Global vitrification and disposal of high-level nuclear waste: Historical insights, current status, and future prospects

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Global vitrification and disposal of high-level nuclear waste: Historical insights, current status, and future prospects

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ABSTRACT

The global management of high-level radioactive waste (HLW) presents one of the most significant technical and policy challenges in the nuclear energy sector. This article provides a comprehensive review of the vitrification and geological disposal strategies employed by nations with active nuclear programs, including the United States, France, Russia, the United Kingdom, Japan, Germany, India, China, and Canada. The historical development of HLW glass waste forms, the evolution of vitrification technologies—such as Joule-heated ceramic melters (JHCM) and cold crucible induction melters (CCIM)—and the role of borosilicate and phosphate glasses are critically examined. While vitrification is a mature and widely accepted technology for HLW immobilization, the composition of glass waste forms and the specific vitrification processes vary across countries due to differences in waste chemistry, fuel cycle strategies, and regulatory frameworks. The article also discusses interim storage practices and progress toward establishing deep geological repositories, with an emphasis on technical, logistical, and socio-political challenges. As global interest in nuclear energy resurges to meet climate goals and energy demands, HLW volumes are expected to increase. In this context, this review highlights the need for innovation in waste form design, improved waste loading strategies, and international collaboration to ensure the safe and sustainable management of HLW in the decades ahead.

1. Introduction

The first spent nuclear fuel (SNF) was generated on December 2, 1942, during the first self-sustaining nuclear chain reaction at the Chicago Pile-1 (CP-1) reactor (Fig. 1), as part of the Manhattan Project. Subsequently, the world's first nuclear explosion, known as “Trinity,” involving a Pu-based bomb named “Gadget,” occurred at Alamogordo, New Mexico, in July 1945. This event along with the dropping of atomic

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bombs – Little Boy (U-based; Fig. 2**a**) and Fat Man (Pu-based; Fig. 2**b**) on Hiroshima (August 6, 1945) and Nagasaki (August 9, 1945) – ushered in a global nuclear arms race, resulting in the generation of millions of gallons of liquid high level waste (HLW) worldwide, also known as “legacy waste” or “defense waste” in the United States. Since the 1990s, most countries have halted the production of nuclear weapons, but many continue to retain or expand their nuclear power generating capacity and operate nuclear reactors for research and medical applications.

According to the International Atomic Energy Agency (IAEA), high-level waste (HLW), by definition, contains a significant concentration of long-lived and heat generating radionuclides [1]. Radioactive waste from hospitals and other medical institutions is typically not categorized as HLW. In contrast, the SNF from civilian power reactors can either be disposed of directly, as part of an open nuclear fuel cycle (NFC), or reprocessed to recover the fissile elements uranium (U) and plutonium (Pu), followed by immobilization/encapsulation of the remaining HLW into a suitable waste form—a process known as the closed NFC. Currently, four countries—France, Japan, Russia, and India—operate a closed NFC, while the United States, Canada, Sweden, and most other nations have adopted an open NFC. Germany and the United Kingdom initially adopted a closed NFC. While Germany never operated a large-scale commercial reprocessing facility, the UK operated two commercial reprocessing facilities at Sellafield – (1) Magnox plant (1964–2022) for the reprocessing of U based fuel from 1st generation Magnox reactors, and (2) Thermal Oxide Reprocessing Plant (THORP, 1994–2018) for the reprocessing of low-enriched uranium oxide (UO_x) fuels from advanced gas cooled reactors (AGRs) and light water reactors (LWRs). The reprocessed waste is still being vitrified in the UK. In the United States, the only commercial SNF reprocessing plant was operated by Nuclear Fuel Services Inc. at West Valley, New York, from 1966 to 1972. Meanwhile, a report by the Chinese Academy of Engineering indicates that China is expected to transition from an open to a closed NFC in the near future [2].

Regardless of the strategy pursued, all fuel cycle pathways ultimately generate HLW that requires safe management. Whether legacy HLW or civilian HLW produced by a closed NFC operation, this waste must be immobilized in a thermally stable and chemically durable, heat- and radiation-resistant solid matrix known as a “waste form,” prior to geological disposal. This ensures containment, prevents environmental release, and protects human health and ecosystems. Vitrification—using borosilicate glass in most countries or phosphate glass in Russia—remains the global benchmark and most widely adopted strategy for HLW immobilization [3,4]. Glass waste forms offer several advantages, including high waste-loading capacity, ability to immobilize a wide range of elements, relatively simple continuous production technology, high chemical durability, mechanical integrity, and excellent thermal and radiation stability [5,6]. Nonetheless, while consensus on the general approach exists across the technical and scientific communities, the compositions of the final waste forms and the production methods vary by country and depend on HLW chemistry and inventory. Table 1 presents representative compositions of HLW glasses from different countries.

Since vitrification of nuclear waste is a mature technology, its scientific and technical aspects have been extensively reviewed [3,7–17]. Therefore, this article does not attempt a comprehensive technical treatment, but briefly highlights the key materials-science factors that connect waste loading, vitrification technology, and repository performance. However, as the world enters an era of rapidly increasing energy demand—driven by population growth, advances in artificial intelligence technology, and the transition from fossil fuels to clean energy, nuclear energy has seen a significant resurgence in recent years. Governments, policymakers, and industry now anticipate a significant expansion in nuclear reactor deployment over the coming decade. This will inevitably increase HLW volumes, encompassing both well-characterized and emerging chemistries, all requiring safe immobilization and disposal. In this context, it is timely to reflect on past efforts in the design and development of glassy waste forms, assess the current state of HLW vitrification, and envision the future role of vitrified waste as a matrix for HLW from advanced reactors. This article provides a concise overview of the historical background, recent developments, and future prospects for HLW vitrification in countries with active nuclear programs across North America, Asia, and Europe. For a comparable discussion on the vitrification of intermediate- and low-level radioactive wastes, readers are referred to our other review article [18].

2. Scientific considerations governing HLW glass formulations

High-level waste streams (liquid, sludge or solid form) contain a

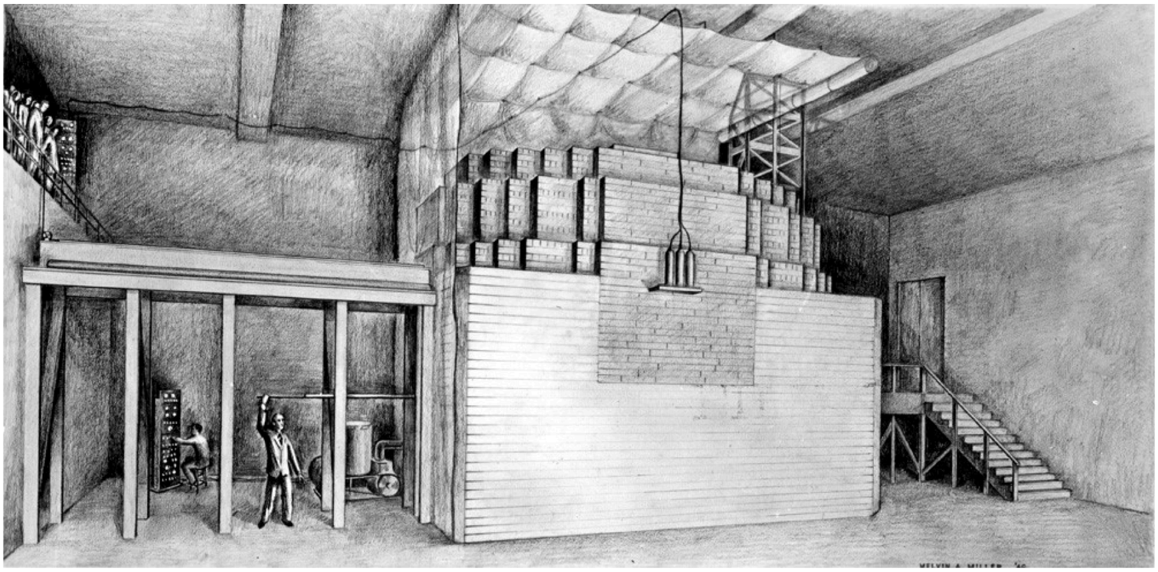

Fig. 1. Drawing (photograph copy) of the nuclear reactor erected beneath the west stands of the University of Chicago’s Stagg Field. With this, a group of scientists achieved the first self-sustaining chain reaction, a controlled release of nuclear energy on December 2, 1942. The reactor consisted of uranium and uranium oxide lumps spaced in a cubic lattice embedded in graphite. In 1943, it was dismantled and reassembled at the Argonne National Laboratory. **Credit:** University of Chicago Photographic Archive, [digital item number: apf2–00,503], Hanna Holborn Gray Special Collections Research Centre, University of Chicago Library.

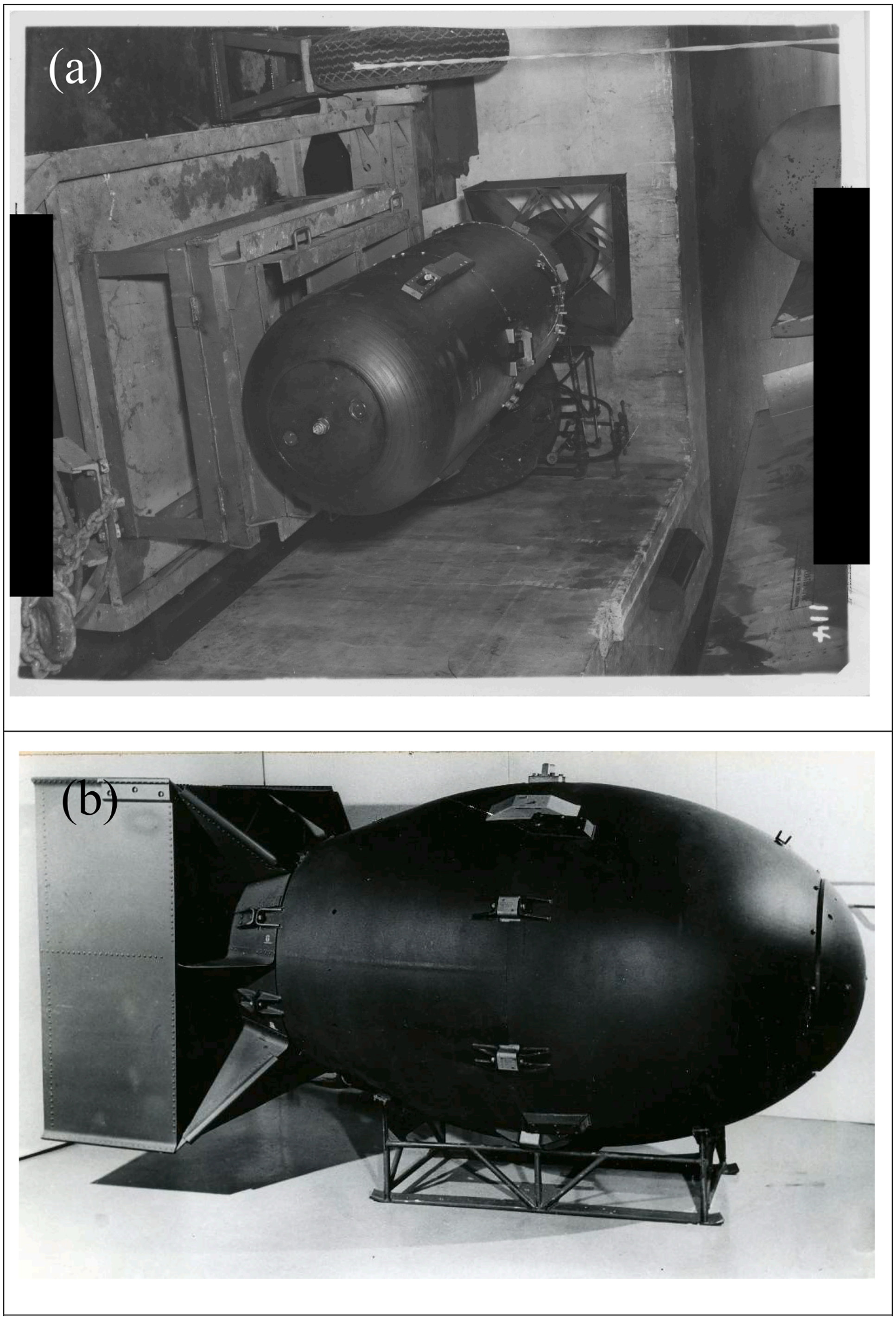

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Fig. 2. Photographs of (a) Little Boy unit (Dimensions: 28 inches in diameter and 120 inches long; weight: ~9700 lbs) on trailer cradle in pit, 1945. **Credit:** National Archives Identifier: 76,048,653; (b) nuclear weapon of the "Fat Man" type (Dimensions: 60 inches in diameter and 128 inches long; weight: ~10,000 lbs), **Credit:** National Archives Identifier: 175,539,928.

broad range of fission products, activation products, corrosion products, actinides, alkali and alkaline-earth elements, transition metals, noble metals, sulfates, phosphates, molybdates, and other components with very different solubilities in glass-forming melts. As a result, the design of HLW glass compositions and the maximum achievable waste loading are governed by a compromise between the processing ability of the glass melt (viscosity, electrical and thermal conductivity, liquidus temperature, volatility of radionuclides), operating melter parameters (temperature, foaming, stirring, refining time), and the performance of the vitreous waste form (chemical durability and radiation resistance) [19,20].

Among these constraints, crystallization and phase separation are particularly important because they may either limit waste loading [20–22,25] or, when properly controlled, offer a route for immobilizing difficult waste components [26,27]. Uncontrolled crystallization or phase separation may generate spinel-type phases [28–30], nepheline in high-Al and high-Na borosilicate glasses [21–24], noble-metal-rich deposits [31–36], or molybdate- and sulfate-rich salt phases commonly referred to as "yellow phase" [20,37–39]. These phases can affect melt homogeneity, melter operation, and long-term glass performance.

Table 1

Representative global HLW glass compositions (wt. %). [Note: Some glass compositions, e.g., MW and CaZn, do not include waste components in their chemistry].

Nation	HLW type	Glass	SiO_2	B_2O_3	P_2O_5	Al_2O_3	PbO	Fe_2O_3	Li_2O	Na_2O	K_2O	MgO	CaO	ZnO	BaO	Bi_2O_3	MnO	SrO	ThO_2	UO_3	ZrO_2	TiO_2	Others	Ref.
UK	Civil	MW	61.2	22.4	-	-	-	-	5.4	11.0	-	-	-	-	-	-	-	-	-	-	-	-	-	[39]
UK	Civil	CaZn	47.6	23.4	-	4.2	-	-	4.2	8.6	-	-	6.0	6.0	-	-	-	-	-	-	-	-	-	[39]
China	Defense	VPC	44.89	12.26	-	5.0	-	2.92	2.18	11.06	-	4.37	6.72		3.50	-	-	-	-	2.22	-	-	4.88	[46, 47]
Russia	Defense	Mayak	-	-	52.0	19.0	-	-	-	21.2	-	-	-	-	-	-	-	-	-	-	-	-	7.8	[5, 48]
USA	Defense	WTP Baseline	31.51–53.0	4.0–20.0	0–2.5	2.02–18.89	-	4.0–17.4	0.93–6.0	4.18–21.4	0–2.55	-	0–3.07	-	-	0–3.2	0–5.73	0–3.68	0–2.84	0–6.3	0–13.5	-	-	[49]
USA	Defense	WTP EWG	22.0–53.0	4.0–22.0	0–0.74	2.38–32.0	-	0.88–20.0	0–6.0	4.1–23.71	0–2.7	0–4.84	0–10.0	0–4.0	-	0–6.15	0–10.0	0–3.65	0–4.16	0–8.45	0–13.5	-	4.58	[50]
USA	Defense	WVDP	39.5–48.4	11.2–14.8	1.0–1.4	5.6–7.1	-	10.7–13.5	3.3–4.2	7.1–8.6	4.1–5.3	0.7–1.3	0.21–0.6	-	0.16	-	0.7–0.9	-	0.1–3.6	0.1–0.8	1.2–1.4	-	1.87	[51]
USA	Defense	DWPF	44.8–54.6	4.3–8.3	0.2–0.6	4.3–9.8	-	8.2–12.6	3.5–5.6	11.3–14.4	0.0–0.1	0.2–2.2	0.4–1.4	0–0.6	0.0–0.1	-	1.1–3.3	0.0–0.3	0.5–0.7	1.0–3.5	0.1–0.2	0.0–0.7	0.2–3.9	[51]
India	Civil	IR-111	60.68	11.39	-	-	-	-	-	0.36	-	-	-	-	-	-	16.55	-	-	-	-	11.03	-	[52]
India	Research	SB-44	38.61	25.32	-	-	-	-	-	12.02	-	-	-	-	24.05	-	-	-	-	-	-	-	-	[53]
India	Research	WTR-62	40.0	13.33	-	-	33.33	13.33	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	[54]
Germany	Civil	SM513FR	58.6	14.7	-	3.0	-	-	4.7	6.5	-	2.3	5.1	-	-	-	-	-	-	-	-	5.1	-	[14]
Germany	Civil	SM527FR	50.0	28.0	-	-	-	-	4.0	11.0	-	-	5.0	-	-	-	-	-	-	-	-	2.0	-	[14]
Germany	Civil	GGWAK1	60.0	17.6	-	3.1	-	-	3.5	7.1	-	2.2	5.3	-	-	-	-	-	-	-	-	1.2	-	[14]
Germany	Civil	SON68FR	54.9	16.9	-	5.9	-	-	2.4	11.9	-	-	4.9	3.0	-	-	-	-	-	-	-	-	-	[14]
France	Civil	R7T7	42.50–51.68	12.40–16.50	0.5	3.60–6.60			2.0	8.10–11.00			4.0	2.5							2.0–4.9			[55]
France	Civil	AVM	38.5–46	16–19.5	0–1.7	9–12.5	-	2.8 (including $NiO+Cr_2O_3$)	0.4	5–18.8	-	2.5–7.5	0.2	-	-	-	-		-	-	0.1–1.0	-	25.5–28.9	[55]
France	Civil	UMo	38.7	13.9	3.1	7.1				9.4			6.1	6.0							3.3		12.4	[13]

Conversely, controlled crystallization can be deliberately exploited in glass-ceramic waste forms, where poorly soluble elements are incorporated into durable crystalline phases dispersed within a residual glassy matrix [13,20,40].

Long-term performance of vitrified HLW also depends on glass alteration after water access in a geological repository. Glass-water reaction generally involves hydration, ion exchange, hydrolysis of the glass network, formation of an altered gel layer, and possible precipitation of secondary phases [11,41–43]. These processes are influenced by temperature, pH, groundwater chemistry, silica activity, radiation field, flow conditions, and interactions with surrounding materials such as canisters, corrosion products, buffer materials, and host rocks [44]. Geological repository is the internationally accepted endpoint for long-lived radioactive wastes, including SNF declared as waste, vitrified high-level waste, and long-lived intermediate-level waste, although national programs differ in inventory, host-rock selection, repository design, and implementation schedule [45]. Therefore, HLW glasses reported across different countries should not be interpreted only as operational choices, but as the outcome of waste chemistry, glass formulation, thermal history, melter design, repository concept, and performance requirements. Also, countries processing compositionally different HLW feeds may adopt different glass families, such as alkali borosilicate, aluminoborosilicate, phosphate, or glass-ceramics.

3. Glass as a high-level waste form – A historical perspective

The idea of immobilizing radioactive waste in vitreous (glass) or crystalline materials has been around for over 50 years. After World War II, fission product solutions began accumulating in countries involved in nuclear fuel reprocessing for defense purposes, such as the USA, USSR, UK, Germany, and France. For instance, the official website of the US Government Accountability Office (https://www.gao.gov/nuclear-waste-disposal) reports that the United States defense program produced about 90 million gallons (340,687 m^3) of liquid HLW with a radioactive inventory $> 1 \times 10^9$ Curies. These highly radioactive, complex solutions, containing around 40 different elements, required constant stirring and cooling to manage their thermal energy. Storing them in liquid form was not feasible, so these nations quickly turned to exploring solidification methods. Research on the solidification of fission product solutions started in the 1950s in the USA, France, UK, and Canada. The development of glass materials for nuclear waste immobilization progressed internationally at different times, with varying glass formulations and strategies being explored, as presented in Fig. 3. To illustrate these developments, the following sections highlight country-specific milestones, beginning with early efforts in the United States.

3.1. United States

An initial concept of immobilizing radioactive elements in an assemblage of mineral phases was proposed in 1953 by Hatch from Brookhaven National Laboratory [57]. Shortly thereafter, between 1956 and 1957, Goldman et al. [58] at the Massachusetts Institute of Technology (MIT) developed the first borosilicate glass formulation for nuclear waste immobilization. The MIT researchers examined calcium-aluminosilicate porcelain glazes modified with boron oxide (B_2O_3) to reduce its viscosity and minimize radionuclide volatilization. The inclusion of oxides such as ZrO_2 and Fe_2O_3, commonly found in nuclear waste, did not negatively affect the glass quality [58]. As a result, several compositions in $CaO–Al_2O_3–B_2O_3–SiO_2$ and $Na_2O–CaO–Al_2O_3–B_2O_3–SiO_2$ systems were proposed for further development in immobilizing fission products in borosilicate nuclear waste glasses. Building on the progress achieved, the Pacific Northwest Laboratory (PNL) Waste Solidification Engineering Prototypes (WESP) program (1966–1970) demonstrated the use of pot calcine, phosphate microcrystalline ceramic, phosphate glass, and borosilicate glass for the

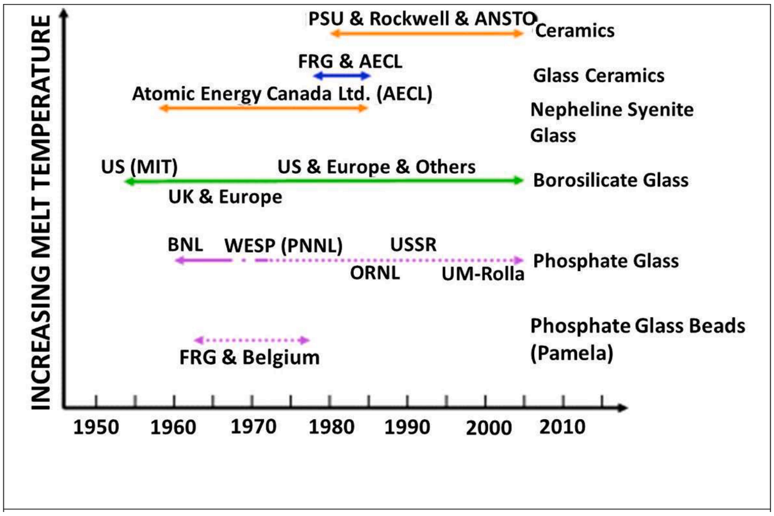

Fig. 3. Historical development of glass and ceramic waste forms in the United States and Europe. *Reprinted from* Bao et al. [56] *with permission from The American Ceramic Society and Wiley Periodicals, Inc.*

immobilization of HLW [59,60].

During the 1970s and early 1980s, the Office of Nuclear Waste Isolation (ONWI) emphasized solid-state stability as the primary requirement for waste forms. This promoted extensive research into various ceramic-based crystalline waste forms, including super-calcine ceramics, alumina-based tailored ceramics, and titania-based SYNthetic ROCk (SYNROC). At the same time, “low leachability” became a key criterion for waste form comparison, sparking a debate between the proponents of crystalline ceramics and glass [6]. Between 1979 and 1981, the US Department of Energy (DOE) conducted the National High-Level Waste Technology Program to evaluate potential waste forms for immobilizing HLW at the DOE defense sites. Out of seventeen candidate waste forms evaluated (Table 2), the following seven were selected for further development: (1) Borosilicate glass; (2) SYNROC; (3) Tailored ceramic; (4) High-silica glass; (5) FUETAP (Formed Under Elevated Temperature and Pressure) Concrete; (6) Coated particles; (7) Glass marbles in a lead matrix [61–63]. Following a competitive selection process—known as the "Atlanta shoot-out,"—involving DOE national laboratories and independent laboratories, peer reviews, product performance assessments, and processability evaluations, borosilicate glass was recommended as the reference material for immobilizing HLW, while SYNROC ceramics were identified as promising alternatives [59].

In 1980, the West Valley Demonstration Project (WVDP) authorized DOE to solidify ~600,000 gallons (~2271 m^3) of liquid HLW stored at the West Valley facility in New York. DOE selected borosilicate glass for this purpose and vitrified the HLW inventory into borosilicate glass between 1996 and 2002. In parallel, the DOE’s Defense Waste Processing Facility (DWPF) at the Savannah River Site (SRS) located in South Carolina began processing legacy HLW in 1996 and has produced more than 16 million pounds of borosilicate glass so far.

By contrast, the Hanford Site in central Washington hosts the largest volume of legacy HLW in the United States, about 5.6 million gallons (~21,200 m^3; ~10 vol.% of the total tank waste inventory) [3]. This waste is slated for vitrification in borosilicate glass at the Waste Treatment and Immobilization Plant (WTP), which remains under design and construction [64]. Collectively referred to as “tank waste,” the HLWs at West Valley, SRS, and Hanford primarily consist of liquid raffinates from nuclear material processing that were stored in underground tanks.

All US HLW vitrification facilities employ (or have employed) Joule Heated Ceramic Melters (JHCMs), a technology pioneered by PNL (now PNNL) in the 1970s [65,66]. In this process, liquid HLW waste is combined with glass formers (or a glass frit) and fed directly to the JHCM where it is converted to molten glass. HLW glass compositions are generally characterized by significant (mass percentages) concentrations of SiO_2, Na_2O, Fe_2O_3, Al_2O_3, B_2O_3, and Li_2O. The molten glass is then poured into stainless steel canisters, which are then sealed, and stored in an engineered facility until final disposition in a geological repository. A schematic of the melter/pour cell used at West Valley is shown in Fig. 4.

With respect to the disposal of vitrified HLW, the steel canisters containing HLW glass are intended for disposal in a geological repository—most notably, the proposed Yucca Mountain nuclear waste repository in Nevada [67,68]. For the history of the US site selection and development of Yucca Mountain, see [68,69]. The Yucca Mountain disposal concept is distinctive compared to those of other countries, as the repository is designed to be located above the groundwater table and within an oxidizing environment [70]. Once forced ventilation is discontinued, temperatures within the drifts and surrounding rock are expected to exceed the boiling point of water, with the borosilicate glass reaching temperatures of up to 300–400 °C [72,73]. Although these temperatures remain below the glass transition temperature of HLW glass (~500 °C), they are still expected to affect the long-term performance of the vitrified waste form. Consequently, based on extensive research on the behavior and durability of vitrified waste under geological repository conditions [74–76], the U.S. DOE established stability criteria in the Waste Acceptance Product Specifications for Vitrified High-Level Waste Forms (WAPS) [77].

However, the Yucca Mountain project was defunded in 2010. In

Table 2
Candidate waste forms considered for immobilization of HLW for disposal [63].

Vitreous (Glass)	**Cementitious**
• Borosilicate glass • High silica glass • Phosphate glass	• Normal concrete • Hot-pressed concrete • FUETAP concrete
Ceramic (Sintered)	**Consolidated**
• SYNROC • Clay ceramic • Tailored ceramic	• Titanate ion exchanger • Stabilized calcine • Pelletized calcine • Matrix forms*
Composites	• Coated sol-gel particles • Disk-pelletized coated particles
• Glass ceramic • Cermet (ceramic-metallic)	

[*] Based on multibarrier concepts (e.g., glass marbles encapsulated in a lead matrix)

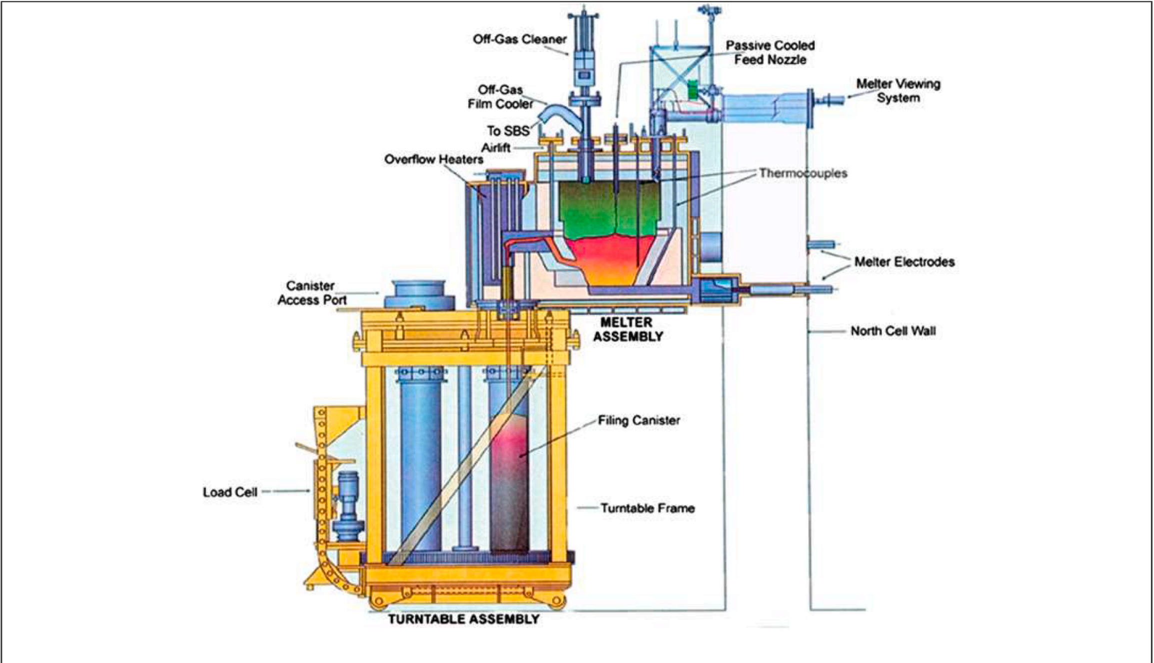

Fig. 4. Schematic of West Valley Demonstration Project Melter (SBS: Submerged Bed Scrubber) [71]. *Courtesy: US Department of Energy, West Valley, NY.*

March of that year, the DOE filed a motion to withdraw its license application for repository construction, stating that Yucca Mountain was "not a workable option" for long-term nuclear waste disposal. Although the motion was denied in June 2010, the licensing proceedings were formally suspended in 2011 [78,79]. Since then, the Yucca Mountain project has remained dormant, with no substantive progress toward its development as a functioning nuclear waste repository [80]. Nonetheless, in May 2016, the Nuclear Regulatory Council (NRC) released its final supplement to the US DOE's environmental impact statement on the proposed Yucca repository. Both the environmental impact assessment and the NRC's experts established that the repository design would prove safe for one million years [81].

The Idaho site chose a different route for managing the legacy HLW generated from reprocessing SNF from the U.S. Navy ships and submarines. The waste was converted into granular solid materials through calcination as it was generated, instead of being stored long-term in underground tanks [82]. These calcines have been safely stored in bins on site for decades. Vitrification of Idaho HLW remains a leading disposition pathway, benefitting from the proven success of borosilicate glass processing in the U.S. and reduced regulatory complexity compared to alternative strategies. The current status and future challenges pertaining to processing and vitrification of these wastes are discussed in Sections 4 and 5, respectively.

Beyond legacy waste, vitrification is recognized as a potential option for future generated HLW waste, including those generated using advanced commercial reactors. For example, phosphate-based glasses—once deemed unsuitable for legacy tank wastes due to high melt corrosivity and poor durability [60]—are receiving renewed attention for immobilizing dehalogenated salt wastes from Generation-IV sodium-cooled fast reactors (SFRs), where metallic fuel is reprocessed by electrochemical/pyrometallurgical methods, and from chloride-based molten salt reactors (MSRs) [83,84].

3.2. *France*

Although early borosilicate glass formulations were developed in the United States, most vitrification process development and testing during the early 1960s took place in Europe [16]. The French nuclear waste management program, initiated at Saclay in the 1950s under Roger Bonniaud, was focused on creating synthetic minerals like phlogopite mica [7]. However, glass emerged as a material capable of immobilizing all the elements in such complex nuclear waste solutions. In 1957, the first radioactive glass was synthesized at a laboratory scale at Saclay in France, followed by a larger production in 1964 at Fontenay-aux-Roses.

Glass pellets with chemistries similar to $14.7\ Na_2O - 0.8\ K_2O - 0.8\ CaO - 11.4\ Al_2O_3 - 11.5\ Fe_2O_3 - 60.8\ SiO_2$, wt.%, were synthesized. By the 1960s, radioactive glass was produced on a larger scale (10 kg) at Marcoule through sol-gel process in the Gulliver cell. By 1968, the PIVER (Pilote Industriel de Vitrification) facility at the Marcoule Pilot Plant was capable of vitrifying high activity solutions – fission products from natural uranium graphite gas reactors and the Phenix fast reactor — on a semi-industrial scale (200 kg). Between 1969 and 1973, a total of 25 m^3 (~6600 US gallons) of solution was vitrified, producing 12 tonnes of glass. Subsequently, the Marcoule vitrification facility (l'Atelier de Vitrification de Marcoule, AVM) – comprising a two-step process, i.e., calcination of fission products followed by melting (Fig. 5) – was commissioned in 1978 for the industrial-scale production of nuclear waste glass, followed by two more vitrification plants at La Hague in 1989 (R7) and 1992 (T7), respectively [7]. The Orano La Hague R7 and T7 facilities cumulate over 35 years of industrial operation, with a strong track record of safety, reliability, and product quality, in line with reprocessing plant operational objectives. From commissioning until the end of 2024, the R7 and T7 facilities have produced tens of thousands of canisters and successfully vitrified over 10 billion curies. Since the start of industrial operations, continuous efforts have been made to advance processes, integrate innovative technologies, and develop new glass formulations.

France initially used induction heated metallic melters (IHMM, also known as Hot Wall Induction Melter (HWIM); Fig. 6) for vitrification, but due to the corrosive nature and requirement of high melting temperature, cold crucible induction melters (CCIM) were preferred for the vitrification of waste streams difficult to process in conventional Joule-heated melters. These included, in particular, solutions rich in alkali elements and potentially containing sulfur, as well as reprocessing UMo solution enriched in molybdenum, which pose challenges related to corrosion, volatilization, and crystallization phenomena [86]. In 2010, Orano and the French Alternative Energies and Atomic Energy Commission (Commissariat à l'Énergie Atomique et aux Énergies

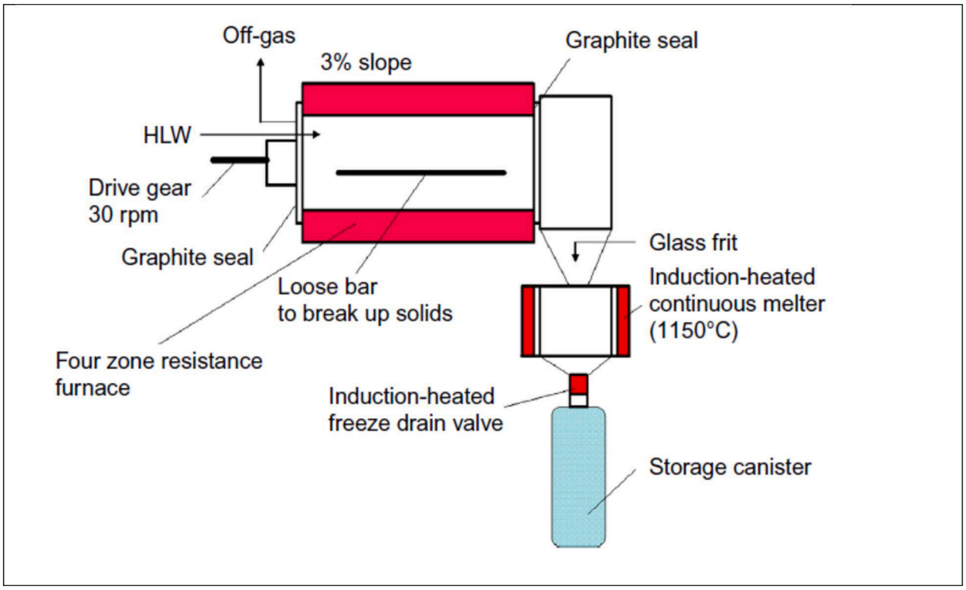

Fig. 5. A schematic of two-stage vitrification process developed at AVM. Reprinted from Ojovan and Lee [85] with permission from Elsevier.

Fig. 6. Photograph of La Hague hot-walled induction melter (HWIM). *Reprinted from* Vienna et al. [10] *with permission from The American Ceramic Society and Wiley Periodicals, Inc*.

Alternatives, CEA) successfully installed a CCIM (Fig. 7) on one of the six vitrification lines at the La Hague facility [7,87]. An alkali aluminoborosilicate glass was developed as the waste form. Upon slow cooling, this glass exhibited phase separation and partial crystallization, leading to the formation of powellite ($CaMoO_4$) crystals dispersed within a glassy matrix. The resulting glass-ceramic waste form enabled a waste loading approximately 4 × higher than that of conventional borosilicate glasses. The UMo Vitrification campaigns using the CCIM technology at La Hague were conducted between 2013 and 2020 [13].

UMo waste solutions are fission products from the reprocessing of U-Mo-Sn-Al SNF used in Gas Cooled Reactors (GCRs). The main constituents of UMo solution are MoO_3 (137 g/l), P_2O_5 (42 g/l), Na_2O (11 g/l), Others (15 g/l). The total volume of UMo solution was estimated to be ~66,000 US gallons (250 m^3) with total activity of $< 222 \times 10^{10}$ Bq/l.

3.3. United Kingdom

The development of waste forms in the United Kingdom began in the 1950s. In 1957, Amphlett [88] published experimental work exploring natural soils as a base material for glass formation. Since the starting materials, or batch, required heating at 1500 °C to produce bubble-free

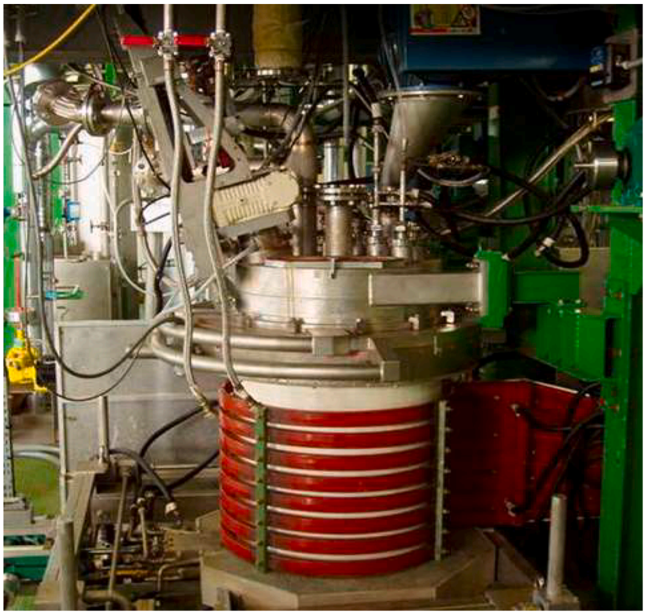

Fig. 7. 650 mm diameter CCIM at the CEA Industrial Scale Demonstration Platform in Marcoule, France. *Reprinted from* Vienna et al. [10] *with permission from The American Ceramic Society and Wiley Periodicals, Inc.*

glass—considered high for HLW processing due to concerns over volatility of radionuclides and melter —borax was added as a fluxing agent to reduce the melting temperatures to 1000 °C. By 1958, research focused on identifying a suitable glass composition for incorporating wastes rich in aluminum, uranium, and iron. During the 1960s, the FINGAL (Fixation in Glass of Active Liquors) process (Fig. 8) was demonstrated for the vitrification of highly active liquors and subsequently developed to the HARVEST (High Activity Radioactive Waste Vitrification System) batch vitrification process (as a scaled-up version of FINGAL) in 1972 [9]. In alignment with the US and French programs, a variety of glass compositions were selected in the $CaO–B_2O_3–SiO_2$ and $Na_2O–B_2O_3–SiO_2$ ternary systems until the 1980s, when waste oxide contents reached up to 30 wt.% and the glasses were melted at 1100°C [4,89,90]. Several other compositions, including aluminosilicates (as in Canada, discussed below) and various phosphate glasses, were also investigated. Ultimately, borosilicate glass compositions were chosen as the reference waste forms. In the 1980s, the UK selected the "MW" glass—short for "Mixed Waste," "Magnox Waste," or "Mixture Windscale" (different terms used by different researchers)—as the base composition for vitrifying HLW at the Sellafield site. This glass was formulated within the $Li_2O – Na_2O – B_2O_3 – SiO_2$ system, featuring an equimolar ratio of Li_2O and Na_2O [89].

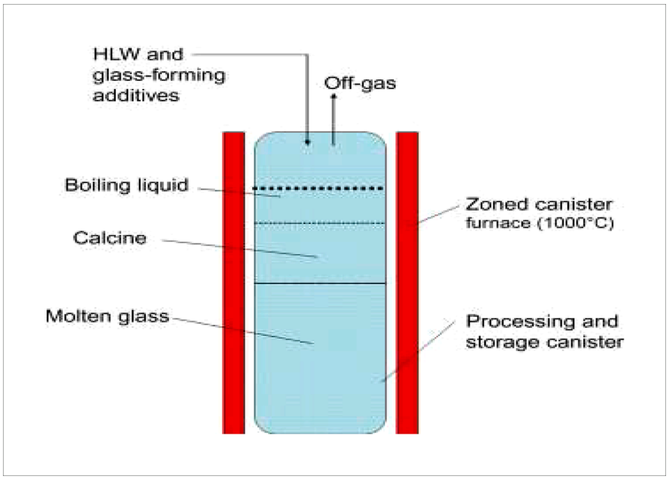

Fig. 8. A schematic of FINGAL process for the vitrification of highly active liquors. Reprinted from Ojovan and Lee [85] *with permission from Elsevier.*

In 1980, British Nuclear Fuels Limited (BNFL) proposed acquiring the French AVM process to vitrify the waste generated from its commercial reprocessing activities. In 1982, the UK decided to adopt the second-generation French AVH (Atelier de Vitrification La Hague) process developed by Orano (formly known as Cogema) for its Waste Vitrification Plant (WVP). Fig. 9 shows the flow sheet diagram of vitrification technology currently used to immobilize HLW in the UK. During the late 1980s, WVP lines 1 and 2 were commissioned at the Sellafield site, beginning operations in 1990. A third WVP line followed, commissioned in the late 1990s and becoming operational by 2002.

Since the beginning of its commercial operations in 1990, the WVP has successfully vitrified pure Magnox Waste and blends of 75/25 and 50/50 THORP/Magnox wastes using MW glass

[9,39]. Over time, the oxide waste loading has been increased to 35 wt.% [9]. In 2004, the Vitrification Test Rig (VTR) was integrated into the facility to improve throughput. The typical operating temperature of WVP and VTR melters is approximately 1050°C [9,39,92]. These facilities continue functioning at present.

3.4. Canada

According to the official website of the Canadian Nuclear Safety Commission (https://www.cnsc-ccsn.gc.ca), three tanks of high-level liquid waste from reprocessing activities conducted between 1949 and 1956 are stored at the Chalk River Laboratory (CRL). The last transfer of radioactive liquid solutions to these tanks occurred in 1968, and no additions have been made since then.

The Canadian program to immobilize radioactive waste initiated in 1956 at the Chalk River Nuclear Laboratories (CRNL), and the first highly active glass waste form was produced in May 1958 [93]. Nepheline syenite (a silica deficient igneous rock) was proposed as a base material for the production of a glass-based nuclear waste form. A mixture of 65 – 85 wt.% nepheline syenite, consisting of nepheline ($NaAlSiO_4$) and two feldspars, albite ($NaAlSi_3O_8$) and microcline ($KAlSi_3O_8$), was mixed with 15 – 35 wt.% lime (CaO) as a flux and the fission products added in an acidic solution. The mixture was melted at temperatures ranging from 1250 – 1350 °C yielding a glassy waste form [94]. A pilot plant was constructed at Chalk River to demonstrate the feasibility of the vitrification process on a kilocurie level. In 1958, the program produced 50 glass blocks, each weighing about 2 kg, which remain safely stored onsite. This was the first time that a process was developed to incorporate radioactive waste into a glass [93,95]. Glass waste form development was terminated in 1960 when no fuel reprocessing was anticipated in Canada. However, a new HLW treatment program was re-initiated in 1976, focusing on interim storage of spent fuel and its immobilization for ultimate disposal. The plan was to adopt existing solidification techniques while emphasizing research and development into waste form development and characterization, including alternatives to borosilicate glass. Between 1976 and 1981, a dilute borosilicate glass containing ≤3 wt.% fission products was examined as the reference waste form, designed to limit decay heat within the canister [60]. Canada ultimately became the ninth country in the world to adopt borosilicate glass as the reference nuclear waste form [16].

3.5. Germany

The use of nuclear power in Germany began with the enactment of the first Atomic Energy Act of 1959. According to the Federal Ministry of the Environment, Nature Conservation, Nuclear Safety and Consumer Protection of the Federal Republic of Germany (https://www.bundesumweltministerium.de), a total of 7132,645 US gallons (27,000 m^3) of

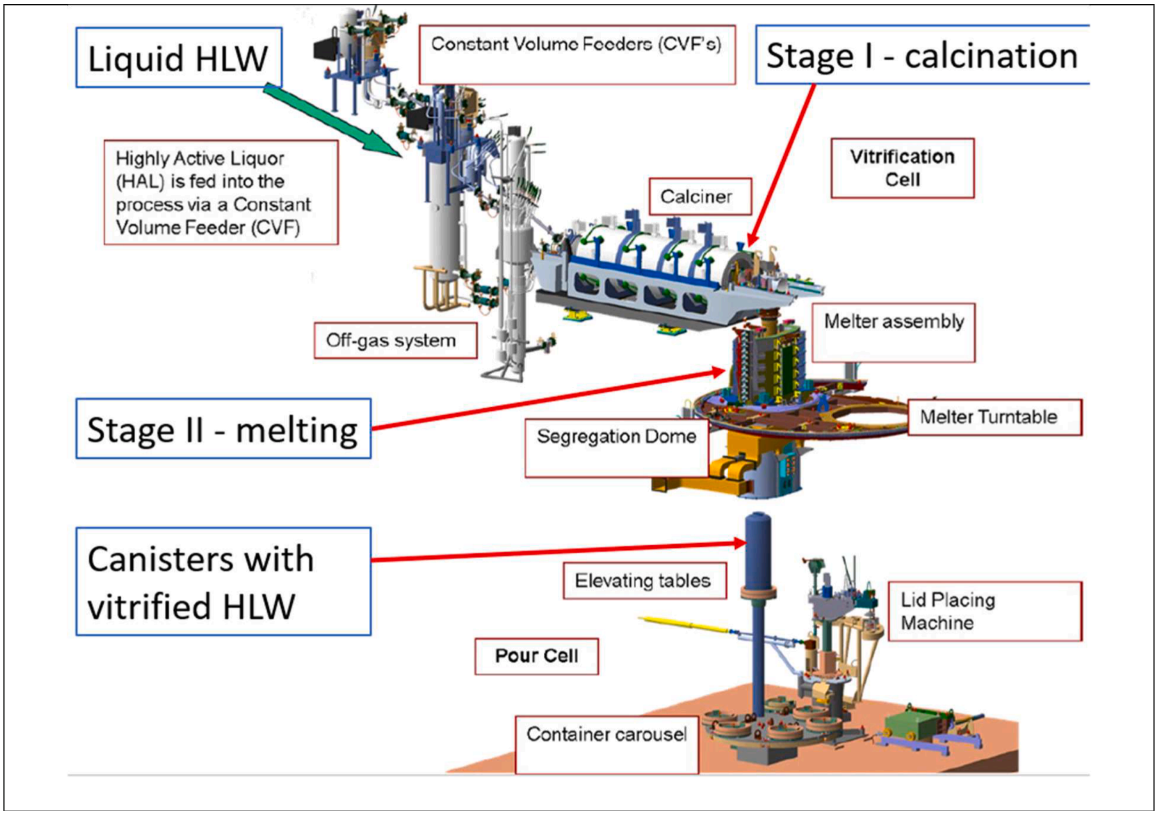

Fig. 9. Schematic of the two-stage vitrification process used in the UK at the waste vitrification plant (WVP). *Reprinted from Ojovan* [91] *with permission from Elsevier.*

HLW (also referred to in Germany as Highly Active Waste Concentrate – HAWC, or high level liquid waste – HLLW) has been generated by German nuclear reactors since 1960 [96]. The experimental work on nuclear waste immobilization in Germany began in the mid-1960s and early 1970s. The following four processes were considered for solidifying HLW [97,98]:

- **VERA** (Verglasungsanlage für radioactive Abfälle): A two-stage process (calciner + metallic crucible) was developed at the Karlsruhe Nuclear Research Center (KfK, now merged into Karlsruhe Institute of Technology (KIT)) for producing borosilicate glass.
- **FIPS** (Fission Product Solidification): Another two-stage process developed at the Jülich Nuclear Research Center (KfA), in which the waste solution and glass frit were dried on a drum drier, scraped off, and subsequently fed into the melter container to produce borosilicate glass [99].
- **PHOTHO:** A phosphate glass process developed by Gelsenberg at KfA for solidifying thorium-bearing reprocessing waste [60].
- **PAMELA:** A joint German-Belgian liquid fed JHCM facility in Mol, Belgium, developed for vitrifying reprocessed waste from the Eurochemic Reprocessing Plant [100].

Initially, PAMELA was planned as a continuous waste vitrification melter to produce “Vitromets,” i.e., phosphate glass beads embedded in a metallic matrix. However, in 1977 all German HLW solidification activities were consolidated to develop a demonstration plant for the PAMELA process. The PAMELA plant was constructed during 1981 – 1984 in Mol, Belgium, at the Eurochemic site in order to convert 12,416 gallons (47 m^3) of liquid waste into borosilicate glass. It was the first facility to apply slurry-fed JHCM technology to vitrify HLW (Fig. 9) operating during 1985 – 1991 [100]. By September 1991, approximately 237,755 gallons (900 m^3) of HLW from the reprocessing of SNF was successfully converted into 500 metric tonnes of borosilicate glass, stored in 2201 canisters. After 6 years of operation, one ceramic melter and three other large components were dismantled, decommissioned, and immobilized in cement mortar [101]. The liquid-fed JHCM was later used at Karlsruhe vitrification plant (Verglasungseinrichtung Karlsruhe,

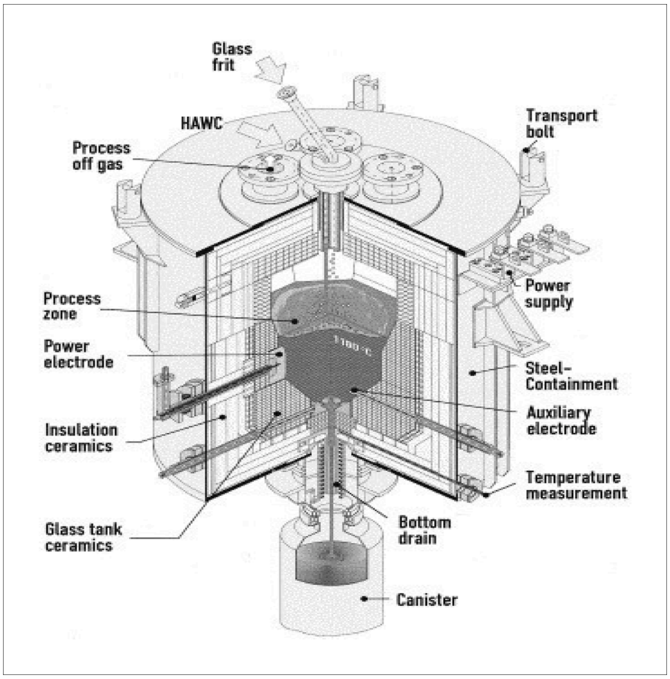

Fig. 10. Perspective view of the liquid fed JHCM designed to vitrify ~70 m^3 (18,492 US gallons) of high-level waste concentrate (HLWC) for Karlsruhe vitrification plant (VEK). A similar JHCM with dimensions 2 × 2 × 2 m^3 was built at PAMELA vitrification facility. Reprinted from Roth and Weisenburger [14] *with permission from Elsevier.*

VEK) (Fig. 10) during 2009 – 2010 to vitrify HAWC from WAK (Wiederaufarbeitungsanlage Karlsruhe Betriebsgesellschaft) pilot-scale reprocessing plant [98].

Beyond these pilot-scale operations, Germany relied heavily on other European countries for reprocessing and vitrification of nuclear waste into borosilicate glass. Between 1970 and 1994, Germany transported over 4550 tonnes of SNF (heavy metal equivalent) in 1904 flasks across 1562 shipments for reprocessing and vitrification into borosilicate glass, as noted by Fasten et al. [102]. Most of the SNF was transported to the reprocessing plants in France and the UK, while some SNF was also shipped to Belgium (9 shipments in 1970), Denmark (1 shipment in 1983) and Sweden (12 shipments during 1977 – 1988).

Following the 2002 amendment of the German Atomic Energy Act, reprocessing of German SNF was prohibited starting in 2005, leaving dry storage as the sole treatment or "temporary" management option [103]. According to the official website (https://www.base.bund.de) of the Federal Office for the Safety of Nuclear Waste Management of Germany, (*Bundesamt für die Sicherheit der nukleraren Entsorgung*, BASE), Germany had shipped a total of 6670 tonnes of SNF for reprocessing and vitrification by 2005—851 tonnes to Sellafield (UK) and 5379 tonnes to La Hague (France) [104]. The resulting vitrified waste was stored at facilities in France and the UK, but under contractual obligations, German nuclear plant operators—as the legal waste owners—must eventually reclaim the material for further management [104].

3.6. Japan

The first commercial nuclear reactor in Japan was commissioned in 1966 [105]. Until 2011, Japan was operating a fleet of 54 light water reactors (LWRs) generating ~30% of the country's electricity. Due to limited natural resources, Japan adopted a closed NFC, commissioning its first pilot-scale reprocessing plant—the Tokai Reprocessing Plant (TRP)—in 1977. During 1977 – 2009, TRP reprocessed 1140 tonnes of SNF using the Plutonium Uranium Redox Extraction (PUREX) process. The resulting HLW was stored as liquid, but it was clear that this waste would later be converted into a chemically durable solid.

In 1980, the Atomic Energy Commission of Japan (AEC, 原子力委員会, Genshiryokuiinkai) introduced a policy, *Promotion of R&D on conditioning and disposal of high-level wastes*, which recommended vitrifying HLW into borosilicate glass using JHCMs. Consequently, two technical-scale solidification facilities were built: one with a current-heated ceramic melter and the other with a direct mega-Hertz induction-heated melter. Both were designed to operate at temperatures between 1200 – 1350 °C, a range higher than the 1100 – 1150 °C used in most other countries. The goal was to produce glass compositions with greater chemical durability. However, operating at 1350 °C led to high corrosion rates for the electro-fused refractories. In parallel, a significant amount of SNF was reprocessed and vitrified abroad. Between 1980 and 2001, more than 7000 tonnes of commercial SNF were shipped overseas— ~2940 tonnes to Cogema (now Orano) in France and ~4100 tonnes to Sellafield in the UK [106].

In 1995, a ceramic melter vitrification facility, known as Tokai Vitrification Facility (TVF), was installed at TRP to vitrify the high-level liquid waste (HLLW) into borosilicate glass using the JHCM technology [107]. However, TVF operations encountered problems, such as an explosion at the TRP's asphalt solidification facility and accumulation of platinum group elements (PGEs) damaging one of the main electrodes of the TVF melter in 1997 [37,108]. Upon resuming operation, the JHCM at TVF operated between 1100 – 1250 °C with a melting rate of 9 kg/h. By 2007, TVF had produced 247 HLW glass canisters (diameter: 430 mm, height: 1040 mm, thickness: 6 mm), but operations were suspended again in 2011 due to seismic re-evaluation and the Fukushima Daiichi nuclear accident. The facility restarted in 2016 aiming to vitrify > 400 m^3 (105,669 US gallons) of remaining HLW. As of July 2025, ~370 m^3 of HLW still awaits vitrification.

For commercial scale reprocessing, Rokkasho Reprocessing Plant (RRP) is the first SNF reprocessing plant proposed by Japan Nuclear Fuel Limited (JNFL, 日本原燃 Nihon Gennen). The construction of RRP began in 1993 and is expected to be completed by March 2027. RRP was planned to reprocess Japan's light water reactor (LWR) SNF with two lines each — glass production lines (A– and B– series) — in its vitrification facility (K-facility) [108]. The K-facility adopted a similar glass melter to that of TVF, but the glass material which was fed together with HLLW was changed from a fiber cartridge to ~2 mm beaded frit. The temperature of the melter bottom was intentionally kept lower (~850 °C) than the upper part to avoid settling of PGEs. The temperature was raised only during glass pouring. After 13 years of construction, RRP was brought to active trial operation in March 2006, and for K-facility in November 2007. During this operation, several problems were observed e.g., PGE deposition at the melter bottom (Fig. 11), clogging in the outlet nozzle caused by low-viscosity Mo, bending of the stirring rod in the melter, leakage of HLLW, and a roof brick dropping into the melt pool. After the trial operation, new countermeasures such as a numerical code for heat power control, dilution methods of feed HLLW and glass melt in the melter to decrease the concentrations of problematic elements, were developed. Stable melter operation with these countermeasures was then verified by full-scale mock-up melter tests in 2009 – 2011 and at the K-facility with actual HLLW in 2012.

During these years, France and the UK vitrified and returned to Japan the majority of HLW generated from the reprocessing of Japan's SNF. France vitrified and return to Japan a total of 1310 glass cansiters during 1995 – 2007 while UK vitrified and returned to Japan a total of 520 canisters of HLW glass till 2016. France shipped the first batch of 28 HLW glass canisters to Japan in 1995 and the last batch of 130 canisters in 2007. Similarly, the UK shipped the first consignment of 28 canisters of HLW borosilicate glass to Japan in 2010 and the last consignment of 132 canisters in 2016.

3.7. Russia

Research on nuclear materials in the Soviet Union began in the 1920s with the launch of the "Uranium Project." By October 1940, eight months before Germany invaded the Union of Soviet Socialist Republics (USSR) during World War II, the USSR Academy of Sciences approved a comprehensive plan, focusing to develop methods for uranium isotope separation, constructing necessary equipment, inspecting uranium deposits, studying uranium ore processing and treatment methods, and investigating the potential for chain reactions in naturally enriched uranium [109].

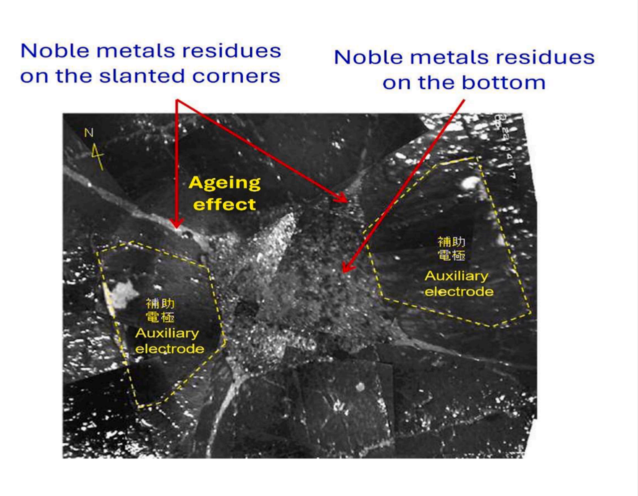

Fig. 11. An image showing the deposition of platinum group elements (PGE, noble metals) in the liquid fed JHCM (image not to scale). *Courtesy:* Japanese Nuclear Fuel Ltd.

The German invasion of the Soviet Union accelerated the shift of expertise, infrastructure, and knowledge from the Uranium Project toward military applications of uranium. By the early 1940s, the drive to develop a Soviet nuclear bomb under Joseph Stalin became the top national priority, referred to as “Goal No. 1.” In 1949, the Soviet Union successfully detonated its first atomic bomb “Joe-1,” followed by the testing of its first hydrogen bomb in 1953 at the Semipalatinsk testing ground in what is now Kazakhstan.

Subsequently, the USSR expanded its nuclear agenda beyond weapons development to include energy production. This effort led to the deployment of the 5 MWe reactor, AM-1 (“peaceful atom”), in 1954 in the closed city of Obninsk, Kaluga region. This reactor, based on graphite-moderated technology initially developed for plutonium production, became the world’s first to generate electricity [109,110]. Further advancements included the development of the fast breeder reactor (FBR), BR-1, in 1955, which was later upgraded to the 5 MWt reactor, BR-5. This reactor supported foundational research for designing sodium-cooled FBRs. In 1983, after significant reconstruction, the BR-5 was converted into the BR-10, a sodium-cooled fast neutron research reactor with an 8 MWt capacity [110]. The BR-10 reactor operated until its decommissioning in 2002.

In 1956, the Soviet Union unveiled an ambitious five-year plan to construct and commission multiple nuclear power plants, aiming for a total capacity of 2175 MWe. Until April 26, 1986 — the date of the Chernobyl nuclear reactor accident — the USSR had been heavily investing in the development of nuclear reactors for peaceful power generation. However, the Chernobyl disaster, which released approximately 8 tonnes of U and around 27 kg of ^{137}Cs into the atmosphere, significantly slowed the progress of the Soviet nuclear industry [109]. As of 2024, the IAEA reported that Russia operates 36 nuclear power reactors with a total capacity of 26,802 MWe, accounting for 18.4% of the nation's total electricity production, with 4 new reactors (total capacity: 3850 MWe) currently under construction [111].

The IAEA reports that Russia produces approximately 700 tonnes of SNF annually from its nuclear power plants. Driven by concerns over uranium availability and the potential efficiency of fast breeder reactors, Russia has strongly advocated for a closed NFC, with ROSATOM (Государственная корпорация по атоМной энергии «РосатоМ»; Gosudarstvennaya korporatsiya po atomnoy energii "Rosatom") aiming to fully implement this by 2030. Since 1977, commercial reprocessing of SNF has been conducted at two sites: Mayak Production Association (Производственное объединение «Маяк»; Proizvodstvennoye obʹyedineniye "Mayak") in Ozersk and the Mining and Chemical Combine (MCC) in Zheleznogorsk. Prior to commercial operations, Mayak developed and tested technologies for the vitrification of liquid radioactive waste between 1969 and 1977. By 2011, about 16% of the SNF from research, naval, VVER-440 (PWR), and BN-600 (sodium-cooled fast breeder) reactors were reprocessed.

As of 2014, Russia held approximately 486 million m^3 (12.83×10^{10} US gallons) of liquid radioactive waste (LRW) with a total radioactivity of 4.27×10^{19} Bq. HLLW accounted for less than 0.5% of the total LRW volume but represented about 42% of the total LRW activity. Similarly, out of 87 million tonnes of solid radioactive waste (SRW), less than 3% was HLW, which contributed approximately 98% radioactivity of the total SRW [112].

In the late 1970s, the Russian Federation (then being a part of USSR) explored various host matrices, including phosphate and borosilicate glasses, for immobilizing highly radioactive waste. Following extensive research, aluminophosphate glasses were chosen for the immobilization of HLW due to their superior ability to incorporate larger amounts of actinides, molybdates, and sulfates compared to borosilicate glasses. Since 1987, the HLW from the RT-1 nuclear fuel reprocessing plant at Mayak has been immobilized in aluminophosphate glass [1]. Meanwhile, borosilicate glasses have been used for immobilizing intermediate-level waste (ILW) [113]. Detailed discussions of the Russian aluminophosphate and borosilicate glasses have been provided elsewhere [5,48]. The representative base glass compositions are given in Table 1. The sodium aluminophosphate glass is particularly notable because it is the only established use of a phosphate glass matrix for immobilizing the HLW arising from commercial reprocessing of SNF. For vitrified HLW used in Russia, properties have been determined including uniformity of composition, maximal heat release, thermal and radiation resistance, leachability, mechanical strength, and thermo-physical parameters [114,115].

Russia commissioned its first JHCM for HLW vitrification, the EP-500/1, at Mayak Production Association in Ozersk in 1987, but it was shut down in 1988 and the second EP-500 melter (EP-500/2) was commissioned, which processed 11,500 m^3 of HLW during 1991 – 1997 [5,48]. Third melter, EP-500/3, processed 8000 m^3 of HLW during 2001 – 2006; fourth melter, EP500/4, processed 8100 m^3 of HLW during 2006 – 2010 [5,48]; and fifth melter, EP500/5, operated from 2016 to 2020. Russia’s Mayak Production Association in Ozersk is currently continuing to install a sixth furnace aiming to continue the HLW vitrification program in Russia [116].

The Mayak EP-500/x JHCM melters use a single-stage vitrification process, wherein phosphoric acid (H_3PO_4) is added directly into the melter, together with waste slurry, producing essentially HLW-loaded sodium aluminophosphate glass. The melters are 9.48 m × 4.2 m × 3.2 m in size and equipped with molybdenum electrodes. They consume on an average 1.52 kWh/L and operate at temperatures as high as 1200 °C. The throughput of these melters is 400 ± 100 L/h of HLW at specific electricity consumption of 2 kW h/L. The level of glass melt in the melter is 0.35 – 0.45 m (Fig. 12).

3.8. India

As of May 2026, India operates 23 nuclear reactors and plans to add 19 more by 2035. India has adopted a closed NFC for the efficient utilization of uranium resources available in the country. The HLLW in India, generated as a result of reprocessing of SNF to extract Pu and U, is stored in underground steel tanks and exhibits 95% of India’s inventory of radioactivity due to the presence of fission products and minor actinides [8,117,118]. For vitrification, India has adopted a three-step strategy involving immobilization, interim storage followed by ultimate geological disposal of HLLW [117]. Currently, there are three waste vitrification facilities operational in India, each co-located with the reprocessing facilities.

Indian history of HLLW vitrification began with the establishment of a research and development program in the 1960s, comprising a wide range of areas, including the selection and optimization of glass matrices, process parameters, and the development of a flowsheet to be deployed for remote operations in hot cells [119]. These efforts led to the establishment of the first Waste Immobilization Plant (WIP) at Tarapur in the 1990s for vitrification of HLLW from power reactors using a multi-zone induction based metallic melter (Fig. 13) [120]. Inconel 690 was the material of choice for the melter vessel and susceptor. To minimize heat loss, thermal insulation was provided between the susceptor and coil. There were multiple copper coils powered and controlled independently of each other, positioned at different zones along the melter vessel, which helped in maintaining each of these zones at different temperatures [119,121]. The top zone was maintained at 130 – 150 °C where the concentrated liquid waste and glass constituents in the form of slurry were introduced and they underwent evaporation. In the next zone below, calcination took place where the temperature was 400 – 700 °C [120]. The calcined material on moving further down to the bottom zone was maintained at 900 – 950 °C for 8 h where it fused into molten glass. The melt was drained into stainless steel AISI 304 L canister by activating the freeze valve at the bottom of the melter [52].

Subsequently, with an aim to increase the efficiency and overcome the limitations of metallic melter, a JHCM was commissioned at Advanced Vitrification System (AVS), Tarapur [8]. A second vitrification plant was commissioned at Bhabha Atomic Research Centre (BARC),

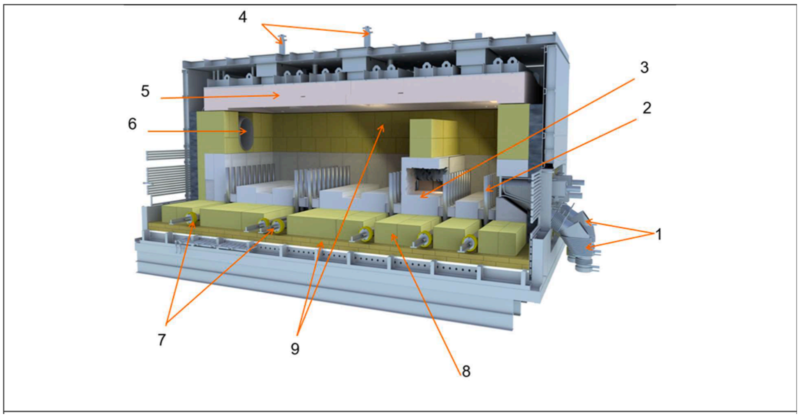

Fig. 12. The design of EP-500 melter. 1 – drainage (pouring unit), 2 - molybdenum electrodes, 3 – overflow window, 4 – feeders, 5 – arch, 6 – off-gas duct, 7 – water cooled electric power supply, 8 – Bakor refractory blocks, 9 – Fireclay refractory blocks. Reprinted from Ojovan [91] *with permission from Elsevier.*

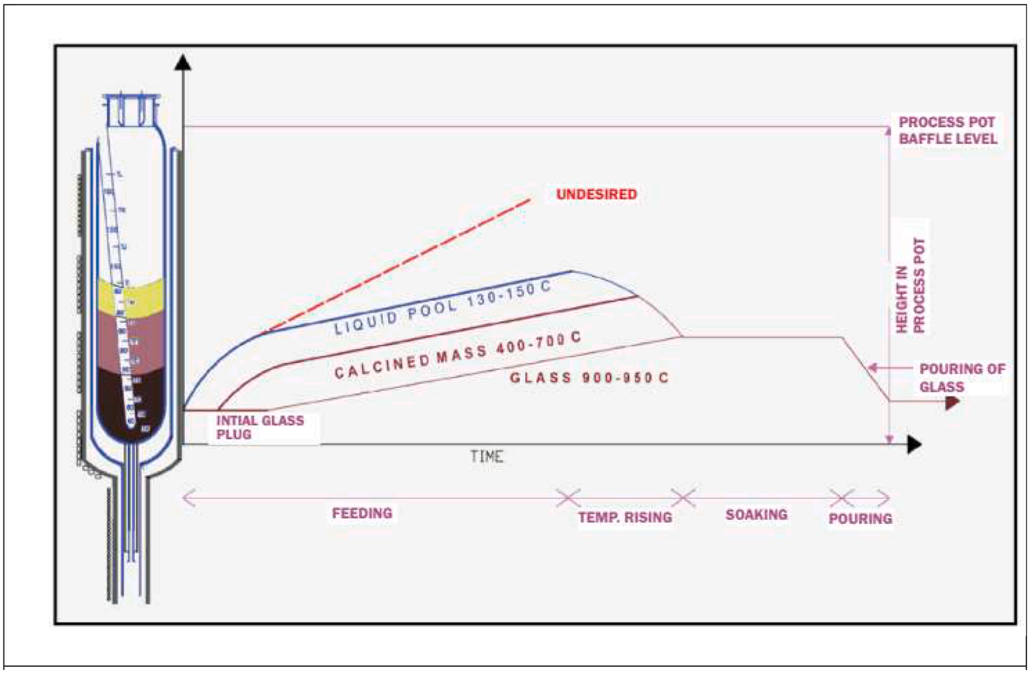

Fig. 13. Schematic of induction based metallic melter with temperature profile along various positions. *Reprinted from* Mehta et al. [120] *with permission from Bhabha Atomic research Centre (BARC) India* .

Trombay in 2002 to immobilize the HLLW resulting from the reprocessing of SNF from research reactors (Fig. 14) [121,122]. A third vitrification facility was commissioned at Kalpakkam in the year 2017, typically for HLLW generated by Pressurized Heavy Water Reactors (PHWR) and Fast Breeder Reactors (FBR) [123]. Considering the diverse compositional chemistries of the HLLW, three glass formulations have been developed: modified borosilicate glass (IR111 Tarapur), lead-based borosilicate glass (WTR-62 Trombay), and barium-based borosilicate glass (SB-44 Trombay) (Table 1).

A typical canister used for the storage of HLLW borosilicate glass in Trombay facility has a diameter of 324 mm, 10 mm thickness and 720 mm in length, and can accommodate around 45 liters of vitrified waste. Three such canisters are placed in a stainless steel overpack having 356 mm outer diameter, 10 mm wall thickness and 2000 mm height [118]. The overpacks are stored in an interim storage facility for 30 – 40 y to investigate the heat decay and radiation stability of vitrified waste product (VWP) before their final disposal in a geological repository.

The first Solid Storage and Surveillance Facility (SSSF), with a capacity of storing 1700 overpacks, was commissioned in Tarapur (Fig. 15) [121]. Another interim storage facility, known as “Vitrified Waste Storage Facility (VWSF),” was commissioned at Kalpakkam in 2020.

To further reduce the volume of VWP, India is adopting an ongoing process known as “partitioning and transmutation of HLLW” which involves the separation of fission products from long-lived actinides (Np, Am, Cm) to be used as fuel for fast reactors or accelerator driven sub-critical systems (ADS). Based on extensive laboratory-scale research and bench-top experiments, the world’s first industrial scale Actinide Separation Demonstration Facility (ASDF) was commissioned in Tarapur in 2013 [117,122]. A second facility was later established at Trombay to separate not only minor actinides but a host of radionuclides like ^{137}Cs, ^{106}Ru, and ^{90}Sr [124].

This partitioning and separation strategy provides dual benefits of reduction of HLLW volume as well as recovery of valuable radionuclides for societal benefits. For example, ^{137}Cs after separation in pure form is vitrified in glass in the form of a pencil (diameter: 23 mm, length: 204

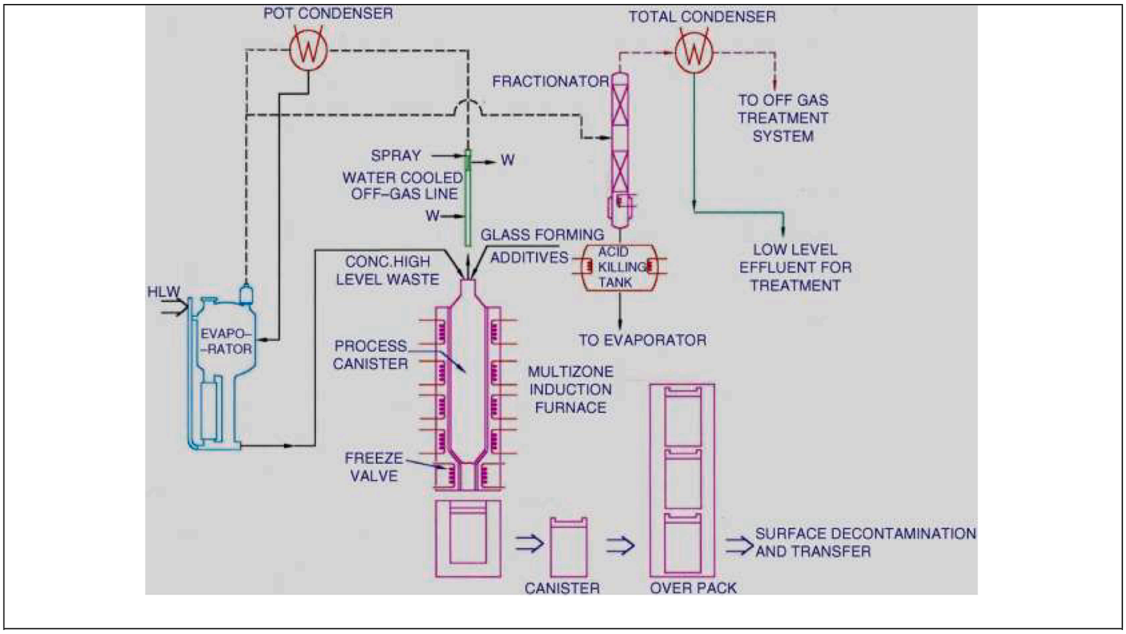

Fig. 14. Schematic of vitrification process employed at Bhabha Atomic research Centre (BARC), India. *Reprinted from* Manohar and Kaushik [122] *with permission from BARC, India.*

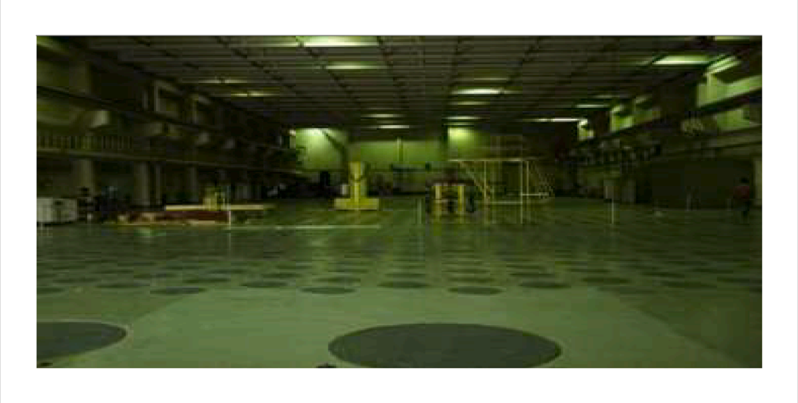

Fig. 15. Solid Storage and Surveillance Facility (SSSF) at Tarapur, India. *Reprint from* Mishra [121] *with permission from BARC, India.*

mm) for use in blood irradiators (Fig. 16) [125,126]. ^{106}Ru is used for brachytherapy in the form of plaque for treatment of eye cancer. ^{90}Sr separated from HLLW, is used as source for milking ^{90}Y, which is used for nuclear medicine [122,127,128]. Through these initiatives, India has the demonstrated the concept of “wealth from waste” — a paradigm shift in the HLLW management.

In parallel, India has also developed and demonstrated CCIM technology for future vitrification of HLLW. An industrial-scale CCIM (diameter: 500 mm) was set up at BARC, Trombay, operated at 1250 °C, having a throughput of 25 l/h [129]. It has completed >100 operations and is expected to be installed in hot cells shortly.

3.9. China

Research into the vitrification of Chinese HLLW began in the late 1970s, drawing on techniques similar to the PIVER process. Initial efforts focused on fundamental studies and laboratory-scale investigations. In the early 1980s, a small-scale cold pilot facility was constructed, completing 30 operational runs by 1986, which accumulated 438 h of operation and yielded 134.5 kg of inactive waste glass. Although valuable experience was gained, the limited throughput of the batch process led to its discontinuation by 1985 in favor of continuous vitrification [130].

In the late 1980s and early 1990s, China transitioned from a batch-based vitrification process to a high-throughput, continuous liquid-fed JHCM technology (also abbreviated as LFCM in some articles/reports) for immobilizing HLLW. To support this shift, a full-scale non-radioactive mock-up facility, known as the Vitrification Plant Mock-up (VPM), was developed by the Beijing Institute of Nuclear Engineering (BINE) in collaboration with the Institut für Nukleare Entsorgungstechnik (INE) of the Karlsruhe Nuclear Research Centre (KfK) and a German industrial consortium (KKN) comprising Siemens-Kraftwerk Union AG (KWU), Kraftanlagen Heidelberg GmbH (KAH), and Nuklearchemie und -metallurgie GmbH (NUKEM). The large-scale melter, constructed and tested at INE in 1992, was delivered to China in 1994 for installation at the VPM facility. This facility was constructed and operated by the China Nuclear Energy Industry Corporation (CNEIC), a key entity under the China National Nuclear Corporation (CNNC), responsible for international nuclear technology trade and project execution [131–133].

The liquid-fed JHCM technology adopted for the VPM facility was designed to continuously feed HLLW and glass-forming additives into a refractory-lined melter. The melter vitrified the waste at 1150°C and

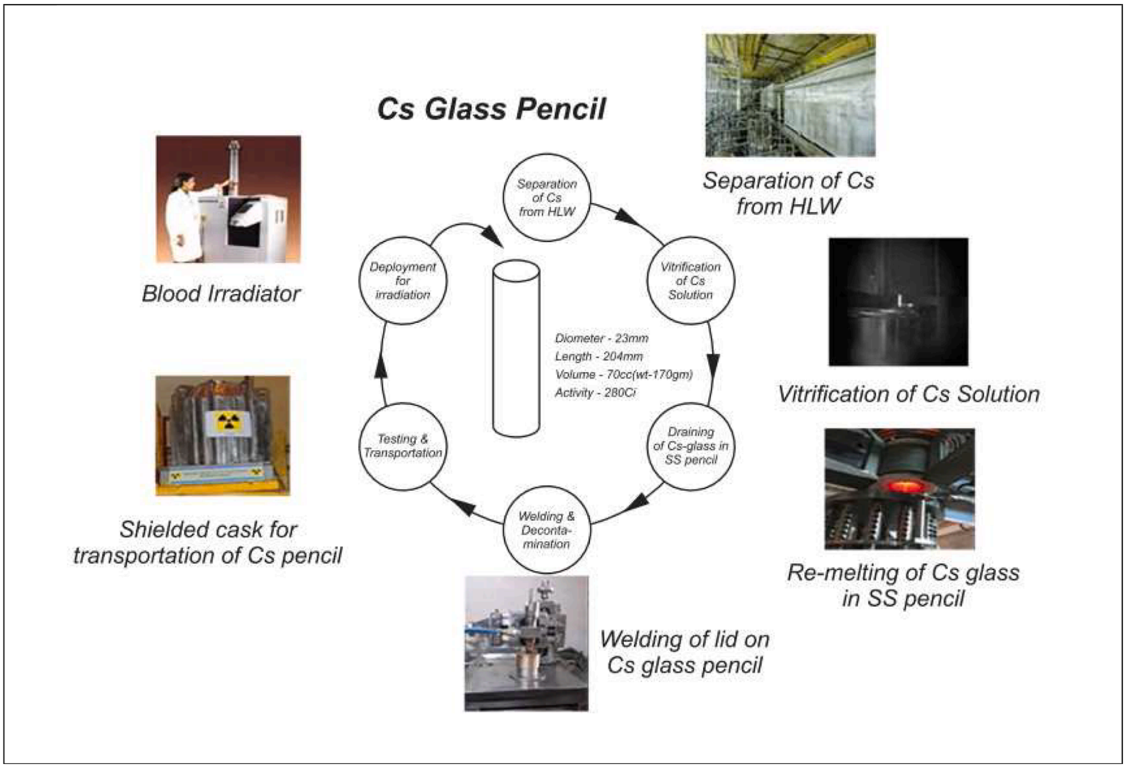

Fig. 16. Overview of process for making cesium glass pencil. *Reprinted from* Diwan et al. [125] and Kaushik [126] *with permission from BARC, India.*

included air bubblers to enhance mixing and heat transfer, along with robust off-gas treatment to manage radionuclides. During two cold commissioning tests in 2000 and 2001 using waste simulants, VPM successfully demonstrated stable glass production, effective melter control, and reliable off-gas handling. This validated the technology's suitability for Chinese HLLW streams, confirmed the glass product's durability, and provided critical engineering data for future scale-up and hot-cell vitrification initiatives [131,133].

Despite successful cold test runs of the liquid-fed JHCM, a critical challenge emerged: the formation of the so-called “yellow phase.” This immiscible salt layer also referred to as “gall” arises when the sulfate content in the Chinese legacy HLLW exceeds its solubility limit in the glass melt. Rich in alkali and alkaline-earth sulfates, the yellow phase also concentrates key radionuclides such as technetium (Tc), strontium (Sr), and cesium (Cs) [132].

Once formed, the salt layer would float on top of the melt resulting in the corrosion of sidewalls of the JHCM and short-circuiting the heating electrodes [26,134,135]. Hence, the practical concentration of these species in the borosilicate waste form was intentionally kept much lower than their solubility, thus, limiting the target waste loading to 16 wt.% [132,133].

To address this limitation, a two-pronged strategy was adopted: (1) Exploration Cold Crucible Induction Melter (CCIM) technology and (2) Optimization of glass formulation for JHCM, as discussed below.

In the early 2000s, CEA (France) proposed CCIM technology for the vitrification of Chinese HLLW. CCIM enables direct induction heating of the melt, allowing for significantly higher operating temperatures without compromising equipment integrity. This enabled the tailoring of glass composition, thus effectively suppressing the formation of the yellow phase. All process components were water-cooled and shielded by a self-forming solidified glass layer, which mitigated corrosion and extends melter’s lifespan, potentially indefinitely. The system’s modular, compact, and remotely maintainable design further enhanced operational reliability. Moreover, the absence of an upper temperature limit allowed CCIM to accommodate glass formulations with higher fractions of glass formers, enabling enhanced waste loading and waste form performance.

Accordingly, a proof-of-concept demonstration was conducted at CEA-Marcoule in 2003. During the trial, 1114 liters of sulfate-rich simulant Chinese HLLW (relative density at room temperature: 1.22; water content: 90 wt.%) was fed into a 650 mm diameter CCIM at a rate of 20 l/h. Simultaneously, a borosilicate glass frit with composition (2.3 Li_2O – 5.0 Na_2O – 1.0 MgO – 5.4 CaO – 3.5 Al_2O_3 – 1.0 TiO_2 – 22.0 B_2O_3 – 59.8 SiO_2, wt.%) was introduced into the melter in the form of 1 – 2 mm diameter beads. Operating at 1250 °C, the melter produced 730 kg of borosilicate glass over 62 h, with an additional 210 kg of glass extracted post-operation. The overall production rate was 11.8 kg/h, encompassing all operational phases, including those used to determine the system’s maximum capacity [132].

In parallel, efforts to optimize the JHCM glass formulation continued. In 2008, KIT (Germany) developed a new composition tailored for sulfate-rich feeds. While the original glass developed by the China Institute of Atomic Energy contained ~82 wt.% of $SiO_2 + B_2O_3$, the KIT formulation incorporated higher levels of alkaline-earth oxides such as CaO, MgO, and BaO, known to enhance sulfate solubility. Cold tests at KIT using 6.5 m^3 of waste simulant demonstrated that the new glass could accommodate >1 wt.% SO_4^{2-}, effectively suppressing yellow phase formation, while this improvement did not raise the total waste loading beyond the 16 wt.% design target. This is because the limitation is not solely driven by sulfate solubility but also by the overall concentrations of other key components (e.g., alkalis, actinides) and the need to meet stringent glass durability standards. This formulation (Table 1) ultimately served as the basis for glass production at the Vitrification Plant China (VPC), the first HLLW vitrification facility in China.

During the VPC development phase, CCIM remained in the pilot stage, lacking mature large-scale operation data and facing severe engineering constraints. The optimized JHCM system—with proven maturity, lower cost, and simpler operation—was selected for the first Chinese vitrification facility.

In 2010, the Sichuan Environmental and Protection Engineering

Company (SEPEC) signed an agreement with China Nuclear Power Engineering (CNPE) to construct and operate the VPC melter. Although the VPC shares the general, liquid-fed JHCM configuration with Germany's Karlsruhe Vitrification Facility (VEK), which processes HAWC, it incorporates key adaptive modifications tailored to China's sulfate-rich legacy waste. These include optimized melt-pool geometry to improve mixing and upgraded off-gas treatment systems to handle higher levels of corrosive species. Construction of the facility began in 2014 in Guangyuan, Sichuan province, and the melter was commissioned in 2021 [98]. The VPC melter vitrifies the legacy defense wastes, but the technology may be usable later for civil waste [136].

3.10. Summary

To date, HLW vitrification has been implemented in several countries, successfully immobilizing HLW into a durable waste form suitable for disposal. The WTP at Hanford, which is currently undergoing commissioning, will be a recent addition when it becomes operational. Table 3 shows a current summary of the international HLW vitrification programs.

4. Present status and future directions

4.1. United States of America

In the United States, legacy HLW is largely consolidated at three locations: (1) Hanford Site in Washington State, (2) SRS in South Carolina, and (3) Idaho Nuclear Technical and Engineering Center (INTEC) in Idaho State. The WVDP completed the immobilization of all West Valley HLW (from the reprocessing of commercial SNF) into glass in 2002, which is stored on site in West Valley, New York State.

As mentioned earlier, the production scale vitrification of HLW in the U.S. began with the DWPF (SRS) and WVDP in 1996 [139,140]. To date, DWPF has produced >4200 HLW borosilicate glass canisters and permanently closed eight HLW waste tanks by filling them with grout. The remaining legacy HLW inventory at the SRS is expected to be vitrified by 2035, producing an additional 4000 canisters.

The Hanford Site holds approximately 56 million gallons (about 212 million liters) of radioactive and chemical waste, stored in 177 underground steel tanks (149 single-shell and 28 double-shell). Hanford was the source of the plutonium used in the world's first nuclear detonation—code-named "Trinity"—in Alamogordo, New Mexico, in July 1945, as well as in the second atomic bomb, "Fat Man" previously mentioned. Today, Hanford accounts for ~ 60% of the tank waste currently in storage in the United States. Most of the remaining waste is stored at SRS 1. Fig. 17 illustrates a cross-sectional schematic of a typical waste tank at the Hanford Site, showing the layered arrangement of radioactive and chemical waste. These tanks are distributed across the 200-East (91 tanks) and 200-West (86 tanks) areas. The contents of the Hanford tanks will be separated into two categories: (1) HLW, constituting approximately 10 vol.% of the total inventory with 95% of the radioactivity, and (2) Low-Activity Waste (LAW), constituting approximately 90 vol.% of the waste inventory with <5% of the radioactivity.

Table 3
Successful industrial implementation of HLW vitrification in the world [91,137,138].

Country	Vitrification facilities	Immobilized activity, TBq	Total produced quantity, tonnes
France	R7/T7 (La Hague)	3.75×10^8	9800 (end 2022) In progress
	AVM (Marcoule)	2.2×10^7	1220 (end 2022) Campaign completed
USA	DWPF (Savannah River)	2.7×10^6	7870 (1996 – 2024)
	WVDP (West Valley)	8.9×10^5	550 (1996 – 2002) Campaign completed
	WTP HLW (Hanford)	-	Expected to start in 2035
Russia	EP-500 (MCC, Mayak)	2.8×10^7	7700 (1987 – 2020)
UK	WVP (Sellafield)	5.9×10^7 (estimate)	2500 (on April 1, 2022)
Belgium	Pamela (Mol)	0.5×10^6	500 (1985 – campaign completed)
Japan	TRP (Tokai)	2.5×10^6	100 (1995 – 2022)[#]
Germany	VEK (Karlsruhe)	0.8×10^6	55 (2010 – campaign completed)
India	WIP (Tarapur), AVS (Tarapur), WIP (Trombay)	9.6×10^3	28 (2012 – In progress)

[#] At the time of writing, the TRP facility is suspended while melters #2 and #3 are being replaced.

The U.S. Department of Energy (DOE) is constructing the Waste Treatment and Immobilization Plant (WTP) in the 200-East Area of the Hanford Site, covering approximately 27 hectares, to separately vitrify low-activity waste (LAW) and high-level waste (HLW) into borosilicate glass at 1150°C using joule-heated ceramic melter (JHCM) technology. The LAW facility has been completed and has commenced treatment of waste retrieved from tanks in the 200-East Area, whereas the HLW facility remains under construction. Under current infrastructure constraints, vitrification activities are primarily focused on wastes stored in the 200-East Area, while treatment pathways for tank wastes in the 200-West Area remain dependent on additional infrastructure development and operational considerations. Consequently, in addition to the LAW and HLW vitrification flowsheets, a third flowsheet—referred to as the West Area Risk Management (WARM) project—is being developed to retrieve and treat the supernate and dissolved saltcake from single-shell tanks in the 200-West Area, followed by grouting and off-site disposal of the decontaminated waste.

Recent developments at the Hanford Waste Treatment and Immobilization Plant (WTP) have also led to the evaluation of alternative processing strategies, including Direct Feed Low-Activity Waste (DFLAW) and Direct Feed High-Level Waste (DFHLW) approaches.[*] These strategies emerged not only from engineering and operational challenges associated with the Pretreatment Facility, including management of high-shear-strength slurries and process complexity, but also from major advances in the design of HLW glass formulations. Earlier WTP baseline glass formulations were generally limited to Al_2O_3 concentrations of ~13 wt.% while meeting required melter throughput constraints, which necessitated extensive caustic and oxidative leaching operations for aluminum and chromium removal. However, subsequent Enhanced Waste Glass (EWG) studies demonstrated significantly higher tolerance for these components, with Al_2O_3 concentrations approaching ~30 wt.% in advanced HLW glass formulations and Cr loadings substantially exceeding earlier baseline limits [50]. Recent analyses have suggested that advanced glass formulations may significantly reduce projected HLW glass inventories even under limited-leaching or no-leaching flowsheets [141–144].

Analyses based on advanced glass models projected that the total Hanford HLW inventory could potentially be immobilized in as few as ~13,530 canisters (~40,860 metric tonnes of glass) and reduced-leaching flowsheets as compared to larger inventories predicted using

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[*] To expedite the treatment of radioactive waste, the US DOE is exploring a sequenced approach for vitrifying LAW and HLW at the Hanford Site. This strategy involves two parallel pathways: Direct Feed LAW (DFLAW) and Direct Feed HLW (DFHLW). In the DFLAW process, waste from the tank farms is pretreated by filtration and subsequent ion exchange (by crystalline silicotitanate) in the Tank Side Cesium Removal (TSCR) facility. Similarly, DFHLW would allow HLW to be processed without relying on the pre-treatment facility, potentially enabling earlier startup of vitrification operations. For HLW, this could involve omitting or replacing ultrafiltration and caustic leaching steps—typically carried out in the pre-treatment facility—with interim alternatives such as in-tank leaching and settling.

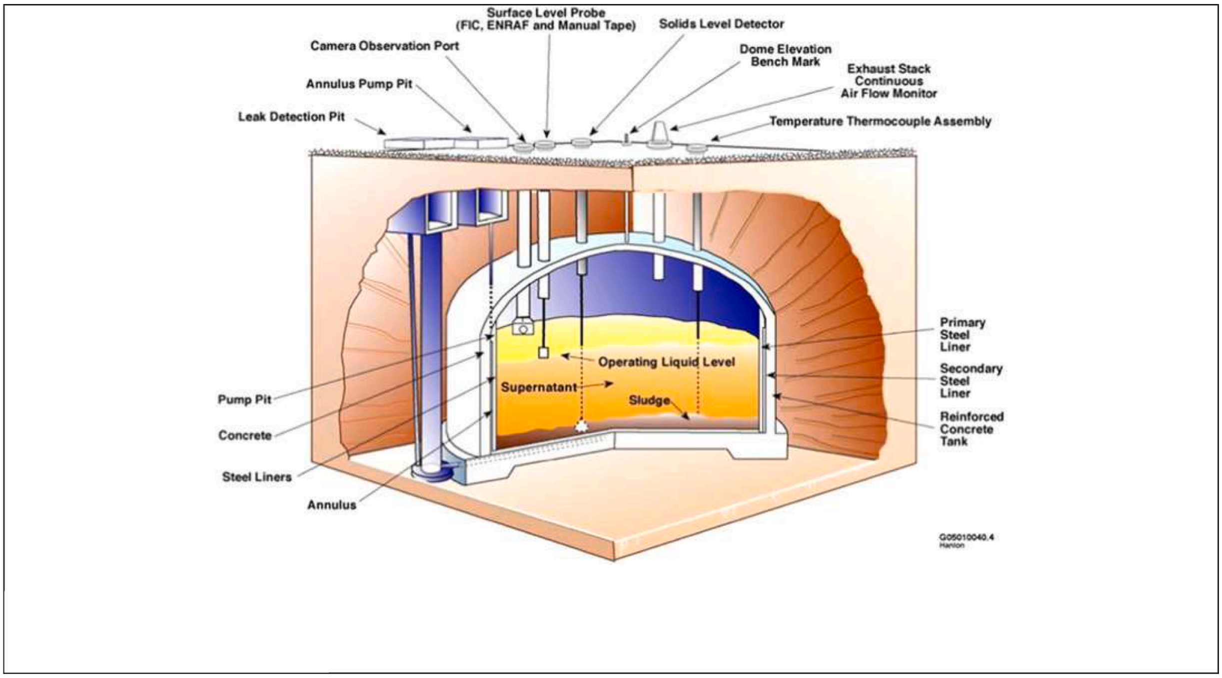

Fig. 17. Cross-sectional schematic of a stainless-steel tank. *Reprinted from* Goel et al. [3] *with permission from Elsevier.*

earlier formulation constraints [50,141,143]. At the same time, direct-feed approaches entail significant engineering and processing trade-offs, including the potential for increased HLW glass volume under conservative operating conditions. Nevertheless, improved understanding of crystallization phenomena, sulfur tolerance, and melter behavior has reduced some earlier concerns regarding melter operability at high waste loadings [3]. In particular, Hanford-relevant Cr–Ni–Fe spinel phases have been shown to form predominantly as relatively fine crystals (typically tens of micrometers in size) that remain suspended within the melt and are discharged during routine melter operation rather than accumulating as large agglomerates capable of obstructing melter flow paths or significantly shortening melter lifetime. These developments have contributed to renewed interest in higher waste-loading strategies and simplified vitrification flowsheets for Hanford HLW treatment. The HLW vitrification using a DF flowsheet is anticipated to begin by 2035. For detailed definitions, distinctions in the chemical compositions of HLW and LAW, and an in-depth discussion of the scientific and technical challenges involved in vitrifying HLW at the WTP, please refer to our related review articles [3,18].

The calcined HLW at INTEC may need additional treatment prior to disposal primarily to treat chemical hazards such as Hg and Cd. One potential treatment option is vitrification since it has been declared the “best-demonstrated-available technology” for treating mixed hazardous HLW by the US Environmental Protection Agency [145]. However, as per the amended record of decision published by the U.S. DOE on January 4, 2010 [146], it was decided that the calcined HLW will be converted to a glass-ceramic product using hot-isostatic pressing (HIP), with completion of treatment by a target date of December 31, 2035. In 2016, an independent analysis recommended deferring any final decision on treatment technology until the disposal path was better defined. More recently, five vitrification technologies have been evaluated for vitrifying the calcine [147–149]:

- Induction Heated Metallic Melter (IHMM) [French Alternative Energies and Atomic Energy Commission (CEA) and Orano]
- Cold Crucible Induction Melting (CCIM) [CEA and Orano]
- DEM&Melt In-Can Melter (ICM) [CEA and Orano]
- GeoMelt In-Container Vitrification (ICV) [Veolia Nuclear Services]
- JHCM [Atkins-Realis]

Several types of glass matrices have been designed and developed considering the (1) chemistry of calcines, i.e., Zr-based calcines, Al-based calcines and Al/Na blend calcine, and (2) temperatures reachable with different vitrification processes.

Apart from the legacy waste, the U.S. also has an HLW inventory of SNF stored at various power reactor sites throughout the U.S., including over 90,000 t of commercial SNF, based on heavy metal content or tHM)[†] and over 3400 tHM of US DOE and research reactor SNF [150]. This inventory is a result of the NFC deployed in the U.S., and the SNF will be dispositioned without reprocessing. Currently, the U.S. DOE is exploring the possibility of storing SNF at one or more consolidated federal interim storage facilities using a consent-based siting process.

Additionally, the U.S. is considering the deployment of several Generation IV advanced reactors (ARs) to address growing energy demands and to reduce carbon emissions in the power sector. While this growth will increase the inventory of HLW, advancements in back-end fuel processing and innovative reactor designs are expected to reduce the overall volumes of HLW per KWh electricity generated. Vitrification is anticipated to play a critical role in the future of the nuclear industry.

4.2. France

Beyond legacy HLW, France’s civil HLW is primarily generated by its 57 PWRs operating under a closed NFC, using UO_x and mixed oxide (MO_x) SNF. The HLW associated with UO_x SNF contains elements, like nitrates, alkali metals, alkaline earths, chalcogenides, transition metals, noble metals, lanthanides, and minor actinides. The recycling of MO_x fuel has begun, as a French – Japanese collaboration, and is expected to produce HLW with higher actinide concentrations than the UO_x stream. According to a World Nuclear News article published in February 2025 [151], Japan’s Federation of Electric Power Companies (FEPC) has

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[†] According to the website of the United States Nuclear Regulatory Commission, as of February 21, 2025, a total of 94 licensed light water reactors (LWRs) – comprising pressurized water reactors (PWRs) and boiling water reactors – are currently operating in the United States. According to the website of US DOE – Office of Nuclear Energy, the nuclear power reactors in the United States generate approximately 2,000 metric tonnes of SNF each year.

announced plans to transport 400 tonnes of MO_x-based SNF to Orano (a French commercial waste processing company; formerly known as AREVA and Cogema) for reprocessing research. Additionally, France announced plans in March 2024 to extend its nuclear fuel recycling program beyond 2040, including studies for a new MO_x fuel fabrication plant [152].

For civil HLW vitrification, the CEA and Orano employ R7T7-type borosilicate glass at the La Hague plant. Since 2020, CCIM technology has been used to vitrify very high-level fission products. The R7T7 UO_x HLW glass achieves an average incorporation rate of 1 Ci/g and can be loaded with up to 3 wt.% platinoids (RuO_2, Rh, Pd) [153].

Looking ahead, France aims to construct at least six new European Pressurized Reactors (EPRs) by 2050, anticipating increased HLW volumes. To address this, CEA and Orano are pursuing developments aimed not only at increasing waste loading, but also at improving vitrification capacity, reducing the footprint of vitrification facilities, and strengthening process robustness and reliability. Future glass formulations must also remain flexible enough to accommodate the evolution of spent fuel inventories, including UO_x and MO_x fuels, higher burnup fuels, and potential future fast-reactor fuel cycles. Depending on waste composition, these matrices may remain fully glassy or may deliberately allow controlled crystallization in glass-ceramic waste forms to immobilize components with limited solubility in conventional borosilicate glasses.

Meanwhile, next-generation reactor technologies are being explored, which once deployed are expected to generate HLW chemistries different from the current legacy wastes.

When discussing the plans for disposal of HLW glass, France intends to dispose of its vitrified HLW in a deep geological repository. The deep geological disposal concept is similar to those adopted by other countries such as Finland and Sweden. However, while France plans to dispose of vitrified HLW, Finland and Sweden intend to dispose of SNF directly. To this end, the French radioactive waste management agency, Andra, is developing a repository, named Cigéo, about 500 m below the groundwater table in clay formations near Bure, in eastern France. This facility is intended for the disposal of 83,000 m^3 (~22 million US gallons) of vitrified HLW and long-lived intermediate-level waste (ILW). The excavated volume required for vitrified HLW is approximately 5 × smaller than that needed for direct disposed SNF from a once-through fuel cycle [154]. The repository is designed to operate at temperatures of up to 90 °C, a condition expected to be reached approximately 20 years after waste emplacement and offers favorable conditions for long-term containment, including low radionuclide mobility under reducing conditions. A license application for its construction was submitted in 2013 [155]. The construction is expected to begin in 2027 and is expected to cost €26 – 37.5 billion.

4.3. United Kingdom

The UK's spent fuel processing plants, THORP (Thermal Oxide Reprocessing Plant) and Magnox, were shut down in 2018 and 2022, respectively. THORP was used for reprocessing the enriched UO_x fuels from the Advanced Gas-Cooled Reactors (AGRs) / LWRs, while Magnox was used for reprocessing natural U-based fuel from 1st generation Magnox reactors. With these closures, the UK's HLW vitrification is now moving toward Post-Operational Clean Out (POCO) activities for Highly Active Liquor (HAL). The current plan is to have all UK HLW produced, treated, and in interim storage by 2030 [156]. It is anticipated that POCO of the Sellafield Highly Active Liquor Evaporation and Storage (HALES) plant will commence in 2025, and POCO of the Sellafield WVP will commence in 2030 [157].

To immobilize the Mo-rich HLW expected to be generated from POCO activities, the "CaZn" glass has been developed, tested on the VTR, and has been successfully implemented at WVP since 2018. This glass (Table 1) represents a substantial change from the "MW" formulation and introduces Al_2O_3, CaO and ZnO while also changing the proportions of the other oxides [9,39,158]. If vitrified at the desired oxide waste loading of 25 – 35 wt.% using an alkali borosilicate glass such as "MW," the POCO-generated Mo-rich wastes could lead to the formation of water soluble, Na_2MoO_4-based "yellow phase" [39,92]. However, the "CaZn" base glass preferentially forms insoluble powellite ($CaMoO_4$) phases, suppressing formation of soluble alkali molybdate phases [39,92,158].

As of April 1, 2022, the UK had 2500 tonnes (6191 steel canisters) of vitrified HLW. The future outlook for the UK HLW vitrification focuses on completion of the vitrification campaigns for remaining HAL and POCO vitrification activities. Research will also continue to improve the understanding of the long-term behavior of these vitrified products in interim storage and geological disposal conditions.

The UK is progressing toward the development of a Geological Disposal Facility (GDF) for vitrified HLW glass, following a consent-based, partnership approach. According to Nuclear Waste Services (NWS), a UK government organization, while no specific site has yet been selected, all necessary consents and permits have been secured.

A Test of Public Support has confirmed the host community's willingness to accommodate the facility. The latest projections indicate that the GDF could begin receiving ILW between 2050 and 2060, with HLW glass and SNF disposal starting in 2075.

4.4. Canada

Since Canada follows an open fuel cycle, it does not reprocess SNF or actively pursue the vitrification of HLW. Instead, the country is focused on developing a deep geological repository (DGR) for the long-term disposal of SNF and vitrified HLW.

After a lengthy consent-based siting process, the Nuclear Waste Management Organization (NWMO) has identified the Wabigoon Lake Ojibway Nation and the Township of Ignace as potential host communities for the DGR. The proposed repository will be constructed 650–800 m underground in a stable rock formation, with an underground footprint of ~600 hectares.

Following the completion of regulatory and licensing approvals, construction of the DGR is expected to begin in the early 2030s, with an estimated budget of $26 billion. Operations are anticipated to commence between 2040 and 2045.

4.5. Germany

As outlined in Section 2.5, until 2005, German nuclear power plant operators sent SNF to reprocessing facilities in France (La Hague) and the United Kingdom (Sellafield). The resulting vitrified HLW has since been gradually repatriated to Germany. According to the official website of the Bundesgesellschaft für Zwischenlagerung mbH (BGZ; https://rueckfuehrung.bgz.de/en/), Germany's federal company responsible for the interim storage of nuclear waste, several shipments of vitrified HLW have been completed in recent years. In November 2020, six casks of vitrified HLW were transported from the UK to the Biblis interim storage facility. In November 2024, four CASTOR (Cask for Storage and Transport of Radioactive Material) casks containing vitrified HLW from France were delivered to the Philippsburg interim storage facility. Most recently, in April 2025, seven CASTOR casks were returned from Sellafield to the Isar interim storage facility in Bavaria. Germany is still awaiting the arrival of an additional seven casks of vitrified HLW, which are scheduled to be stored at the Brokdorf interim storage facility. The distribution of vitrified HLW casks to the four sites – Biblis, Philippsburg, Isar and Brokdorf – is based on regional "polluter pays" principle, ensuring the waste is stored in regions where it originated. The casks will later be moved from these four facilities to a repository or to a nearby conditioning plant.

Germany is also actively developing a deep geological repository for the final disposal of HLW, including vitrified waste. The Bundesgesellschaft für Endlagerung (BGE), a federally owned company, is responsible for this process. Initially, the site selection process was scheduled to be completed by 2031 [159]. However, in 2022, BGE announced a significant delay, projecting completion between 2046 and 2068 [159]. A subsequent report published in 2024 by the Federal Office for the Safety of Nuclear Waste Management (BASE – Bundesamt für die Sicherheit der nuklearen Entsorgung), which oversees BGE, estimated the site might not be selected until 2074 [160]. In the meantime, vitrified HLW is securely stored in CASTOR® casks at licensed interim storage facilities, designed for long-term dry storage.

With ongoing delays in implementing the geological repository in Germany, long-term interim storage has emerged as a major technical and regulatory challenge for the management of vitrified HLW and SNF. Most German CASTOR® dual-purpose casks were originally licensed for interim storage periods of approximately 40 years. However, extended storage durations have increased attention toward aging management of storage and transportation systems. Potential degradation mechanisms include radiation-induced embrittlement, thermal and mechanical creep, relaxation of bolted joints, aging of metallic gasket systems, hydrogen generation through radiolysis, and corrosion-related degradation [161]. These concerns are particularly important because several safety-critical components, including the cask body, basket assembly, primary lid system, bolts, and metallic seals, cannot be readily replaced after prolonged storage periods [161]. To address these issues, modern dual-purpose cask systems incorporate multiple engineering safeguards, including double-lid containment systems, inert gas atmospheres, multilayer corrosion-protection coatings, and continuous monitoring capabilities. In addition, periodic safety reassessments, inspection programs, and updated transport safety evaluations are expected to play increasingly important roles as interim storage durations extend beyond their originally anticipated operational lifetimes. Nevertheless, operational experience accumulated from more than 1000 CASTOR® casks worldwide, representing over 10,000 years of storage, has thus far demonstrated no significant degradation affecting their required safety functions [161–163].

4.6. Japan

In Japan, SNF is reprocessed to recover uranium and plutonium for reuse within the nuclear fuel cycle. The remaining high-level liquid waste is immobilized by incorporating it into a glassy matrix through vitrification. As of March 31, 2024, the total volume of SNF generated is equivalent to approximately 27,000 canisters of vitrified waste. Currently, 2505 stainless steel canisters containing vitrified HLW are stored in the country.

The Rokkasho Reprocessing Plant (RRP), still under construction, is scheduled for completion by March 2027. Once operational, it will have the capacity to reprocess more than 800 tonnes of SNF annually and extract around 8 tonnes of plutonium for use in MO_x fuel fabrication.

Japan is actively pursuing advancements in vitrification technology, including the development of glass formulations with higher waste-loading capacities and improved tolerance to platinum group elements (PGEs), as well as strategies to suppress the crystallization of molybdenum-rich “yellow phase” precipitates.

In parallel, Japan is planning the establishment of a deep geological repository for the long-term disposal of vitrified HLW. According to the official website of Nuclear Waste Management Organization of Japan (NUMO, ニューモ - 原子力発電環境整備機構; https://www.numo.or.jp/en), the facility will be constructed in stable bedrock at depths exceeding 300 m. The NUMO is currently working to identify a suitable site. The site selection process involves three stages: (1) literature survey, (2) preliminary investigation, and (3) detailed investigation. Once operational, the repository is expected to dispose of over 40,000 vitrified HLW overpacks and 19,000 m^3 of low-level radioactive waste. According to 2020 estimates from the Agency for Natural Resources and Energy (ANRE, 資源エネルギー庁, Shigen-enerugī-chō), the total cost of this project, including technology development, surveys, land acquisition, design and construction, operation, dismantling and closure, monitoring, and project management, should be around 4 trillion yen (∼ $28 billion).

4.7. Russia

Russia is implementing a two-component nuclear power generation strategy that utilizes both slow and fast neutron reactors within a closed NFC. This approach aims to reduce the need for uranium by recycling actinides and to extract useful isotopes from the process [164–166]. As of June 2025, Russia operates 36 nuclear reactors across 11 nuclear power plants, with an additional 7 reactors under construction, totaling a combined capacity of 5290 MWe. Looking ahead, Russia aims to significantly expand its nuclear energy program, targeting an installed nuclear power capacity of 33.6 GW by 2035 to enhance energy security and reduce carbon emissions.

Russia generates ∼ 800 metric tonnes of heavy metal (MTHM) of SNF annually including SNF returned from Russian-designed reactors operating abroad. The reprocessing of 1 MTHM of SNF results in the production of approximately 13 – 31 m^3 of HLW, 59 – 78 m^3 of ILW, and 1552 – 1875 m^3 of low-level waste (LLW) [167]. As the number of reactors increases, a corresponding rise in HLW generation is expected.

Although detailed updates on HLW vitrification efforts since 2020 are limited, two reports from 2021 and 2022 indicate that the Mayak Production Association is planning the construction of a new vitrification facility with an annual capacity of 490 tonnes of HLW [168,169]. The facility, scheduled to become operational by 2028, will utilize two types of JHCM: the EP-250/6 melter for processing liquid HLW and selected SNF into aluminophosphate glass, and a smaller EPM melter for converting reprocessing waste from power reactor fuel into borosilicate glass. Construction and equipment installation were slated to begin in 2023, with commissioning expected in 2028 [168,169].

In parallel, Russia is making steady progress in developing a deep geological repository for HLW [167]. The country’s first underground repository for vitrified HLW and cemented ILW is planned for construction in the crystalline Archean granite–gneiss formations of the Yeniseisky site in the Krasnoyarsk region of Siberia. The geological and hydrogeological conditions at this location are considered favorable and are comparable to internationally recognized repository sites such as Forsmark in Sweden and Olkiluoto in Finland [167]. According to the World Nuclear Association, a decision regarding the construction of the repository is anticipated in 2025, with completion targeted for 2035. The first phase of the facility is expected to accommodate 20,000 tonnes of ILW and HLW, with provisions for retrievable storage.

4.8. India

India has initiated a site selection program to identify suitable geological locations for the long-term disposal of vitrified HLLW. Granite is the considered host rock, with a sodium smectite clay–based engineered barrier system. The design of Indian geological repository comprises a main access shaft (6 m in diameter) and a ventilation shaft (4 m in diameter). The facility at 500 m depth comprises two orthogonal transportation tunnels, each extending 800 m. A total of 63 disposal tunnels, each with a capacity of holding about 40 waste overpacks, will be aligned perpendicular to the transportation tunnels and parallel to the direction of principle *in situ* stresses [170] Fig. 18 **(a and b)** present the schematic of planned layout of disposal pit and four panels of the Indian geological disposal repository, respectively.

Considering India's projected growth of nuclear power, nearly 10^4 t of vitrified waste is expected to be generated by 2075 without employing partitioning of HLLW, which will be stored at proposed interim storage facilities like Kalpakkam. Projections indicate that in addition to existing interim storage facilities, eight more will be needed to accommodate this future waste volume [117].

However, with India actively pursuing the partitioning of minor actinides and fission products, as per operational data, the volume of

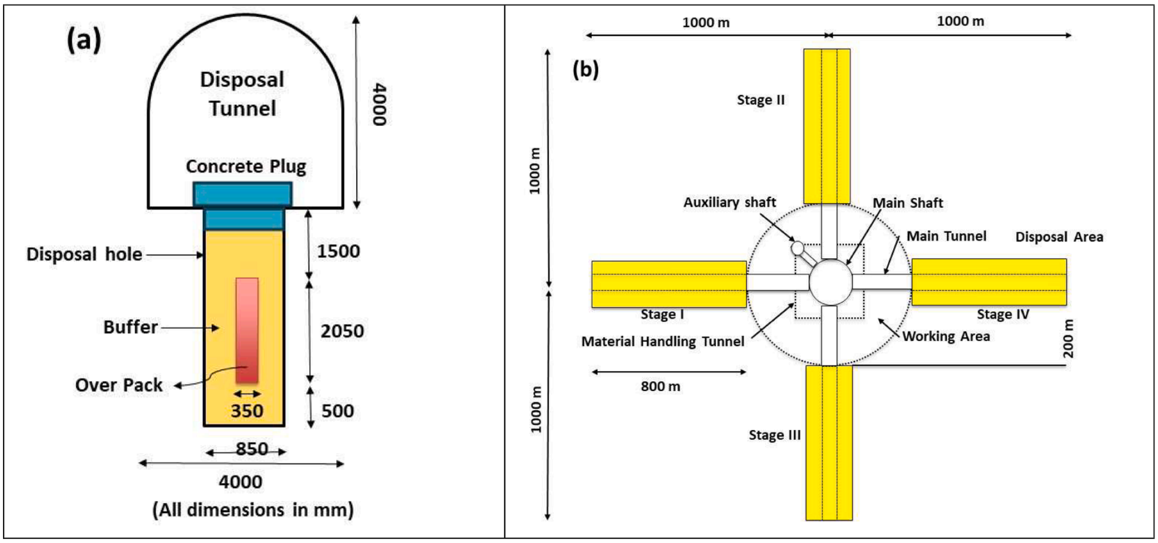

Fig. 18. (a) Layout of disposal pit in Indian context and (b) Schematic view of four panel Indian geological repository. Reprinted from Narayan and Bajpai [170] with permission from Indian Association of Nuclear Chemists and Allied Scientists.

vitrified waste is expected to decrease by 15 to 20 times [117]. This advancement in reprocessing technology means the need for a geological disposal facility is not an immediate concern. Instead, a facility will be constructed on a modular basis, with the shifting of vitrified waste from interim storage expected to begin only after 2072, depending on its cooling period.

4.9. China

As of May 2026, China has 58 nuclear power units in operation, with an additional 30 units under construction and 23 units approved for construction, for a combined capacity of 113 GW. This makes China the world's largest nuclear power infrastructure country [171]. Currently, China generates approximately 1500 metric tonnes of SNF annually and this production is projected to increase to ~2400 metric tonnes by 2035.

China will adopt a closed NFC for its commercial reactors and the generated HLW will be vitrified and stored in a 500 m deep geological repository [172,173]. Accordingly, a SNF reprocessing plant with an annual capacity of 800 metric tonnes is under construction to address the growing needs for fuel cycle management [174]. A huge amount of high-level waste (HLW) will thus be generated from the commercial reprocessing plant. With the VPC now operational and a nearly identical vitrification plant under construction in Diwopu, Gansu province, the need for additional vitrification facilities appears inevitable. To support this expansion, an inactive melter facility will be established in Diwopu, Gansu province to gather operational insights and conduct vitrification trials. By 2030, the infrastructure is expected to include an inactive melter, an active melter, and an interim storage facility for waste glass canisters [175].

For the long-term storage and disposal of the HLW glass, construction of a deep geological disposal underground research laboratory (URL) was initiated in Beishan, Gansu Province in 2021. Plans are in place to construct experimental platforms at depths of 280 m and 560 m underground, respectively. To date, the first platform has been completed and opened to international research access. The on-site repository near the underground laboratory is planned to be operational by 2050, and will accept produced and interim stored HLW glass [176–179].

4.10. Other countries

Beyond the major nuclear nations like the U.S., U.K., France, Germany, China, Russia, India, and Canada, a number of other countries—such as Australia, South Korea, and Slovakia—are charting distinct pathways for nuclear waste management. Their approaches reflect smaller but strategically significant programs, with emphasis on long-term disposal concepts, international collaboration, and technology development. Although Australia does not generate nuclear power, it operates a multipurpose research reactor, which plays a critical role in supplying medical isotopes for both domestic and international healthcare applications. Additionally, the reactor serves as a neutron source for materials science research and supports industrial irradiation processes, such as the treatment of silicon ingots used in high-performance semiconductors. As a result, Australia has been engaged in nuclear waste treatment research for several decades, focusing on advanced waste forms including ceramic and glass-ceramic materials based on SYNROC [180], as well as more conventional sodium borosilicate glasses. Currently, the country employs a glass waste form for immobilizing ILW through hot isostatic pressing (HIP) technology. A first-of-its-kind SYNROC facility is now being commissioned to support this effort. Given HIP's versatility, Australia continues to advance the technology for producing glass-ceramic and ceramic waste forms suitable for both HLW and ILW streams less compatible with traditional vitrification [180].

South Korea has yet to establish a strategy for SNF reprocessing and HLW vitrification. However, under the Special Act on High-Level Radioactive Waste Management enacted in 2025, a deep geological repository for SNF is scheduled to become operational by 2060.

Slovakia's HLW vitrification program began in 1997, targeting the immobilization of 31 m^3 (~8189 US gallons) of aqueous HLW derived from SNF stored in a 3–5% solution of potassium chromate and dichromate (K_2CrO_4 and $K_2Cr_2O_7$, known as chrompik). The waste contained key fission products including ^{134}Cs, ^{137}Cs, and ^{90}Sr, as well as transuranic elements. To address this, the VICHR (Vitrification for Chrompik) vitrification facility was established at the A-1 Nuclear Power Plant (NPP) in Jaslovské Bohunice, Slovakia. Designed with a melt capacity of approximately 2 kg/h, the plant produces waste glass stored in 7.5 l stainless steel canisters. A total of 18.8 m^3 of HLW has been successfully vitrified, yielding 211 canisters with a combined volume of 1.53 m^3 and specific activities ranging from 0.3 to 1.2 GBq/L [181], which are now safely stored at the NPP A-1 site.

5. Future challenges

Although vitrification is a mature and widely deployed technology for immobilizing HLW, several legacy wastes and emerging waste streams from advanced nuclear technologies continue to challenge existing glass formulations and conventional processing methods. These evolving waste chemistries necessitate the development of new waste forms, advanced processing technologies, and improved long-term performance assessment frameworks.

One of the most significant future challenges arises from the anticipated deployment of Generation IV advanced reactors (ARs). These systems employ diverse moderators, coolants, fuels, and fuel cycle strategies, generating waste streams that differ substantially from those produced by conventional light water reactors [182,183]. In particular, molten salt reactors (MSRs), sodium-cooled fast reactors, and pyrochemical processing technologies may generate halide-rich, metallic, or chemically aggressive waste streams that are not readily compatible with traditional borosilicate glass systems. Furthermore, many AR concepts remain under active development, resulting in substantial uncertainties regarding the chemistry and characteristics of future waste inventories. While this uncertainty complicates waste form design, it also provides an opportunity to integrate waste management considerations early into reactor and fuel cycle development [184].

Halide-bearing waste streams, especially chloride- and fluoride-rich salts generated from MSRs and pyroprocessing of metallic fuels, are particularly difficult to immobilize in conventional oxide glasses due to their limited solubility and volatility during thermal treatment. To address these challenges, extensive research is underway on alternative glass systems including silicate [185–188], phosphate [189–191], and tellurite glasses [192]. In parallel, glass-ceramics (GCs), glass composite materials (GCMs), and glass-bonded ceramic waste forms are attracting increasing attention because they can accommodate higher concentrations of otherwise incompatible species such as halides, sulfates, and molybdates. Examples include hot-isostatically pressed (HIP) sodalite glass-ceramics for chloride salts [193,194] and fluorite-based GCs for fluoride salts [195]. These multiphase materials combine the process flexibility of glasses with the superior chemical durability and radionuclide partitioning capabilities of ceramics. This concept can be extended to glass-bonded ceramic waste forms (or glass composite materials, GCM), for example glass-bonded sodalite [83,196,197] where the glass constitutes an additional barrier for protection against groundwater corrosion and radionuclides dispersion within a future deep geological disposal facility. This type of immobilization barrier concept has been similarly examined for spent TRISO-coated high-temperature gas reactor fuel particles, where the feasibility of embedding these particles in a glass matrix for geological disposal has been considered [198,199].

The management of challenging legacy wastes also remains a critical issue. A recent collaborative effort between researchers in France and the United States revisited vitrification approaches for the calcined HLW inventory stored at Idaho [147]. Technologies such as cold crucible induction melters (CCIMs) and DEM&MELT In-Can Melter (ICM) systems were evaluated because of their ability to process higher-viscosity melts, glass-ceramics, and chemically aggressive waste streams at elevated temperatures. CCIM technology offers advantages including higher waste loadings and compatibility with alternative waste form chemistries, while ICM systems reduce volatilization losses and corrosive off-gas generation by carrying out vitrification directly within disposable canisters [147]. As a technology, CCIM is already mature, being currently deployed at La Hague for the vitrification of certain HLW. Orano and the CEA have developed the DEM&MELT ICM technology up to full-scale pilot demonstration. For the calcined HLW at Idaho, both ICM and CCIM require further demonstration, including pilot testing, to confirm process performance, e.g., waste loading and glass throughput reachable at industrial scale.

HIP processing occurs within completely sealed HIP canisters at high temperatures (typically 1000 – 1300 °C) and pressures (~30 MPa and 100 MPa) and remains a strong candidate for the calcined HLW at Idaho [180]. Benefits include flexibility in the waste form class produced (ceramic, glass, and advanced composite waste forms such as cermets and GCs are all feasible), high waste loadings, and prevention of volatilization and corrosive chemical emissions into the off-gas system. Australia's Nuclear Science and Technology Organisation (ANSTO) is currently commissioning a HIP facility for the immobilization of Australian radiopharmaceutical wastes within a glass waste form, and in doing so will demonstrate the industrial deployment of the technology for the first time.

Another major challenge concerns the immobilization of volatile radionuclides, particularly iodine. The International Atomic Energy Agency (IAEA) has identified iodine separation and immobilization as a critical research priority for advanced fuel cycles [200]. The long half-life of ^{129}I ($t_{1/2} = 1.57 \times 10^7$ years) combined with its mobility in the biosphere, makes its long-term management especially important [201]. However, iodine incorporation into glass is difficult because of its high volatility and low retention during melting [202]. Various borosilicate, silver phosphate, and lead phosphate glasses are under investigation, together with GC and GCM systems [203]. HIP processing again offers significant advantages because it suppresses iodine volatilization during consolidation [203,204].

Crystallization and phase separation during vitrification remain additional constraints limiting waste loading in homogeneous glass waste forms. The formation of radionuclide-rich secondary phases containing sulfates, molybdates, or halides can compromise chemical durability and processability [19,205]. In such cases, GCMs provide an attractive alternative because they intentionally incorporate multiple phases and can tolerate compositions incompatible with traditional borosilicate glasses [206,207].

Finally, ensuring the long-term performance of vitrified waste forms under geological disposal conditions remains one of the most fundamental scientific and engineering challenges [208,209]. HLW glass waste forms are expected to retain radionuclides for timescales extending over thousands to millions of years. Consequently, understanding glass corrosion and alteration processes under repository-relevant conditions is essential. Glass corrosion is inherently complex and depends strongly on glass composition, groundwater chemistry, canister corrosion products, microbial activity, and evolving repository environments [210–213]. Laboratory experiments, field investigations, and studies of archaeological and natural analogue glasses—including uranium- and thorium-bearing natural glasses—continue to provide critical insights into long-term alteration mechanisms and radionuclide retention behavior [214–216].

6. Conclusion

The global vitrification of HLW represents a mature yet evolving strategy for ensuring the long-term safety of radioactive materials generated from both defense and civilian nuclear activities. Over the past seven decades, significant advancements have been made in waste form design, melter technology, and the implementation of vitrification facilities across countries with diverse fuel cycle policies and reactor technologies. Borosilicate glasses remain the predominant choice for HLW immobilization due to their chemical durability, compositional flexibility, and thermal stability, while phosphate glasses have found niche applications, notably in Russia.

This article has highlighted how national strategies—shaped by defense legacies, energy policies, reprocessing capabilities, and geological disposal planning—have led to country-specific adaptations of glass compositions and vitrification technologies, such as Joule-heated ceramic melters (JHCM) and cold crucible induction melters (CCIM). Despite these differences, a shared commitment to safe interim storage and eventually deep geological disposal of vitrified HLW underscores a global consensus on containment as the foremost principle in radioactive waste management.

As the world enters a new phase of nuclear resurgence driven by decarbonization goals, the volume and complexity of HLW are expected to increase, especially from advanced reactor designs and closed fuel cycles. Future directions must therefore focus on: (i) enhancing waste loading to reduce volume and cost; (ii) tailoring glass formulations to accommodate challenging waste streams; (iii) deploying next-generation melters for corrosive or actinide-rich wastes; and (iv) accelerating international cooperation on repository development and long-term performance evaluation of vitrified waste forms.

In conclusion, while vitrification is a proven technology but must continuously evolve to meet the challenges posed by modern nuclear systems and dynamic policy landscapes. Continued research, cross-border collaboration, and investment in infrastructure will be critical to ensuring vitrification remains at the core of global HLW management for decades to come.

CRediT authorship contribution statement

Ashutosh Goel: Writing – review & editing, Writing – original draft, Visualization, Validation, Supervision, Resources, Project administration, Investigation, Funding acquisition, Formal analysis, Data curation, Conceptualization. **Rajan Saini:** Writing – review & editing, Writing – original draft. **Paul A. Bingham:** Writing – review & editing, Writing – original draft, Visualization. **Sophie Schuller:** Writing – review & editing, Writing – original draft, Visualization. **Kazuyoshi Uruga:** Writing – review & editing, Writing – original draft. **Kai Xu:** Writing – review & editing, Writing – original draft. **Jake W. Amoroso:** Writing – review & editing. **Haruko M. Wainwright:** Writing – review & editing, Writing – original draft. **Wooyong Um:** Writing – review & editing, Writing – original draft. **Daniel J. Gregg:** Writing – review & editing, Writing – original draft. **Michael I. Ojovan:** Writing – review & editing. **R. Raja Madhavan:** Writing – review & editing, Writing – original draft, Visualization. **John D. Vienna:** Writing – review & editing. **John S. McCloy:** Writing – review & editing. **Albert A. Kruger:** Writing – review & editing.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper. The author is an Editorial Board Member/Editor-in-Chief/Associate Editor/Guest Editor for this journal and was not involved in the editorial review or the decision to publish this article.

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