CONDITIONING; DOMAIN INSIGHT

Authors: Maxime FOURNIER, Florence BART (CEA)

Reviewers: Lionel CAMPAYO, Céline CAU DIT COUMES, Jean-Baptiste CHAMPENOIS, Kahina HAMADACHE, Jacques LACOMBE, David LAMBERTIN (CEA)

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OVERVIEW

After their characterization, waste stabilization by conditioning prior to long-term storage and deposition is a key activity of radioactive waste management. Radioactive waste conditioning consists of transforming the waste into packages suitable for transport, short or long-term storage and disposal. This is a mandatory step for all radioactive wastes, coming both from the operation of fuel cycle facilities (industrial and research) and from dismantling and decommissioning (D&D) of end-of-life nuclear facilities. These wastes can be specific according to one or other of these origins — *e.g.*, fission products for fuel cycle operation and graphite for D&D operations — or common — *e.g.*, adsorbents, sludge. If the waste from fuel cycle facilities is mostly well known and their conditioning routes identified, the volumes of D&D waste are yet to come, often poorly known, and exhibiting a great physico-chemical variability.

The waste package constitutes the first engineered confinement barrier to isolate the waste from the biosphere. Conditioning may or may not be preceded by waste treatment steps. A viable package does not show any surface contamination and is able to be handled before being directed to a suitable storage site. Its shapes are adapted to its contents, as well as storage and disposal facilities.

Depending on the waste radiological classification, the conditioning operations are more or less complex to perform. Thus, waste with the lowest radiological activities, termed as very low-level waste (VLLW) or low-level waste (LLW), is usually conditioned in big bags or crates before surface disposal without any major challenges. On the other hand, the conditioning of intermediate and high-level long-lived waste (ILW-LL and HLW) is much more challenging both on technological — processes having to operate in the presence of very high radioactivity — and chemical levels, especially when considering the materials choice — depending on the physico-chemical nature of the waste, its radioactivity, the heat to be evacuated, etc.

A package consists of a container made of metal or an alloy (*e.g.*, steel, stainless steel, non-ferrous metals such as copper) or a composite based on hydraulic binder (*e.g.*, fiber concrete) and of a content made up of waste, whether compacted or

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not, and possibly blocked or incorporated into a material called a confinement matrix. This matrix must ensure the confinement of the radionuclides over a long period against various attacks, in particular that of water. Given the diversity of treated waste, several matrices have to be designed: its choice and optimization obey long and well-controlled research and development (R&D) processes. The criteria for choosing a matrix suitable for the waste to be conditioned are typically its chemical compatibility with the waste, its resistance to water — in particular the absence of labile activity —, its mechanical properties, its resistance to radiation, etc. The main matrices are: (i) cementitious matrices (*e.g.*, for sludge, evaporation concentrates, incineration ash) (ii) bitumen matrices (for sludge and evaporation concentrates), (iii) vitreous matrices (in particular for fission products solutions) and (iv) polymer matrices (for ion exchange resins). Such matrices must be compatible with the acceptance criteria of the sites on which they will be stored, if they exist.

To conclude, conditioning gathers all operations consisting in introducing the waste, possibly pretreated, into a container where it can be blocked or incorporated into a matrix to form a waste package. The choice of the container and the matrix is mainly linked to the radiological and physico-chemical characteristics of the waste. It also aims to optimise the conditioned volume of waste and to comply with the waste acceptance criteria applying to storage and disposal facilities.

KEYWORDS

Barrier, Conditioning, Direct conditioning, Disposal, Leaching, Matrix, Package, Predisposal, Radioactive waste, Storage

KEY ACRONYMS

D&D - Dismantling and decommissioning

GBS – Goals Breakdown structure

HLW – High-Level Waste

IAEA – International Atomic Energy Agency

ILW - Intermediate Level Waste

LLW – Low-Level Waste

R&D - Research and development

VLLW - Very Low-Level Waste

1 TYPICAL OVERALL GOALS AND ACTIVITIES IN THE DOMAIN OF INVENTORY

This section provides the overall goal for this domain, extracted from the EURAD Roadmap goals breakdown structure (GBS).

Domain Goal	
2.2.3 Conditioning of radioactive waste (sub-theme of 2 Pre-disposal, 2.2 Implementation)	
Domain Activities	
Phase 1: Planning and Program Initiation	Establish waste categorisation (nature and radiological activity), eventually taking into account disposal options and waste acceptance criteria.
Phase 2: Program Implementation	Product a waste package satisfying the decided requirements and criteria, and compatible with the possible waste treatments, transport, storage and disposal.
Phase 3–4: Program Operation/Optimisation and Closure	Check the waste package compatibility with the critical characteristics and properties.

2 INTERNATIONAL LEGISLATION

National **governments** are responsible for putting in force the **regulatory framework** for the safe management of radioactive waste. For waste conditioning, as for all other radioactive waste management activities, the following aspects are taken into account: human health, safety issues, environmental impacts, radiation risks and legacy for future generations. Such analysis is not limited to the extent of the national territory and potential impacts for neighboring countries must also be taken into account.

Therefore, **national policies and strategies** are defined for the management of radioactive waste (IAEA, 2006; IAEA 2016a). They must cover all categories and volumes of waste, all steps of their management, including their interdependencies and their compatibility with existing or planned storage facilities, disposal routes and facilities. The policy must define the **roles and responsibilities of the stakeholders** in the management of radioactive waste. In order for the radioactive waste management to be safe, the **regulatory body** must enact requirements to serve as the basis for regulatory decisions. It makes available adapted principles, procedures and guidance — with associated criteria — for operators to meet the requirements. **Operators** and their subcontractors are responsible for the safety of the conditioning operations they undertake in compliance with national legal and regulatory requirements. They carry out safety demonstrations by means of safety cases and periodic safety reviews, enhancing and ensuring safety culture (IAEA, 2009; IAEA 2016b).

3 GENERIC SAFETY ISSUES

3.1 Planning and Program Initiation

Waste is categorised according to the nature of the radionuclides it contains and their radiological activity. At a finer level, this categorisation also takes into account existing and future storage and disposal options. Indeed, the conditioning options have to satisfy the established or anticipated acceptance criteria. Thus, regulators and operators for transport, storage and disposal will be involved in the choice of conditioning options.

3.2 Program Implementation

The conditioning of radioactive waste includes all operations enabling the production of a package satisfying the requirements and criteria, and compatible with the previous steps (treatments and pre-treatments) and the following steps (transport, storage and disposal) of waste management. From these basic principles, some practical rules can be summarized (IAEA, 2009).

- **Confining radioactivity** for as long as possible to protect natural resources from any pollution and to reduce as much as possible controls, monitoring and maintenance for the operation of storage facilities for future generations (ALARA “as low as reasonably achievable” principle).
- **Reducing the volume of waste** as much as possible.
- Limiting both volumes of **secondary waste** produced and **effluent discharges**.
- **Solidifying** the waste if it is not already solidified: convert **liquid waste in a solid form** and immobilize **dispersible waste** using an appropriate matrix.
- Enclosure of the waste and/or matrix in a **container** — and an appropriate **overpack** if needed — in order to interpose multiple barriers between the waste and the biosphere.

3.3 Program Operation and Closure

To the extent practicable, a package with the following characteristics and properties should be obtained (IAEA, 2016c):

- physical and chemical compatibility of the waste with any matrix materials and with the container,
- homogeneity,
- low void content,
- low permeability and leachability,
- chemical, thermal, structural, mechanical and radiation stability for the required period of time,

- resistance to chemical substances and microorganisms. Critical issues, information, data or knowledge in the domain of Inventory

4 CRITICAL ISSUES

Radioactive waste, possibly in **heterogeneous mixtures** and of **poorly known compositions**, has very **variable radiological and physico-chemical characteristics**. If a large volume of them can be managed without particular difficulty, some lead to **complex scientific and technical issues**. Their management is a major challenge from a technical, economic, regulatory, environmental and industrial point of view. Their potential toxicity requires protecting humans and the environment by **isolating radioactive waste**. To do so, several protective barriers are implemented: the package and its conditioning matrix are the first of them. Such packages are compatible with the main steps of waste management — transport, storage and disposal — it has to be **easily handled in compliance with safety and radiation protection rules**.

The development of processes and techniques for conditioning radioactive waste is a necessity, aiming at the following objectives:

- **reducing the volume or activity of waste** in order to optimize its distribution between the dedicated management routes and the use of disposal resources,

- producing matrices of **repeatable quality** compatible with available containers and **acceptance criteria** of storage sites when they exist, controlling the **package safety** in disposal, by **transforming the waste into the most inert physico-chemical form**, ,
- enabling **management of waste without existing route** by defining their treatment and conditioning process in order to make the waste compatible with existing or future routes.

In order to **limit the number of packages to be produced** (saving space in storage and disposal facilities, cost savings), **the waste loading in the matrix are as high as possible without degrading the desired package properties**. This requires many optimizations and very strict practices, in particular aimed at the development of **new formulations of conditioning matrices**.

The conditioning of waste has to be **industrialized as easily as possible** while **limiting the volume of secondary waste** generated by the process. As such, **standardization** is increasingly sought after. Conditioning operations are always carried out under quality assurance control in accordance with international standards.

Waste conditioning not only takes into account formulation of conditioning matrices, processes implementation but also the **packages behaviour under storage and disposal conditions**. With a view to long-term management of packages, the quality of this barrier has to be assessed over time. Given the timeframes to be considered — especially for geological disposal — a simple extrapolation over time of R&D laboratory-scale results is not sufficient. The first step consists in understanding and prioritizing the phenomena occurring during the package lifetime, through experiments or the observation of natural and archaeological analogues. Based on this understanding, packages evolution can be described using **models**.

5 MATURITY OF KNOWLEDGE AND TECHNOLOGY

Since the 1950s, many conditioning processes have been studied and developed to manage the radioactive waste produced by nuclear facilities. These conditioning operations lead to a variety of packages, from 200 L metal drums in carbon steel to metal or concrete boxes of several cubic meters. With the exceptions of direct packaging and compaction, the other conditioning operations implement steps during which the physico-chemistry of materials plays an important role, making the ILW-LL and HLW packages high technology objects. The matrices most commonly used are hydraulic binders, bitumen, glasses and polymers.

5.1 Spent fuel direct conditioning

Direct conditioning mainly applies to spent fuel assemblies to be placed in deep geological disposal (Figure 1). Such a management option is available for countries wishing to implement it instead of the spent fuel treatment-recycling strategy. The containers developed are one of the engineered barriers necessary to ensure the confinement of radioactivity (especially in the absence of a confinement matrix). Such developments are very important because the spent fuel rods by themselves cannot guarantee the long-term radionuclides confinement.

Figure 1. Copper containers designed by SKB (Sweden) for direct conditioning of spent fuel assemblies.

5.2 Hydraulic binders

Cements

Cementation is the **most widely used process for the conditioning of low-level and intermediate-level radioactive waste**. Indeed, cementitious materials have many advantages: availability, low cost, easiness of implementation, flexibility (*i.e*, ability to confine many physico-chemical forms of waste), insolubilization of a large number of radionuclides, good mechanical resistance and stability over time for well-designed materials. These materials are prepared by mixing anhydrous cement (mainly comprising calcium silicate and calcium aluminate phases), aggregates of various sizes (including sans and filler), water and admixtures. The presence or absence of aggregates, their size and the water-to-cement ratio lead to distinguish the following materials.

- **Pure pastes**, consisting solely of cement and water.
- **Grouts**, pure pastes or fine mortars containing little sand and rich in water. This composition gives grouts a rheology favourable to flow after mixing. Grouts are mainly used as matrices for **embedding** waste, produced either by mixing inside the container or by continuous mixing outside the container.
- **Mortars**, containing aggregates (sand) smaller than 6.3 mm. Mortars are used for **blocking operations** (*i.e.*, blocking of massive waste in a container or blocking of a primary container in a secondary container) carried out by injection into existing voids to form a **monolith** (Figure 2 Blocking in cement of Magnox fuel cladding swarf (Image: Savannah River National Laboratory).).
- **Concrete**, including, in addition to sand, aggregates sized between 6.3 mm and 80 mm. To increase its tensile strength, concrete can be reinforced with an iron framework or short metal fibres. Concrete is used for the manufacture of **containers** and the production of **structural elements** in storage and disposal sites.

In the presence of water, cements form hydrates by dissolution-precipitation, which are organized into a **cohesive structure**. After hardening, they are resistant to water. The most commonly used cement is **Portland cement**: it results from the grinding of clinker — an artificial rock produced at around 1,450°C from limestone (80%) and clay (20%). At high temperature, the chemical elements of the clinker recombine to give crystalline phases. In the presence of water and a calcium sulfate source (anhydrite or gypsum), these phases form calcium silicate hydrates (70%), calcium hydroxide or portlandite (20%), hydrated calcium aluminates and sulfoaluminates. For waste conditioning applications, Portland cement is often blended with blast furnace slag and/or fly ash. These supplementary cementitious materials contribute to the properties of the hardened product through hydraulic or pozzolanic activity. For instance, they make it possible to decrease the temperature rise resulting from the exothermic dissolution of anhydrous cement phases, and to refine the pore network of the hardened material, thus improving its confining properties. In addition, various **additives** can be added to the cement matrix cement (*e.g.*, barium oxide, iron oxides, zeolites, organic compounds) in order to **modulate its properties** (*e.g.*, workability in the fresh state, radiological protection, mechanical resistance and radionuclide retention).

Figure 2 Blocking in cement of Magnox fuel cladding swarf (Image: Savannah River National Laboratory).

Overall, cements show **good resistance to irradiation** for cumulative doses of up to 10^{10}-10^{11} Gy, without mineralogical transformation. The main consequences of irradiation are due to the **radiolysis of water** and the **production of dihydrogen gas** (H_2), which causes a safety problem. Insignificant in the majority of cases, the radiolytic production of H_2 is notable for a few families of packages characterized by a high β, γ or α activity.

Once hardened, cement is a **heterogeneous material** consisting of a porous solid (micropores and capillaries of 0.05 to 1 µm), a liquid phase and usually a gaseous phase present in the pores. The solid phase is formed of hydrated minerals and possibly residual anhydrous cement. The interstitial solution is strongly basic (pH between 12.5 and 13.6) and its composition varies with the material ageing. **The basicity of the interstitial solution enables the insolubilization of a large number of radionuclides**, this capacity being even reinforced in some cases by the action of the products of cement degradation by water.

Geopolymers

Geopolymers are particularly interesting materials for **conditioning problematic waste**. Although their industrial implementation has begun in a few countries (*e.g.*, Czech Republic, Slovakia), these matrices are still the subject of **development work**. Geopolymers are **aluminosilicate inorganic polymers** synthesized by alkaline activation of alumino-silicate materials and composed of cross-linked chains of silicon and aluminum tetrahedrons sharing oxygen atoms. Geopolymers exhibit an amorphous structure and a local order similar to the one of zeolites. The interstitial solution has pH values close to those encountered for the usual cementitious materials (12-13). Geopolymer have **good mechanical properties** — typically around 40 to 50 MPa in compression — and good performance with respect to the main usual durability criteria of civil engineering. Although they remain to be further investigated, the work carried out to date confirms the good resistance of these materials to leaching by water. As in the case of cementitious materials, irradiation mainly leads to radiolysis reactions of the pore solution, therefore to the **generation of H_2** with a yield depending on the water content and composition.

Geopolymers are, for example, of interest for **magnesium fuel cladding immobilization** (Figure 3a). Indeed, the use of usual Portland cementitious materials causes some difficulties, the main one being the production of H_2 by magnesium corrosion. Therefore, it is interesting to have a binder that keeps the advantages of cements while being compatible with corrosion inhibitors such as geopolymers — rich in silicates and without calcium, thus avoiding the precipitation of fluorine.

Another possible application of geopolymers is the **direct conditioning of radioactive organic liquid waste** (RLOW) (Figure 3 . (a) Cutting of a 200 L drum containing surrogate Mg-Zn alloy waste blocked in a geopolymer matrix (Image: CEA). (b) Geopolymer samples including 30 vol.% of oil (Image: KIPT, PREDIS).b). The principle is based on the emulsification of RLOW in an alkaline silicate solution (so-called "activation solution") followed by the addition of an alumino-silicate source, leading to the RLOW-geopolymer setting. The advantage of RLOW direct conditioning is the absence of chemical interactions that can cause setting delays or negative effects on the final material. Moreover, it enables to achieve high RLOW incorporation rates of 40 to 50 vol.%. Such material exhibits a good leaching resistance since the RLOW droplets of 10 to 50 µm are dispersed in the

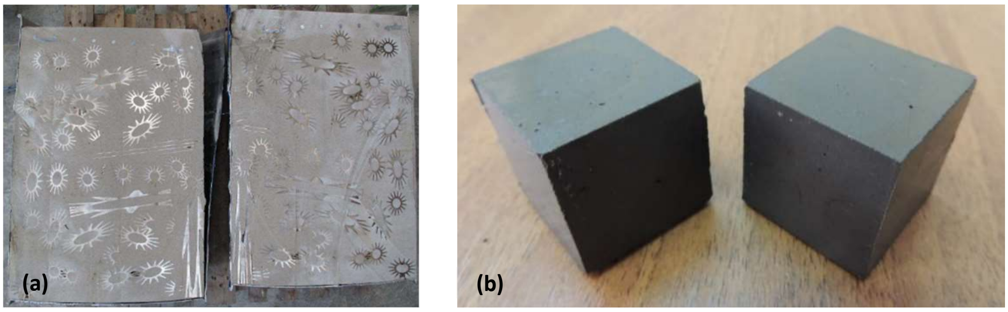

Figure 3 . (a) Cutting of a 200 L drum containing surrogate Mg-Zn alloy waste blocked in a geopolymer matrix (Image: CEA). (b) Geopolymer samples including 30 vol.% of oil (Image: KIPT, PREDIS).

geopolymer. The conditioning of various types of RLOW has been successfully tested.

5.3 Organic binders

Bitumen

Bitumen is the "historical" industrial matrix to condition **co-precipitation sludge** resulting from the treatment of liquid effluents. Indeed, bitumen has interesting properties with regard to its high binding power, its high chemical inertness, impermeability, low water solubility, high containment power, moderate cost and availability. However, it is an **organic matrix** — therefore **flammable** and **sensitive to radiolysis** — and it is often being **replaced by cementation or vitrification** depending on the nature of the waste to be treated.

Bitumen is the heavy fractions from the distillation of petroleum and are known in their natural state. It is a **continuum of organic compounds**, with molar masses varying between 400 and 4000 $g{\cdot}mol^{-1}$, mostly unsaturated and polycyclic. Its typical overall composition is around 80% carbon, 10% hydrogen, 5% sulfur, nitrogen and oxygen. The process of embedding by bitumen consists in mixing waste with bitumen at a temperature typically above 100°C. Such an operating temperature allow the evaporation of water from the waste and ease the mixing between bitumen and the waste. A typical waste loading in bitumen is of the order of 40%. The mixture obtained is dehydrated and poured into a container where it is cooled.

The mixture between bitumen and waste is carried out hot and the packages have a significant thermal inertia, so that they remain typically above 100°C for several hours. At this temperature level, the risk is the development of **exothermic reactions** causing a temperature increase, which can cause the bitumen to **ignite if the temperature exceeds 250°C**. Such risk is mainly mitigated upstream, when the potential reactivity of the waste-bitumen mixture is checked by **calorimetric measurements**. Thus, if the fire risk exists during manufacturing, it can be controlled as soon as the chemical reactivity is correctly characterized and taken into account.

The two phenomena to be taken into account to assess the mid- and long-term behaviour of bitumen are **radiolysis due to self-irradiation** of the package and the **water uptake** induced by possible contact of the package with water. Under irradiation, bitumen emits **radiolysis gases, mostly H_2**, which first dissolve in the matrix until saturation (≈1 vol.%). Beyond this limit, they form **gas bubbles**, the growth of which can lead to **swelling** of the bitumen that can cause **overflow** or **pressurization** of the package, requiring appropriate management. This swelling phenomenon can be managed by **limiting the activity and the package filling**, or the use of a **hydrogen scavenger** (*e.g.*, cobalt salts).

Although pure bitumen is **very slightly permeable to water** and dissolved species, the presence of salts in the material composition promotes a water uptake by diffusion and osmosis, leading to the swelling of the bituminized waste. In the case of a disposal containing bituminized waste packages without any residual porosity, this swelling could lead to the internal pressurization of disposal cells, which might be deleterious. The **radionuclide release rates that accompanied water uptake are slow** and compatible with disposal safety requirements. However and in order to mitigate its potential consequences, the swelling behaviour of bituminized waste packages has be taken into account during the disposal design.

Polymers

Polymers used for conditioning radioactive waste are usually **polystyrene**, **copolymers of polyesters and styrene** or **epoxy resins**. They have a **very low solubility in water** and **good resistance to biological agents and radiation** (this type of matrix only undergoes the effect of radiation for a short time since embedded waste are short lived). The process is implemented directly in the container and more rarely by pouring the mixture into the container. This conditioning method is used to embed **ion exchange resins** from reactors operation or to adsorb organic or aqueous liquids (Nochar “N” Series Polymer), thus generating **solid and stabilized waste**.

5.4 Vitrification

Vitrification is the **industrial reference for the conditioning of high activity fission products solutions** resulting from spent fuel reprocessing and military applications. It is used in France, Japan, Russia, the United Kingdom and the United States of America; some facilities have been in operation for over 30 years. Vitrification can be extended to lower activity level waste when the economics are appropriate: technological developments — *e.g.*, Cold Crucible technology or In-Can Vitrification — enables, for example, to limit the secondary waste induced by the process or to broaden its field of application.

Vitrification consists of mixing in a crucible and at **high temperature** (typically 1,100-1,300°C) radioactive waste — generally liquid or previously calcined — with a **glass frit** in order to integrate, **at the atomic scale**, all the radionuclides of the waste to the **vitreous network** in a homogeneous way. The material obtained after cooling is **monolithic and homogeneous** (Figure 4), without grain boundaries, which reduces the contact surface with any aggressive media, in particular the **underground water** of a geological disposal.

Figure 4. Nuclear R7T7 glass block (Image: CEA).

The confinement glasses are **tailor-made materials**: the chemical composition of the glass frit is adapted to be compatible with the waste to be vitrified and also to optimize the physicochemical properties of the material, which have to be satisfactory from the molten state to the solid state. The search for a nuclear glass composition — often in the **borosilicate** domain — is a compromise between the **material properties** and its **technological feasibility** on an industrial scale. The **melting process** is a complex succession of chemical reactions, often out of thermodynamic equilibrium, between the glass frit and the waste. Several stages follow one another: heating, primary melting, degassing, refining and homogenization; the mixture can be prepared directly inside the container (so-called In-Can vitrification) or poured into a stainless-steel container. Industrially, **glasses are easy to produce** there are few manual operations and the process can be easily adapted to the nuclear field.

Due to its chemical composition, **glass is very resistant to heating and irradiation, it has good chemical durability over thousands of years**. Assuming deep geological disposal of vitrified waste packages, the glass will be exposed to groundwater: **leaching of the vitreous matrix by water is the main factor that can lead to the release of radionuclides** into the environment. In addition, the highly radioactive glass package irradiates itself and the surrounding environment: **radiolysis of the underground water** can generate aggressive chemical species and **self-irradiation** can induce modifications of the vitreous network. Numerous theoretical and experimental works have led to the conclusion that the **glass properties would not be modified in the short, medium and long term**, due to irradiation (minor modifications of macroscopic properties) or temperature (negligible crystallization). The study of the mechanisms which control the leaching kinetics of nuclear glasses according to the environmental conditions enables to build **models** used to evaluate the performances of geological repositories and thus to ensure that their impact remains below the authorized limits.

Glass-ceramics

If glasses and ceramics (see “Alternative composite matrices”) each offer specific advantages, glass-ceramic materials can represent an interesting compromise.

These materials can be synthesized during the **glass-cooling scenario** or by carrying out an **additional thermal treatment** after the vitrification step. Easier to manufacture than most of the conventional ceramics, glass-ceramics have very good thermal and mechanical properties, and generally adapt better to variations in the waste composition. For example, the industrial vitrification of Mo-rich fission products solutions (produced after UMo metal fuel reprocessing) leads to the formulation of a glass-ceramic material characterized by Mo-P-Zn enriched nodules uniformly distributed in a glassy phase (Figure 5).

Figure 5. Nuclear UMo glass-ceramic block (Image: CEA).

In situ vitrification

In situ vitrification (ISV) is a **thermal soil rehabilitation technology** that uses electricity (through electrodes) to heat a radiocontaminated soil to high temperatures (1,600 to 2,000°C) in order to transform it into a **glass** (Figure 6). Then, the radiological activity of the soil is **confined in the glass**, which constitutes a **barrier preventing its dispersion**. The gases emitted during the vitrification process are captured and treated. This technology has already been implemented, including for pollution other than radiological.

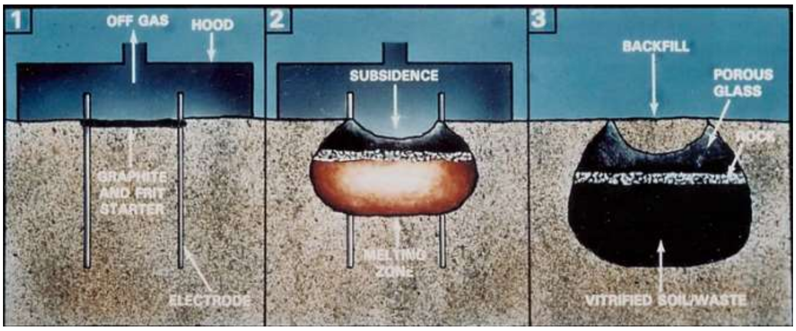

Figure 6. In situ vitrification process sequence (Image: U.S. Department of Energy).

5.5 Alternative composite matrices

Ceramics and Synroc®

While vitrification is the reference solution for conditioning high-level waste, conditioning some radionuclides in a glass matrix can be difficult because of a **low solubility in the glass network** or a **high volatility during glass melting at high temperature**. Then, ceramic matrices can be an interesting alternative solution, being **optimized for the radionuclide to be confined**, thus offering maximized properties. The study of these ceramic matrices combines experimental approaches and atomistic modelling, in order to determine the formulations exhibiting a good long-term behaviour — resistance to self-irradiation and leaching — and adapted to the radionuclide to be confined.

Examples (not exhaustive) for the confinement of specific radionuclides include:

- for **minor actinides**: zirconolite, britholite/apatite, monazite/brabantite or thorium phosphate diphosphate,
- for **caesium**: hollandite,
- for **iodine**: vanadium-phosphorous-lead apatite,
- for **chloride salts** (generated by pyrometallurgical processing of spent fuel): chlorinated apatite.

These ceramic conditioning matrices are still **far from industrial implementation**. It is not clear if the benefits they bring in terms of safety compensate for the additional cost linked to their manufacture. However, they could provide interesting solution for the management of waste associated with some future nuclear fuels.

One exception whose industrial implementation is more advanced: **Synroc® matrices**, developed by ANSTO, are ceramic matrices made up of **assemblies of several minerals** which, when mixed, can incorporate a large number of radionuclides present in HLW. The process uses **hot isostatic press** (HIP) at more than 1,200°C and 150 MPa. The liquid waste is first mixed with additives to obtain a slurry, which is dried and introduced into a container. The treatment at high temperature and pressure enables the material fusion to become a monolithic solid. The container reduced in volume to adopt a cylindrical shape representing approximately two thirds of the initial volume (Figure 7). Additionally, the feasibility of producing this matrix using the French Cold Crucible technology was established through controlled cooling after high temperature melting.

Figure 7 Synroc® container before and after HIP treatment (Image: ANSTO)

6 PAST RD&D PROJECTS ON WASTE INVENTORY

- **H2020 THERAMIN project (2017-2020)**

  The Thermal Treatment for Radioactive Waste Minimisation and Hazard Reduction (THERAMIN) project was a European Commission programme of work partly funded by the Horizon 2020 Euratom research and innovation programme. The objective of the project was to provide improved safe long-term storage and disposal of intermediate-level wastes and low-level wastes suitable for thermal processing. Work carried out within the project aimed to identify radioactive wastes that could benefit from thermal treatment, which treatment technologies were under development in participating countries, and how these could be combined to deliver a wide range of benefits.

THERAMIN successfully demonstrated the applicability of different thermal treatment technologies (incineration-vitrification, vitrification, thermal gasification, HIP) to a range of waste groups representative of those identified in participating countries. The thermally treated products were characterized and these data used to undertake preliminary disposability assessments. Finally, the project developed a value assessment methodology that can be used to identify the benefits and challenges of thermal treatment.

www.theramin-h2020.eu

- **European Joint Programme EURAD (2019-2024)**

  The European Joint Programme on Radioactive Waste Management (EURAD), which includes disposal, has been established to complement the national efforts and enables effective use of resources by fostering and strengthening RD&D collaboration. EURAD is a 5-year project which started in June 2019 and addresses all waste management-related topics. It is divided into 13 work packages applying of research and development, strategic studies and knowledge management. In particular, work package No 9 (ROUTES) aims to describe and compare the different approaches to characterization, treatment and conditioning of long-term waste management routes between member states in order to identify future relevant R&D topics.

  http://www.ejp-eurad.eu

7 UNCERTAINTIES

R&D activities on conditioning materials and their **optimization** follow very strict and rigorous methodologies. This work ensures the **safety** of transport, storage and disposal operations, in particular by enabling the compatibility of the packages with the national **acceptance criteria**, their possible evolution and their future definition when they are not yet known.

For waste generated by **industrial and research activities related to the nuclear fuel cycle** — and in particular regarding spent fuel from pressurized water reactors — conditioning techniques are well known and mastered. Ongoing continuous R&D work enables their optimization and **increases knowledge of the packages behaviour over the very long term** in storage and disposal situations. Although the core data have already been acquired, the most precise possible prediction of the long-term packages behaviour remains a scientific challenge. It requires to **combine experimentation and modelling** in order to overcome the multi-material and multi-scale complexity of future deep geological disposal. Moreover, new developments of conditioning processes and matrices will be necessary to cope with the **evolution of nuclear reactor technologies**: *e.g.*, adaptation of existing matrices in case of burnup increase and new matrices for new fuels (typically for molten salt reactors).

Some wastes coming from **D&D operations** represent new scientific and operational challenges. D&D activity remains relatively new and the **largest waste volumes are yet to come**. While most of this waste will not cause any management difficulty, some of it does not yet have an identified management route and will require **adaptations of conditioning techniques** already known and, if possible, implemented at high technological maturity levels. The upcoming conditioning matrices and their implementation processes will have to adapt to waste deposits with very **variable volumes** (including very small ones), as well as **uncertainties and variability of the physico-chemical characteristics** of “historic” waste deposits accumulated over decades.

8 GUIDANCE, TRAINING AND COMMUNITIES OF PRACTICE

This section provides links to resources, organisations and networks that can help connect people with people, focused on the domain of radioactive waste conditioning.

Guidance
• IAEA (1992). Treatment and conditioning of radioactive solid waste. International Atomic Energy Agency, IAEA-TECDOC-655. online • IAEA (1992). Treatment and conditioning of radioactive organic liquids. International Atomic Energy Agency, IAEA-TECDOC-656. online • IAEA (2009). *Predisposal Management of Radioactive Waste.* International Atomic Energy Agency, IAEA Safety Standards Series No. GSR Part 5. online • IAEA (2016a). *Governmental, Legal and Regulatory Framework for Safety*. International Atomic Energy Agency, IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Requirement 10, Vienna. online • IAEA (2016c). *Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities.* International Atomic Energy Agency, IAEA Specific Safety Guide No. SSG-41, "Processing of radioactive waste". online • IAEA (2017). Selection of Technical Solutions for the Management of Radioactive Wastes. International Atomic Energy Agency, IAEA-TECDOC-1817. online
Training
• IAEA, 12 online learning courses (50 modules and almost 100 lectures) on spent fuel and radioactive waste management, and decommissioning and environmental remediation. online • SCK·CEN Academy, training course on radioactive waste management. online • INSTN, international school in nuclear engineering: nuclear waste management module. online
Active communities of practice and networks
• NUGENIA (Nuclear Generation II & III Alliance), Technical area 5: waste management and decommissioning, https://snetp.eu/nugenia

Key competences that are needed in the area of radioactive waste conditioning include material science, harnessing of technologies, process engineering, radiation safety, radiological measurements and monitoring, data handling and preservation, risk management, programme management.

9 ADDITIONAL REFERENCES AND FUTURE READING

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GLOSSARY

Barrier (confinement barrier): a device able to prevent or limit dissemination of radioactive material.

Disposal: the action of radioactive waste emplacement in a facility specifically laid out to confine it in a potentially permanent way. **Deep geological disposal** is the disposal in an underground facility specifically laid out for this purpose.

Embedding (or encapsulation): the immobilization of radioactive waste through fixation within a material in order to obtain a solid, compact, indispersible, and stable product.

Fission products: nuclides generated either directly through nuclear fission, or indirectly through the disintegration of fission fragments.

Fuel assembly: the **fuel** is the constituent material of a nuclear reactor core, which contains the fissile nuclides. **Fuel rods** are gathered together in clusters, which are set in place with a definite position in the reactor core. The so-called "**assembly**" is the whole of this structure, grouping one hundred to a few hundred rods, which is loaded into the reactor as a single unit. **Spent fuel** corresponds to the fuel assemblies definitively removed from a nuclear reactor after a period of useful energy ouput.

Fuel cycle: the industrial operations which fissile materials are subjected to (ore mining, fissile material concentration, enrichment, fabrication of fuel elements, fuel use in reactors, spent fuel treatment, waste conditioning, and the disposal of the resulting radioactive waste.

Leaching: the contacting of a solid body with a liquid with the purpose of extracting some elements. By extension, refers to any experiment focusing on the alteration of a solid in a liquid.

Matrix: material enabling the encapsulation, blocking or incorporation of radioactive waste, first confinement barrier.

Package: the packaging with its specified radioactive contents, as presented for transport, storage and/or disposal.

Radioactive waste: any radioactive substance for which no subsequent use is planned or contemplated. In this document, "waste" refers exclusively to the substance before treatment and conditioning; in the literature, this notion can sometimes be confused with that of package.

Storage: the action of placing radioactive materials or waste temporarily in a surface or subsurface specially designed facility.