3.1.2 Vitrified HLW, Domain Insight

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Overview

Used nuclear fuel has been reprocessed in France, the UK and to a lesser extent Germany and Belgium. Reprocessing is a chemical extraction process that separates reusable fissile materials (e.g. ^{239}Pu and ^{235}U) from the constituents without future use contained in spent fuel, considered as waste. This highly radiotoxic waste (mainly fission products and actinides) is initially contained in nitric acid and must be immobilised into a very stable and durable waste form, able to contain the radionuclides for several thousands of years over manipulation, transport and storage to final disposal. To produce a glass, the highly radioactive nitric acid solution is then dried, calcined and mixed with glass formers (SiO_2, Al_2O_3, etc.) and modifiers (B_2O_3,Na_2O, etc.) and molten at temperatures >1,000°C. The resulting glass melt is then poured into stainless steel storage containers and allowed to cool, forming a solid and durable glass wasteform that contains the principal radionuclides in a single solid phase for interim storage, operations and eventual geological disposal. This process known as vitrification also serves to reduce the volume of high-level waste (HLW) that requires geological disposal. The glass wasteform can fracture upon cooling. Typical glass compositions are borosilicate. Glass compositions need to be carefully adapted to the composition of the waste and can only vary within a carefully documented specification envelope. Vitrification is a mature technology as glass melting in induction furnaces, and to a smaller degree in ceramic melters and cold crucibles has been successfully conducted for more than 30 years.

By the end of 2021, about 23,000 high level vitrified waste packages have been produced in the France [1] (19030 CSD-V, 3159 AVM and 731 CSD-U). As to 2021, the Savannah River Site Defense Waste Processing Facility (SRS DWPF) has produced ~8,000 tons of HLW glass and 4,242 canisters [3]. In addition to HLW glasses, there also exists some vitrified intermediate-level waste (ILW). For most waste streams, borosilicate waste glass compositions are used for special waste streams. Other compositions have also been applied and are being considered, for example Umo glasses (glass-ceramic matrix) for vitrification of high molybdenum and phosphorous content.

Vitrified waste typically has a very high inventory of radionuclides in the order of some billions of Becquerel per gram. It contains an important quantity of radionuclides with a long half-life, which are not merely embedded in the solid like in a cementitious matrix but are confined on atomic scale in the glass network structure.

Keywords

nuclear waste glass, radionuclide inventories, durability, leaching, secondary phases, radiolysis, radiation damage

Key Acronyms

Nuclear Waste Glass

1. Typical overall goals and activities in the domain of Vitrified HLW

This section provides the overall goal for this domain, extracted from the EURAD Roadmap goals breakdown structure (GBS). This is supplemented by typical activities, according to phase of implementation, needed to achieve the domain goal. Phases are not necessarily discrete but often overlap and are typically iterative (i.e. not simply sequential) and can differ from country to country. Activities are generic and are common to most geological disposal programmes.

Domain Goal	
3.1.2 Identify or confirm the properties, behaviour and evolution of nuclear waste glass under storage and disposal conditions, including the impact on disposal environment (Nuclear Waste Glass).	
Domain Activities	
Phase 1: Programme Initiation	Conduct initial nuclear inventory analyses of waste glass, including projections of future glass production, glass types for various waste streams considering fuel cycle evolution, special waste types including historic wastes and expected nuclear energy use and provide safe long-term interim storage sites.
Phase 2: DGR Site Identification	Update inventory analyses and future projections of heat production, packaging options, space requirements and disposal options.
Phase 3: DGR Site Characterisation	Perform/use adequate studies on site-specific glass/repository interactions and iterate with EBS design and disposal space requirements evaluations; verify interim storage impact.
Phase 4: DGR Construction	Confirmation/actualization of models for nuclear waste glass performance under repository and EBS conditions to support repository license application.
Phase 5: DGR Operation and Closure	Confirm and preserve documentation for emplaced radionuclide inventory for license application for operation/closure. Establish monitoring techniques to confirm expected heat load.

2. Contribution to generic safety functions and implementation goals

This section describes the safety concerns of canistered nuclear waste glass during long-term interim storage and how it (and its specifications and associated information, data, and knowledge) contributes to high-level disposal system requirements using EURAD Roadmap Generic Safety and Implementation Goals (see Domain 7.1.1 Safety Requirements). It further illustrates, in a generic way, how such safety functions and implementation goals are both fulfilled during storage, transport and disposal. It is recognised that the various national disposal programmes adopt different approaches to how interim storage and disposal system requirements are specified and organised. Each programme must develop its own requirements, to suit national boundary conditions (national regulations, different types of nuclear waste glasses, different packaging concept options, different host rock environment, etc.). The generic safety functions and implementation goals developed by EURAD and used below are therefore a guide to programmes on the broad types of requirements that are considered, and are not specific or derived from one programme, or for one specific disposal concept.

2.1 Features, characteristics, or properties of Vitrified HLW that contribute to achieving storage safety as well as long-term safety of the disposal system

2.1.1 Primary goal - relied upon for safety over many decades of dry storage

NUCLEAR WASTE GLASS STORAGE SAFETY - Dry storage is typically organised in large thick walled dual-purpose casks or in pits in concrete structures with evacuation of the heat generated by natural or forced convection. The design assures sub-criticality, radiation shielding and decay heat removal. During long-term dry storage of nuclear waste glass canisters, safety concerns are demonstration, monitoring and potential repair of the tightness of the containers, assuring the safe enclosure of radioactive materials, the safe removal of decay heat, transportability and fire resistance and avoidance of unnecessary radiation exposure. Typical initial dose rates are around 1.4 x104 Gy/h at the glass surface. Thermal heat power is approximately 2.5 kW per canister at the time of production, 1.6 kW after 10 years, and 0.6 kW after 50 years (Fig. 1). After 100 years, ^{241}Am with a half-life around 430 years will have a more important contribution to heat power in comparison to ^{137}Cs/^{137}mBa and ^{90}Sr/^{90}Y [1].

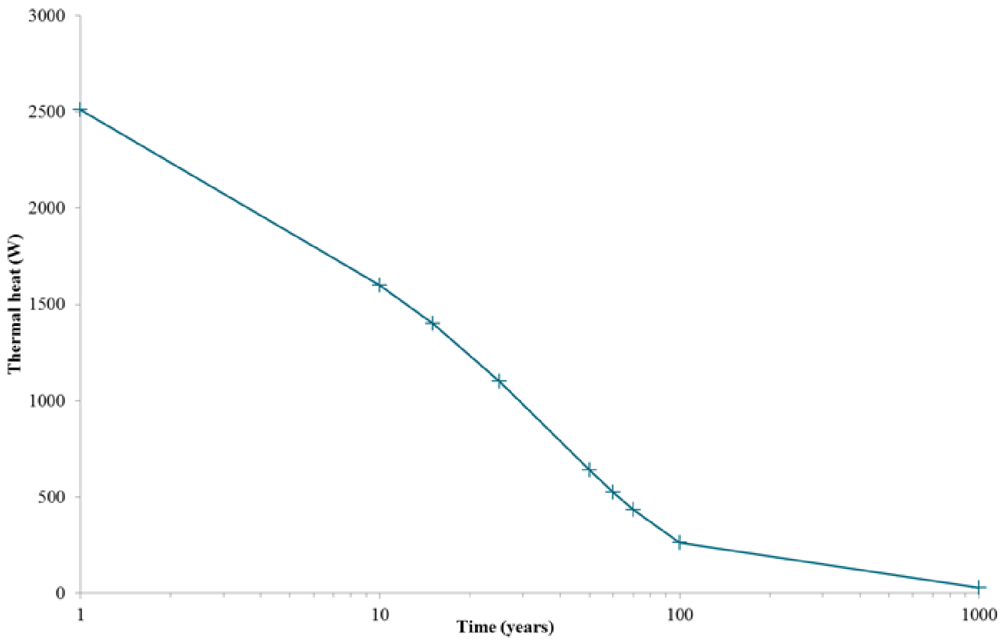

Figure 1: Thermal heat decay for a typical canister produced in La Hague vitrification facilities [1].

By dimensioning of the interim storage system and by ventilation it must be assured that during storage the glass temperature is always kept lower than 500°C. The maximum allowable temperature is defined by the glass crystallisation characteristics. With regards to French R7T7 glass for example, the minimum crystallisation temperature is about 610 °C. Additionally, in case of storage in concrete pits, the concrete temperature limit must be kept lower than 90°C [1]. Scientific studies of glass behaviour under the conditions of an interim storage facility, i.e. a glass submitted to a specific thermal history and to a self-irradiation induced by both beta and alpha decays have shown that the glassy state will be preserved throughout the duration of the interim storage period and that the main function of the glass that is to contain the radioactive isotopes, will also be maintained [1].

2.1.2 Primary goal - relied upon for limiting the thermal load to a repository

There is a thermal waste acceptance criteria imposed by the conception of the extension of the HLW zone of the repository. This criterion assures that temperatures experienced by the repository rock do no increase beyond the thermal threshold imposed for the repository formation considering a reasonable packaging density for optimal use of the repository space. It is for example about 500 W in the French disposal concept. This criterion can be achieved by adapting long-term interim storage times.

2.1.3 Primary goal - relied upon for long-term repository safety

CONTAINMENT RETENTION AND RETARDATION - by providing a chemically very water and radiation resistant solid amorphous matrix, confining the safe immobilisation of the major radionuclide inventories (minor actinides, fission and activation products, exception: halides like ^{129}I) remaining after reprocessing in a glass matrix composed largely of inactive glass formers and modifiers such as SiO_2, Al_2O_3, B_2O_3 and Na_2O. Nuclear waste glasses are never completely homogeneous amorphous solids but contain a few percent of bubbles, noble metal inclusions or refractory oxides, and a few other immiscible components. Only minimal evolution (limited extend of crystallisation and radiation damage) of HLW glass is expected in a repository prior to groundwater access. The stress relieving of the disposal canister (to avoid the risk of stress corrosion cracking at the weld) has to be observed in order to limit the temperature in contact with the glass.

The goal is to limit radionuclide release also after rupture of the disposal container and the stainless steel canister caused by corrosion and geomechanical forces after many thousands to hundreds of thousands of years and subsequent groundwater access to the glass matrix.

In some repository designs, the glass product may comes in contact with vapour prior to groundwater. Upon potential groundwater (vapour) access, the glass matrix remains largely stable, showing only a very slow dissolution rate. Secondary crystalline and amorphous phases will form slowly at the glass surface incorporating the sparingly soluble radionuclides and thus forming a new barrier against radionuclide release. The by far largest part of the most toxic radionuclide (actinides, Tc, etc.) inventories remain only sparingly soluble, limiting any transport by groundwater. Depending on the disposal concept and the intended relative importance of the glass in the overall engineered barrier system, the glass is expected to retain its integrity over tens of thousands to millions of years. Only very small fractions of the soluble radionuclides become released upon water contact.

Measurable or quantifiable actions for enhancing the long-term safety function in repository design:

- Preventing glass/water contact for a very long time (1,000 to 10,000 years, e.g. by a barrier system composed of disposal containers and buffer materials), particularly to ensure full containment during the thermal phase of few thousands of years, where glass corrosion rates are higher.
- Enhance the stability of the nuclear waste glass by assuring geochemical and hydrological conditions, which assure low long-term leach rates by:

  a) optimisation of the near-field environment by creating and maintaining favourable geochemical conditions. Corrosion rates are expected to be high at high pH, while they are expected to be low in case of high concentrations of dissolved silica (affinity effect). High Mg concentrations in glass or near-field may increase glass dissolution rates. Also Ca concentrations may have an effect on glass dissolution rates.

  b) slow groundwater flow limiting the rate by which soluble radionuclides can move away from the container, following eventual container breach and glass leaching.

- In designing the overall near-field around the disposed waste glass, assess the strong coupling, which might exists between the barrier function of the glass and the near-field environment: glass corrosion rate may increase by interaction with container corrosion products or by a potentially present alkaline environment from cementitious materials. These accelerating effects may be limited in time. Glass may impose pH gradients in the near-field.

- Radiation effects are reported but they are often less important than the above mentioned factors. However, in particular, ballistic long-term radiation damage might be an important factor to consider.
- It should be noted that it is the overall performance of the waste package (waste container for disposal surrounding the stainless steel canister filled with glass) as a whole, rather than that of its components, that is the governing factor in judging its transportability (in a specially designed transport container) as well as its stability in the operational and post closure phase.

2.1.4 Secondary goal – acknowledged but not relied upon for long-term disposal safety

STABILITY OF THE STAINLESS STEEL CANISTER - stability of the canister filled with the glass contributes to this generic safety function of the disposal system by limiting water access to the nuclear waste glass matrix. The canister may remain stable for tens of thousands of years, but while stability needs to be assured during long-term storage, this cannot yet be guaranteed for geological disposal. To benefit from the secondary safety function, it would need to be shown that the stainless steel canister will withstand mechanical loads, shear and creep and the risk of rupture needs to be quantified. Interactions between the container and the glass need also to be considered.

2.1.5 Illustrative requirements of critical information for both storage and long-term disposal safety

- Obtained by theoretical calculations of:
  - Radionuclide inventories
  - Decay heat loads
  - Gamma dose rate and its spatial and temporal evolution
  - Neutron source term
- Predisposal history, including container tightness and handling documentation.

2.1.6 Illustrative requirements of critical information for long-term disposal safety

- Glass composition and homogeneity of the glass product/quantity and composition of potential presence of segregated phases
- Radionuclide inventory and if available oxidation state of radionuclides in the glass matrix
- Surface area increase by fracturing upon cooling of the glass melt (a factor of 10 is typical)
- The resistance to thermal, microbial, radiation damage and to devitirification
- Gamma dose rate influencing overpack corrosion
- Thermal history of the glass
- For the case of groundwater access to the glass product after rupture of container in the long-term
- availability of parametrized glass dissolution models including for unsaturated conditions
- Expected long-term pH excursion at the glass/water/near-field boundary
- Initial and residual long-term glass dissolution rate at expected disposal conditions
- Solubility of key radionuclides at the glass/water interface
- Interaction scenarios for glass with near-field materials such as container materials or cementitious environments
- Most of this information is available from existing experimental and theoretical studies

2.2 Features, characteristics, or properties of nuclear waste glass canisters, that contribute to achieving long-term interim storage stability and feasible implementation of geological disposal

2.2.1 Primary goals – relied upon for storage and implementation of disposal

NUCLEAR WASTE GLASS OPERATIONAL SAFETY - The canistered HLW glass ensures safe manipulation during interim storage and transport.

- Radiation dose (gamma, neutrons) at the glass canister and the container surface (transport, experimentally verifiable)
- Safeguards: nuclear materials accounting
- Workers are not exposed to glass fines released from damaged canisters (if they exists)
- Stability of building structures against anthropogenic (air plane crash, terror, war) and natural (e.g. extreme weather) impacts
- Stability of glass containers in accident scenarios (e.g. during transport)
- Suitable monitoring techniques

RELIABILITY OF INVENTORIES - Assess radionuclide inventory and its evolution to the long-term – both for estimating heat production, neutron radiation, radiotoxicity and potential volatility of certain radionuclides in interim storage, operational safety and long-term post closure.

- Reliability of codes, data, waste loading, glass composition and materials accounting in glass manufacturing including accepted uncertainty ranges in compositions and other properties

NUCLEAR WASTE GLASS OPERATIONAL SAFETY AND HEAT OPTIMISATION

- The canistered HLW glass provides passive safety in the operational phase of geological disposal.
- Optimisation of distances among disposal containers with nuclear waste glass – considering maximum target heat production and choice of interim storage duration. Decay heat of nuclear waste canisters is a main dimensioning parameter for the layout of the disposal facility.
- The overpack (body and head) is welded to ensure it remains watertight for hundreds/thousands of years. The welds must also be stress-relieved without reaching temperature that could lead to glass crystallisation

3. International examples of nuclear waste glass disposal variants

This section provides a broad overview of typical examples for the different types of concepts for nuclear waste glasses considered for disposal in a DGR across Europe. Implementation goals, attributes, characteristics, features, and design requirements of glass disposal concepts vary between countries, but generic information is often valid across disposal programme boundaries.

High long-term stability of nuclear waste glasses is expected in many European repository concepts be it in clay, salt or granite formations.

- In France, the planned disposal of HLW glass in clay rock is facilitated by emplacing the canistered glass block in a thick steel overpack in a steel lined horizontal disposal location in the clay rock. A cementitious material is injected into the extrados of the lining (between the lining and the rock) to neutralise the acid transient resulting from clay oxidation. Glass is an important radionuclide release barrier for long-lived mobile nuclides such as ^{79}Se, ^{129}I and ^{36}Cl but the main barrier remains the host rock.

- In Belgium, the planned disposal of HLW borosilicate glass in clay rock is facilitated by emplacing the canistered glass block in a multibarrier system, putting the glass in a carbon steel overpack, surrounded by an OPC based concrete buffer inside a stainless steel liner. Waste package design is focused on the control of corrosion if the overpack. Release from vitrified waste is faster compared to concepts without cement but the strategic choice is not to rely too much on this safety function and put more effort on the overpack stability and the low solubility of radionuclides.
- In Switzerland, the safety barrier system for HLW in the planned disposal concept in clay rock is facilitated by putting the canistered glass in a disposal container in horizontal tunnels containing the waste packages, which are surrounded by emplaced granular bentonite.
- In the UK, standarised waste container designs have been proposed to accommodate three vitrified waste canisters. They would likely be fabricated from copper, with a cast iron insert in the case of disposal in a higher strength rock repository (granite) or from carbon steel for compatibility with repository environments in sedimentary rock or salt, although a final decision on the preferred HLW disposal concept has yet to be made in the UK.

4. Critical background information

The section highlights specific components, key information, processes, mechanism, data or challenges, that have a high impact or are considered most critical for the above mentioned impacts on disposal safety and implementation goals of nuclear waste glass both for the pre-disposal phase, the repository layout and the post-closure phase.

4.1 Pre-disposal

Nuclear waste glass composition design - While nuclear waste glass is a major industrial radioactive waste product with ongoing large scale production in Europe since 30 years, new glass formulations are continuously developed by national waste management programmes for specific waste streams (HLW, ILW, e.g. immobilising reactive metals, sulphur or molybdenum containing wastes) providing glass formulations in a well-controlled compositional space with predictable long term disposal stability [2].

To assess disposability, models for glass dissolution mechanism need to be developed for each targeted glass composition, allowing a long-term safety assessment. Depending on the uncertainties in the model parameters of a given glass product, either, conservatively, the fast initial rate or the much slower long-term rate can be used to assess release rates of soluble radionuclides as input to safety analyses of near-field performance.

Glass ageing during storage. Very little ageing is expected during long-term dry storage.

Availability of methods and techniques for preparation of nuclear waste glass canisters/storage casks for transport and disposal. As the disposal option may not be known at the start of interim storage, this includes potential disposition for repackaging of nuclear waste glass to adapt to a disposal concept, once selected.

4.2 Repository layout

Heat removal. Radioactive decay will generate heat in the nuclear waste glass. Most of the heat generated is due to attenuation of radiation by the glass matrix. This attenuation occurs locally for alpha and beta radiation, but is more widely distributed throughout the container for gamma and neutron radiation. This understanding is critical for modelling post-closure thermo-hydromechanical evolution in the near-field and is critical input data for repository layout optimisation (i.e., to calculate the spacing between disposal units, or disposal areas to meet specified thermal limits for components of the EBS or the host rock).

4.3 Post-closure

Alteration and dissolution of the glass matrix. On contact with water, the glass matrix will slowly alter, with formation of secondary phases, and eventually dissolve. This results in the slow incongruent release of radionuclides contained in the glass matrix. This process is controlled by the geochemical environment in the near-field and in particular the pH and dissolved silica concentration of water penetrating the waste package. This is a safety significant data value used in post-closure safety assessment, today largely accessible by well parametrized coupled reactive transport models, including but not limited to:

- Influence of fractures, both for saturated and unsaturated conditions. Influence of long-term rock stresses on the evolution of glass block fracturing of the glass package and thus on its surface area accessible to water.
- Interaction with canister: both stainless steel canister, and carbon steel overpack, including the question of the nature of the corrosion products and their influence on glass alteration.
- A main issue is to develop mechanistic computations tools able to better assess glass alteration taking into account the evolution of the surrounding conditions.
- Much research and development is done on R7/T7 glass (particularly SON68 glass) or international simple glass, but the level of knowledge is lower on other industrial glass.
- Influence of affinity/protective surface layers on glass dissolution rates.
- Influence of temperature, reactive surface area, hydrodynamics and water constituents, corrosion in vapour, in particular pH, dissolved Si, Mg, Ca, etc.
- Influence of corroding iron (container, etc.) in presence of the glass.
- Solubility, co-precipitation and sorption constraints for radionuclides at the glass/water interface. Look-up tables with maximum solution concentrations may be used under certain conditions.
- For most radionuclides, the oxidation state in the glass matrix is not important since upon release where it is changed by the redox state of the near field. For some nuclides like Se(VI) redox state changes are very slow and non-equilibrium redox states may persist and may become transported through the near field.
- Quantification of uncertainties.

Early stage disposal programmes can estimate and collect data (for input to preliminary safety analyses) by correlating with comparable glass compositions already well characterised under expected repository post-closure conditions (i.e., international data sets are widely available, for example from previous projects of the European commission quoted below). Advanced stage programmes are typically required to confirm, through detailed characterisation of their own glass formulations, more precise dissolution rate data, particularly using site- and concept-specific environmental conditions (principally, groundwater chemistry). An accurate assessment of glass behaviour in disposal relies on a good assessment of THMCR evolution of the high-level vitrified waste disposal cell (saturated/unsaturated conditions, corrosion of the metallic components, groundwater chemistry, etc.)

Homogeneity of distribution of the radionuclide inventory in the glass. While the distribution of radionuclides in the glass matrix is very homogeneous, some few percent of segregated phases (spinel, noble metal fines) might exist. These phases are typically very stable and do not provide an additional source term for release of radionuclides. This situation is different for certain glass compositions leading to larger scale phase separation and to formation of more soluble phases. Such glass compositions can be avoided in many cases by suitable design of the glass fabrication process including the appropriate choice of additives. One may encounter such phase separation for glasses designed for the solidification of difficult to vitrify waste streams like those containing large quantities of sulphate or molybdates. If this is the case, typical waste disposal programmes might be forced to model glass dissolution assuming that a relatively fast initial rate of dissolution will not slow down over the long term.

4.4 Integrated information, data or knowledge (from other domains) that impacts understanding of nuclear waste glass

- Understanding of the disposal environment (see Domain 3.4.1 EBS system). Corrosion mechanisms and corrosion rates for the nuclear waste glass vary depending on the post-closure environmental conditions. EBS configurations that may have a high impact or contribution towards long-term safety and nuclear waste glass stability are:
  - The overall duration of void closure influenced by host rock creep as well as by hydraulic/gas evolution: in clay rock for example, void closure at disposal locations may be slow (>10,000 years). The HLW glass could experience long-term contact with high humidity conditions after container breach but prior to facility re-saturation.
  - The water residence time, the pH of the aqueous media; the solubility limit of dissolved SiO_2 and the temperature. Less important is the pressure.
  - Particularly detrimental for glass stability are alkaline conditions, imposed in some EBS designs by the presence of cementitious materials.
  - There is a direct coupling between container corrosion and glass corrosion: Container corrosion products may adsorb dissolved silica until they become saturated and this consumption of silica reduces silica concentrations in the aqueous phase leading to an increase of glass dissolution rates. This effect is limited in time and bound to transport constraints for dissolved silica in the aqueous phase.
- Waste package evolution (see Domain 3.2.1 HLW and SF disposal containers). The time until container failure under post-closure conditions is important, as it determines the temperature at which glass/groundwater contact starts. Container lifetimes distributed over time are often designed to be longer than 500, 10,000 or even 100,000 years, to bridge the period of high thermal load of exothermic vitrified waste governed by a strong radiation field to assure low glass dissolution rates after container breach.
- Thermal evolution (see Domain 5.1.1 Design specification). Heat generation from the glass will impact the required disposal area (e.g., packing density).

5. Maturity of knowledge and technology

This section provides an indication of the relative maturity of information, data and knowledge for disposal of spent nuclear fuel. It includes the latest developments for the most promising advances, including innovations at lower levels of technical maturity where ongoing RD&D and industrialisation activities continue.

5.1 Technology

Vitrification technology has been mature since 30 years. Large quantities (about 6,000 tons in the EU) of nuclear waste glass canisters with known inventories and associated uncertainties are produced in a quality assured manner and stored in various countries, particularly in France and UK, transportability and interim storage is fully mature.

5.2 Basic understanding

Detailed mechanistic understanding of glass dissolution is available as a function of glass composition and environmental constraints: T, pH, S/V, water flow, presence of near field materials

The interaction of nuclear waste glasses in water has been studied for more than 40 years and the key influencing factors (see above) are known as a function of glass composition, presence of near-field materials, temperature and hydrological and geochemical constraints such as water residence time and pH. Leach tests and tests for transportability of full-scale waste glass canisters have been conducted.

5.3 Advancement of safety case

Nuclear waste glass development, storage, characterisation and safety cases for disposal are mature, with the available knowledge base directly applicable and transferable across RWM programmes.

5.4 Optimisation challenges and innovations

Considering the strong couplings observed between the glass/groundwater reaction, radionuclide release and near field properties, there remains challenges to assess the overall coupling of glass performance and near field evolution at the scale of a waste package (digital twin).

5.5 Past and ongoing (RD&D) projects

Past (RD&D) Projects financed by the EC include:

- Radionuclide release from solidified high-level waste, 1986-1989
- Evaluation of non-destructive methods for the quality checking of highly active vitrified waste, 1986-1989
- Characterisation of HLW glass, 1986-1990
- PReliminary Demonstration Test for Disposal of High Level Radioactive Waste in CLAY (CERBERUS), 1990-1996
- Retention of Pu, Am, Np and Tc in the Corrosion of COGEMA Glass R7T7 in Salt Solutions, 1991-1995
- Active Handling Experiment with Neutron Sources, 1991-1995
- GLASTAB: Long-term behaviour of glass: improving the glass term and substantiating the hypotheses, 2000-2003
- Understanding and Physical and Numerical Modelling of the Key Processes in the Near-Field and their Coupling for Different Host Rocks and Repository Strategies (NF-PRO), 2004-2007

6. Uncertainties

Exchange planned with the EURAD UMAN WP to see if they would be interested to draft a short section on each theme for generic uncertainties.

Uncertainties to be addressed concerning both interim storage and disposal:

- The history and evolution of the national nuclear energy use.
- The deduced heat loads, radiation fields and nuclide inventories.
- The availability of competent staff.

Uncertainties of relevance to interim storage:

- The expected duration of interim storage.
- The manipulation options if the repository concept is not yet selected (i.e., to what extent upstream packaging decisions and treatment options can be made if a DGR site and concept is not yet certain).

Uncertainties of relevance to geological disposal:

- The expected time of container rupture (impact on nuclide inventories and temperature of exposure of the glass with water from the near-field).
- Interaction scenarios with near-field materials, groundwater composition considering mass transfer resistance and corresponding models on multiple materials interaction couplings.

7. Key References

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