEA GSG-1 classification (type)	2.		HLW																
	3.	ILW																	
	4.							ILW											
	5.							LWW	LLW	VLLW									

- 1) Sent to country G.
- 2) Stored in country G.
- 3) Liquid ILW.
- 4) Status of waste is not clear (as is/as disposed).

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National Inventories and Management Strategies for Spent Nuclear Fuel and Radioactive Waste

Radioactive waste inventory data are an essential element in the development of a national radioactive waste management programme since these data affect the design and selection of the ultimate disposal methods. Inventory data are generally presented as an amount of radioactive waste under various waste classes, according to the waste classification scheme developed and adopted by the country or national programme in question. Various waste classification schemes have thus evolved in most countries, and these schemes classify radioactive waste according to its origin, to criteria related to the protection of workers or the physical, chemical and radiological properties of the waste and the planned disposal method(s).

The diversity in classification schemes across countries has restricted the possibility of comparing waste inventories and led to difficulties in interpreting waste management practices, both nationally and internationally. To help improve this situation, the OECD Nuclear Energy Agency (NEA) developed a methodology that ensures consistency of national radioactive waste and spent fuel inventory data by presenting them in a common scheme in direct connection with accepted management strategy and disposal routes. This report provides the final version of the methodology and presenting scheme for spent nuclear fuel and the radioactive waste of all existing types. Additionally, there are recommendations in the report on how to enhance the comparability of national inventory data using the NEA methodology. The NEA support for joint efforts of the International Atomic Energy Agency and the European Commission on harmonisation of the reporting process by member countries to the Joint Convention and European Council Directive 2011/70 EURATOM is also presented in the report.