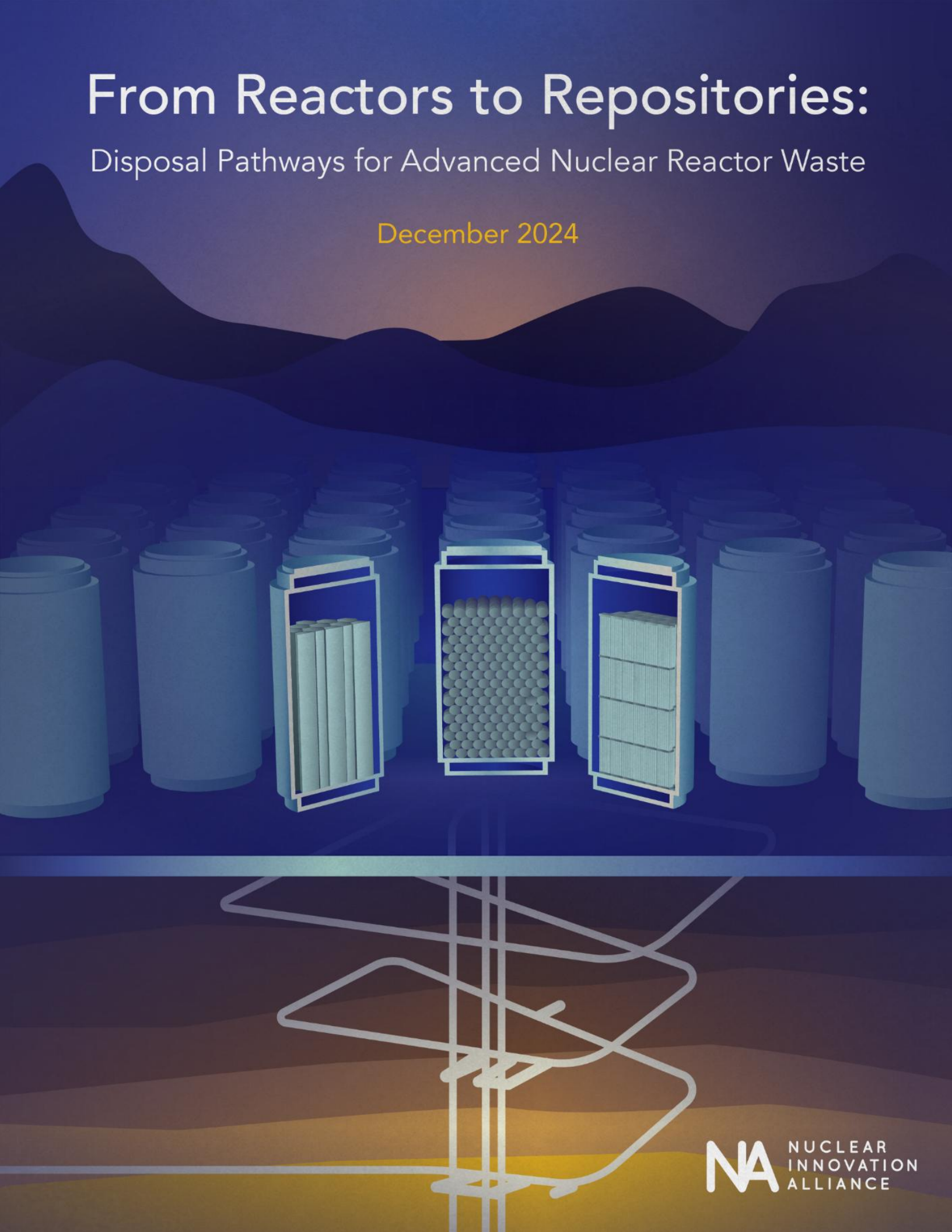
From Reactors to Repositories:

Disposal Pathways for Advanced Nuclear Reactor Waste

December 2024



NUCLEAR INNOVATION ALLIANCE

From Reactors to Repositories:

Disposal Pathways for Advanced Nuclear Reactor Waste

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Acronyms

ALWR	Advanced Light Water Reactor
AVR	German Arbeitsgemeinschaft Versuchsreaktor
BANR	BWXT's Advanced Nuclear Reactor
Ci	Curies
DOE	Department of Energy
EBR	Experimental Breeder Reactor
FLiBe	Fluoride-Lithium-Beryllium
FZJ	German Forschungszentrum Jülich Site
FSV	Fort Saint Vrain
FPC	Fuel Pool Cooling
FPP	Fuel Pool Purification
GW	Gigawatt
GTCC	Greater Than Class C
HLW	High-Level Waste
HTGR	High-Temperature Gas Reactor
INL	Idaho National Laboratory
IPyC	Inner Pyrolytic Carbon
IAEA	International Atomic Energy Agency
IUAT	Inter-Unit Access Tunnel
LLW	Low-Level Waste
MWe	Megawatts electric
MWth	Megawatts thermal
MTHM	Metric Tons of Heavy Metal
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
NRC	Nuclear Regulatory Commission
PIC	Pool Immersion Cell
SFR	Sodium-cooled Fast Reactor
SFC	Spent Fuel Canister
SFISF	Spent Fuel Intermediate Storage Facilities
SFP	Spent Fuel Pool
ISFSI	Independent Spent Fuel Storage Installation
SNF	Spent Nuclear Fuel
TRU	Transuranic
TRISO	TRI-structural ISOtropic
VLLW	Very Low-Level waste
WIPP	Waste Isolation Pilot Plant

Executive Summary

As the global demand for clean firm energy grows, advanced nuclear energy technologies are gaining significant attention and efforts to deploy them are underway. These advanced reactors offer numerous improvements in safety, efficiency, and operational flexibility largely due to the innovative fuel forms and coolants utilized in their design. New advanced reactor designs, however, will introduce new waste streams that may differ significantly from those generated by conventional reactors, depending on the specific reactor design. With new reactors come new waste forms.

Effective nuclear waste management is essential to the future of advanced nuclear energy and must be grounded in the best available information. Policymakers and stakeholders must therefore understand these new waste forms, their unique characteristics, and the specific management strategies needed for their safe storage and disposal.

This report characterizes the various waste streams that are generated by advanced nuclear reactors and examines both interim storage and permanent disposal pathways. Chapter 1 presents an introduction on the importance of understanding nuclear waste management to ensure successful deployment. Chapter 2 provides background information on nuclear waste, how nuclear waste is classified, and the potential disposal pathways for nuclear waste. Chapter 3 characterizes the various waste streams produced by advanced reactors. Chapter 4 discusses interim storage strategies being considered for advanced reactor wastes. Chapter 5 addresses permanent disposal pathways for advanced reactor wastes.

Four key takeaways from this report are as follows:

- **The characteristics of advanced reactor wastes will vary greatly.** The physical, chemical, and radiological properties of these wastes will depend on the specific reactor technology and company-specific design. *(see chapter 3)*
- **Interim storage strategies are robust, and the United States is well-prepared to ensure safe and effective management of advanced reactor wastes that require interim storage until permanent disposal solutions become available.** This is demonstrated by past real-world experience in managing spent nuclear fuel generated by legacy high-temperature gas reactors and sodium-cooled fast reactors, and current plans to manage spent nuclear fuel that will be produced from future advanced reactors. Interim storage will play a critical role in managing wastes generated by advanced reactors, like it does for conventional reactors, because the United States currently lacks a permanent repository. *(see chapter 4)*
- **Advanced reactor waste streams are expected to have little to no impact on the long-term safety performance of geological repositories**, **provided they are properly processed and packaged prior to disposal.** The safety performance of a geological

repository is dominated by engineered and geological barriers, rather than the characteristics of the waste.

- **Permanent disposal pathways for advanced reactor wastes are known to be technically feasible and are currently being explored.** The United States currently has no permanent disposal capability for "Greater than Class C" low-level waste or spent nuclear fuel generated by either advanced or conventional nuclear reactors. For advanced nuclear reactors, their unique spent nuclear fuel and Greater than Class C waste forms (excluding advanced light water reactors because of their similarity to conventional light water reactors) will require new waste management strategies to ensure they can be properly processed and packaged prior to permanent disposal. Developing these strategies may present challenges that vary based on reactor technology. However, solutions to these challenges are known to be technically feasible, and many efforts to develop these solutions have already begun. *(see chapter 5)*

1. Introduction

The global energy landscape is undergoing a significant transformation. As the world grapples with the challenges posed by climate change and energy security, the demand for clean firm energy sources has never been higher. Current projections suggest this demand is poised to grow even further, driven by escalating global energy needs and the urgent requirement for sustainable and dependable energy solutions.[1] With global energy consumption expected to rise substantially and rapidly in the coming decades, the pressure to find sustainable and reliable energy sources is rising.

In response to these challenges, policymakers worldwide are turning their attention to nuclear energy. The announcement made at COP28 in December 2023 by the USA, France, UK, and over a dozen other nations to triple global nuclear energy capacity by 2050 illustrates policymakers' increased interest in nuclear energy.[2]

The United States is currently working to commercialize a large number of new nuclear reactors. Advanced reactor developers are aiming to deploy a wide range of technologies by the end of the decade.[3] Congress and the President have recently enacted legislation to catalyze public-private partnerships and improve regulatory processes to accelerate new nuclear reactor deployment.[4] Public acceptance of nuclear energy is also increasing. Recent surveys indicate that support for nuclear energy in the United States is at its highest level in over a decade.[5]

Much of the increased support from both policymakers and the public is driven by the recognition of the numerous benefits that nuclear energy offers. These benefits include the creation of high-paying, highly skilled, long-lasting jobs, the reliable supply of year-round 24/7 power, and the fact that nuclear energy is one of the safest methods to generate electricity.[6]

Despite these advantages, concerns regarding nuclear waste management, disposal, and potential human health risks can be a barrier to the broader acceptance and deployment of new nuclear reactors. For example, state legislators in Colorado refused to even define nuclear energy as a source of clean energy in January 2024, citing nuclear waste concerns as a primary reason for their decision.[7]

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[1] U.S. Energy Information Administration | EIA projections indicate global energy consumption increases through 2050, outpacing efficiency gains and driving continued emissions growth
[2] Net Zero Nuclear | Pledge sets goal for tripling of nuclear energy by 2050
[3] Nuclear Innovation Alliance | Advanced Reactor Deployment Timelines
[4] ADVANCE Act; Prohibiting Russian Uranium Imports Act; Consolidated Appropriations Act of 2024
[5] Gallup | Americans' Support for Nuclear Energy Highest in a Decade
[6] Our World in Data | What are the safest and cleanest sources of energy?
[7] Colorado General Assembly | Senate Bill 24-039

When engaging with federal, state, and local policymakers, the question "what about the waste?", or some variation of such a question, is frequently raised. Whether stemming from general curiosity, mild concern, or significant fear, this question highlights a two-way gap in understanding between stakeholders and experts, and the need for informed conversations to articulate stakeholder concerns and share the best available information about what nuclear waste is and how it is safely managed. This gap is widened by the complex nature of nuclear waste management, which encompasses technical, environmental, political, and societal considerations.

Policymakers and public stakeholders need more comprehensive and accessible information about the real and perceived risks of handling and storing nuclear waste. They need a clear understanding of what nuclear waste is, how it is managed, how it can be safely stored, and the solutions available for its permanent disposal. This knowledge is crucial for making informed decisions about the deployment of advanced nuclear reactors. Without this information, policymakers may hesitate to embrace the benefits nuclear energy has to offer and fail to identify what policy solutions are needed to address nuclear waste issues.

The challenge of understanding nuclear waste management is complicated by the deployment of advanced nuclear reactor technologies. These advanced reactors offer numerous improvements in safety, efficiency, and operational flexibility, largely due to the innovative fuel forms and coolants utilized in their design. However, these new reactor designs will also produce waste streams that may differ significantly from those generated by conventional reactors. Policymakers, with the help of technical experts and other stakeholders, must be able to assess the United States' current ability to manage these new waste streams to identify gaps and potential solutions. This requires that policymakers comprehend the main characteristics of these advanced waste streams, how well prepared the United States is to store and dispose of them, and what is needed to build and manage robust, safe and effective disposal pathways.

This paper addresses key knowledge gaps regarding nuclear waste management, specifically in the context of advanced reactor technologies. It seeks to answer critical questions like: What will waste from advanced reactors look like? How will it differ from waste generated by existing light water reactors? How prepared are we to manage advanced reactor waste streams? And what areas require further investigation?

A comprehensive understanding of the answers to these questions is crucial to making informed decisions around advanced reactors and advanced reactor wastes. This more detailed understanding should give policymakers, and stakeholders who help inform policymakers, the knowledge they need to help create the conditions for success for advanced nuclear energy so that it can be part of a climate and energy security solution.

2. Understanding Nuclear Waste

To understand the technical characteristics of the waste streams produced by advanced reactors, it is helpful to first understand what nuclear waste is, how it is classified, and what the different disposal pathways for nuclear waste are. The following subsections will delve into these topics, providing the necessary context for a more detailed discussion on advanced reactor waste stream characterization, storage, and disposal.

2.1. What is Nuclear Waste?

Nuclear waste is the radioactive waste generated by the production or use of radioactive materials.[8] This includes nuclear waste generated by:

- commercial nuclear power production,
- defense-related activities,
- scientific research,
- medical activities, or
- mining or other industrial activities.

These materials contain unstable isotopes[9] that decay[10] over time in accordance with their respective half-lives,[11] and release hazardous high-energy radiation in the process. To protect public health and the environment, these hazardous materials must be isolated and carefully managed until they decay to a stable, safe state.

The focus of this report is on nuclear waste generated during commercial nuclear power production. Therefore, the use of the term "nuclear waste" in this report refers to civilian nuclear waste and not defense-related nuclear waste.

Nuclear waste encompasses a wide range of radioactive materials. Like any waste stream, it requires careful management to protect workers, the public, and the environment. Broadly speaking, nuclear waste contains the following:

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[8] U.S. Environmental Protection Agency | Radioactive Waste

[9] An isotope is an atom of the same element that has the same number of protons but a different number of neutrons in its nucleus. For example, carbon-12, carbon-13, and carbon-14 are isotopes of the element carbon, and therefore have the same number of protons but a different number of neutrons. The numbers in the isotopes name indicates the total number of protons and neutrons in the nucleus.

[10] Radioactive decay is the process by which an unstable isotope ejects high-energy particles (i.e., alpha, beta, or gamma radiation) from its nucleus to transform into a different isotope and eventually reach a stable state.

[11] Half-life is the time it takes one half of the atoms of a particular radioactive substance to decay into a more stable form. It represents the rate at which a radioactive substance undergoes radioactive decay. Each radioactive isotope has its own half-life, which can range from millionths of a second to billions of years.

- **Naturally Occurring Isotopes:** Isotopes found naturally on Earth that are radioactive, including materials such as uranium, thorium, and their naturally radioactive decay products. These primordial radionuclides were created before the formation of the Earth and have existed on Earth since its' creation. For example, uranium-235 and uranium-238 are the primary isotopes found in nuclear fuel and naturally decay to other radioactive isotopes (protactinium-231 and thorium-230, respectively) over time.
- **Transuranic Isotopes:** Isotopes of any element that are heavier (i.e., have a greater atomic number) than uranium. All transuranic isotopes, except for trace elements of specific neptunium and plutonium isotopes, are not found in nature and are only created on Earth through nuclear reactions.
- **Fission Products:** Isotopes created as a byproduct of nuclear fission. These are the smaller atoms created by the splitting of a single larger atom. A spectrum of different fission products is created during fission, and most fission products have very short or very long half-lives. Fission products can be highly radioactive and include isotopes such as cesium-137 and strontium-90.
- **Activated Materials:** Isotopes created by the exposure of materials to neutron radiation. Materials within a nuclear reactor can become radioactive through the process of neutron activation, which occurs when a stable isotope absorbs a neutron during reactor operation, causing it to become radioactive. For example, when structural steel components in a nuclear reactor core are bombarded by neutrons, they can become radioactive due to the formation of isotopes such as cobalt-60 from the neutron activation of colbalt-59.
- **Contaminated Materials:** Radiological contamination can occur when non-radioactive materials such as gloves, components, or mixtures come into contact with radioactive substances, resulting in the deposition of radioactive isotopes on their surfaces or in the material, thereby contaminating them. These components are not themselves radioactive but now contain radioactive materials.
- **Spent Nuclear Fuel (SNF):** Nuclear fuel that has been withdrawn from a nuclear reactor following irradiation. SNF contains a complex mixture of radioactive isotopes, including uranium, fission products, and transuranic isotopes all produced during operation of a nuclear reactor.

These nuclear waste materials can be further classified according to their origin and their potential hazards, as discussed in the next subsection.

2.2. Nuclear Waste Classifications

Nuclear waste is classified into different regulatory categories and subcategories based on the waste's origin, isotopic composition, radiotoxicity level, and concentration. These different nuclear waste classifications were designed to enable effective management and disposal of different types of wastes based on a general characterization of their hazards. Classifying

nuclear waste enables selection of appropriate handling, storage, and disposal methods to ensure safety and regulatory compliance for a specific classification of nuclear waste.

In the United States, the two broad categories for nuclear waste are high-level waste (HLW) and low-level waste (LLW). Other countries may have additional nuclear waste categories, such as intermediate level waste (ILW), however his report will use the current U.S. terminology.

In U.S. statute, definitions for HLW and LLW were established in the Nuclear Waste Policy Act of 1982 and can be found in 42 U.S.C. § 10101.[12] They are as follows:

- **HLW:** "The term HLW means - (A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation."
- **LLW:** "The term LLW means radioactive material that – (A) is not high-level radioactive waste, spent nuclear fuel, transuranic waste, or by-product material as defined in section 2014(e)(2) of this title; and (B) the Commission, consistent with existing law, classifies as low-level radioactive waste."

Given the deference in these definitions to the NRC to further define what constitutes LLW and HLW, the NRC provides their own definition of HLW and LLW. These definitions are discussed in the following subsections.

2.2.1. High-Level Waste

The NRC specifies in 10 CFR 60.2 that HLW includes SNF and liquid and solid waste streams that result from reprocessing[13] SNF in. The exact definition of HLW is as follows:

> "(1) Irradiated reactor fuel; (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and; (3) solids into which such liquid wastes have been converted."[14]

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[12] 42 U.S.C. § 10101

[13] Reprocessing involves the chemical separation of fission products, uranium, plutonium, and other elements within SNF to extract the uranium and plutonium for use in recycled fuel. Reprocessing typically involves dissolving the spent fuel in an acidic solution, then using chemical processes to separate the uranium, plutonium, and other fission products. Note that the term "recycling" is often used synonymously with reprocessing, however recycling generally refers to the entire process of reusing fuel material in SNF (including the fabrication of recycled fuel), in addition to reprocessing which is the chemical separation process.

[14] 10 CFR 60.2

Federal regulation defines SNF in 10 CFR 72.3 as:

> "Fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year's decay since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies."[15]

There are no commercial SNF reprocessing facilities in the United States, so SNF constitutes the majority of U.S. HLW.[16] Reprocessing SNF is a potential pathway to reduce the total volume of SNF that requires permanent disposal and recover valuable unutilized fuel material for future use. Certain factors limit the economic feasibility of establishing commercial reprocessing capabilities in the United States, including the relatively cheap cost to mine uranium. However, various programs have been established to evaluate the feasibility of SNF recycling in the United States. ARPA-E, a research agency within the U.S. Department of Energy, has funded 12 reprocessing-related research projects through its "CURIE" program to research novel reprocessing technologies and 11 projects through its "ONWARDS" program to optimize the waste streams from new reactor designs.[17] The private sector, including the advanced reactor developer Oklo, is also taking steps to develop reprocessing capabilities.[18] This report will not focus on recycling SNF, but this is a topic (including what policies are needed to incentivize reprocessing) for future investigation.

2.2.2. Low-Level Waste

The NRC broadly defines LLW as:

> "A general term for a wide range of items that have become contaminated with radioactive material or have become radioactive through exposure to neutron radiation. A variety of industries, hospitals and medical institutions, educational and research institutions, private or government laboratories, and nuclear fuel cycle facilities generate LLW as part of their day-to-day use of radioactive materials…The radioactivity in these wastes can range from just above natural background levels to much higher levels, such as seen in parts from inside the reactor vessel in a nuclear power plant. Low-level waste is typically stored onsite by licensees, either until it has decayed away and can be disposed of as

[15] 10 CFR 70.3

[16] The United States has reprocessed SNF in the past in small quantities. For example, the West Valley Demonstration Project in New York reprocessed SNF from 1963 to 1972, but operations were ceased due to high costs, environmental concerns, and proliferation concerns. In 1977, President Jimmy Carter issued an executive order halting commercial reprocessing efforts in the United States, mainly due to nuclear proliferation concerns.

[17] Good Energy Collective | FAQ: Recycling Nuclear Waste

[18] Oklo | Press Release

> ordinary trash, or until the accumulated amount becomes large enough to warrant shipment to a low-level waste disposal site."[19]

The classification of LLW considers both the concentration of long-lived radionuclides,[20] which pose a persistent hazard over extended periods, and short-lived radionuclides,[21] which can be actively managed until they no longer present a significant hazard.[22] To implement these dual considerations, the NRC further subclassifies LLW into several subcategories based on the concentration of different radionuclides present in the waste.[23] These subclasses include Class A, Class B, Class C, and Greater than Class C (GTCC) waste, which categorize waste by its concentrations of radioactivity, as shown in Figure *1* below. Very low-level waste (VLLW) is also a subcategory of Class A waste, but it has no formal statutory or regulatory definition.[24]

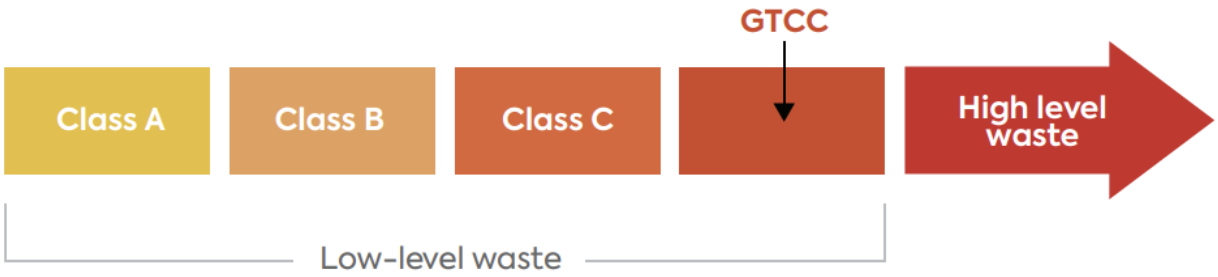

Figure 1: Low-level nuclear waste and high-level nuclear waste from left to right in increasing radiological concentration[25]

For nuclear waste that only contains long-lived radionuclides, Table *1* below (which was derived from 10 CFR Part 61.55) can be used to determine its LLW classification.

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[19] U.S. Nuclear Regulatory Commission | Low-level waste

[20] The term "long-lived", while generally referring to the half-life of the radionuclide, is a loose term with no firm definition. The half-lives of the long-lived radionuclides presented in Table 1 range between roughly 14 years and 16 billion years. Therefore, for the purpose of this report, "long-lived" radionuclides mean the ones presented in this table, consistent with how the term is used in 10 CFR 61.55.

[21] The term "short-lived", while generally referring to the half-life of the radionuclide, is a loose term with no firm definition. The half-lives of the short-lived radionuclides presented in Table 2 range between roughly 5 years and 100 years. Therefore, for the purpose of this report, "short-lived" radionuclides mean the ones presented in this table, consistent with how the term is used in 10 CFR 61.55.

[22] 10 CFR § 61.55

[23] 10 CFR § 61.55; The concentration thresholds for various isotopes differ because each isotope has a different half-life and emits different energies of ionizing radiation.

[24] "In general, VLLW contains some residual radioactivity, including naturally occurring radionuclides, which may be safely disposed of in hazardous or municipal solid waste landfills." (Source: U.S. Nuclear Regulatory Commission | Very Low-Level Waste)

[25] Bowen et al. (2024). "Revisiting GTCC and GTCC-Like Nuclear Waste Disposal in the United States".

Isotope	Concentration (Curies per meter cubed, Ci/m^3)		
	Class A	Class C	GTCC
C-14	≤ 0.8	> 0.8 and ≤ 8	> 8
C-14[1]	≤ 8	> 8 and ≤ 80	> 80
Ni-59[1]	≤ 22	> 22 and ≤ 220	> 220
Nb-94[1]	≤ 0.02	> 0.02 and ≤ 0.2	> 0.2
Tc-99	≤ 0.3	> 0.3 and ≤ 3	> 3
I-129	≤ 0.008	> 0.008 and ≤ 0.08	> 0.08
Alpha emitting transuranics[2]	≤ 10^3	> 10 and ≤ 100^3	> 100^3
Pu-241	≤ 350^3	> 350 and ≤ $3{,}500^3$	> $3{,}500^3$
Cm-242	≤ $2{,}000^3$	> 2,000 and ≤ $20{,}000^3$	> $20{,}000^3$

[1] Indicates the isotope is in an activated metal
[2] Alpha emitting transuranic radionuclides with half-life greater than 5 years
[3] Units are in nanocuries per gram (nCi/g)

Table 1: LLW classification for materials with long-lived radionuclides

For nuclear waste that only contains short-lived radionuclides, Table *2* below (which is derived from 10 CFR Part 61.55) can be used to determine its LLW classification.

Isotope	Concentration (Ci/m^3)			
	Class A	Class B	Class C	CTCC
Total of all radionuclides[1]	≤ 700	> 700	n/a	n/a
H-3	≤ 40	> 40	n/a	n/a
Co-60	≤ 700	> 700	n/a	n/a
Ni-63	≤ 3.5	> 3.5 and ≤ 70	> 70 and < 700	≤ 700
Ni-63[2]	≤ 3 5	> 35 and ≤ 700	> 700 and < 7,000	≤ 7,000
Sr-90	≤ 0.04	> 0.04 and ≤ 150	> 150 and < 7,000	≤ 7,000
Cs-137	≤ 1	> 1 and ≤ 44	> 44 and < 4,600	≤ 4600

[1] Total of all radionuclides with less than a half-life less than 5 years
[2] Indicates the isotope is in an activated metal

Table 2: LLW classification for materials with short-lived radionuclides

For LLW that contains multiple long-lived or short-lived isotopes that each fall within a different class, or a mixture of both long-lived and short-lived isotopes, see Appendix A. It should be noted that the majority of LLW by volume is Class A waste, and it represents nearly 90% of all LLW generated.[26]

Examples of waste materials for each LLW classification can vary, given they will depend on the materials concentration of radionuclides, as discussed above. It is challenging to make prescriptive determinations that a specific waste form material will always fit within a certain classification. For example, it is impossible to classify all contaminated gloves as Class A LLW, but it is possible to generally assess the LLW expected to be produced by reactor operation.

Examples of LLW produced during reactor operation can include:

- **Protective Clothing and Equipment:** Items such as gloves, coveralls, and shoe covers used by workers in nuclear power plants can become contaminated with low levels of radioactive material during maintenance, routine inspections, and other operational activities.
- **Filters:** Filters used to clean air and water systems in nuclear reactors can accumulate radioactive isotopes.
- **Tools and Instruments:** Tools and equipment used in reactor maintenance, such as wrenches, gauges, and monitors, can become contaminated with radioactive materials. While some tools and equipment can be decontaminated for continued use, some tools or equipment may be too challenging or costly to decontaminate.
- **Materials from Decommissioning:** During the decommissioning of nuclear power plants, plant systems, structures, and components are removed from the site. Materials including concrete, piping, wiring, and metal structures may become contaminated with radioactive material or activated neutrons. These materials will be managed based on their specific LLW classification.

2.3. Nuclear Waste Disposal Pathways

A disposal pathway for nuclear waste encompasses the entire process of managing and ultimately disposing of radioactive waste according to its specific waste classification. Various waste management strategies can be utilized to dispose of nuclear waste, and they generally fall within two broad categories: (1) Strategies designed to isolate nuclear waste with relatively short half-lives from the public and environment until that waste has decayed enough to reach safe levels; and (2) Strategies used to dispose of nuclear waste with relatively long half-lives indefinitely such that it is permanently isolated.

[26] U.S. Nuclear Regulatory Commission | Very Low Level Waste

These waste management strategies require facilities that are designed to either temporarily house nuclear waste while it decays, or permanently dispose of nuclear waste. In the United States, these facilities are known as either geological repositories or land disposal facilities.

- A **Geologic Repository** is defined as a system that may be used for the disposal of radioactive wastes in excavated geologic media.[27]
- A **Land Disposal Facility** is defined as the land, building, and structures, and equipment that are intended to be used for the disposal of radioactive wastes, excluding geologic repositories. [28]
    - A **Near-Surface Disposal Facility** is a Land Disposal Facility in which radioactive waste is disposed of in or within the upper 30 meters of the earth's surface.

These U.S. definitions are very broad and provide little clarity into the kinds of facilities that can utilized. It is helpful to look towards the international community's definitions of disposal facilities to gain a clearer understanding of the wide range of disposal methods that exist.

The International Atomic Energy Association provides the following definitions for nuclear waste disposal facilities, and groups them based on nuclear waste classification:

- **Deep geological disposal** involves burying HLW in stable geological formations deep underground. This method isolates the waste for thousands to millions of years and is primarily used for HLW and SNF due to their high radioactivity and long half-lives. The depth and geological stability of the disposal site provide a robust barrier against radiation leakage. The Onkalo repository in Finland is an example of a deep geological disposal site designed for long-term isolation of HLW.
- **Intermediate depth disposal** involves placing waste in facilities that are tens to hundreds of meters below the surface, often in concrete or engineered structures within stable geological formations. This method is generally suitable for intermediate waste and some TRU waste.
- **Near-surface disposal** involves placing LLW in shallow, engineered facilities near the ground surface. These facilities often use multiple barriers to prevent the release of radioactivity. This method is used for LLW, which includes items like contaminated clothing, tools, and reactor components with relatively low radioactivity and shorter half-lives.
- **Landfill disposal** involves placing VLLW in landfills, similar to those used for municipal waste but with additional controls to prevent environmental contamination. This method is suitable for VLLW, which includes materials with minimal radioactive contamination that pose a very low risk to the public and the environment. Certain

[27] 10 CFR 60.2
[28] 10 CFR 60.2

decommissioned materials from nuclear sites, such as lightly contaminated building debris, may be disposed of in specially designated landfills.

- **Decay storage** involves storing waste with short half-lives until its radioactivity has decayed to safe levels. This temporary storage allows the waste to become non-hazardous over time. This method is used for waste that contains short-lived isotopes, which can decay to safe levels within a few years to decades.

These disposal methods can be viewed graphically based on the waste materials radioactivity and half-life in Figure *2* below.

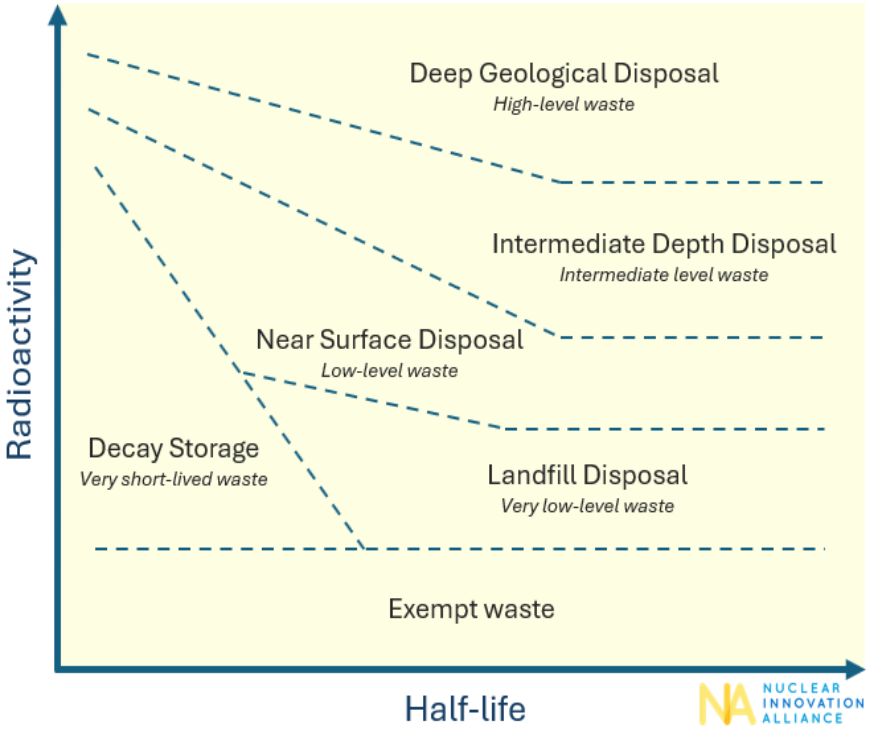

Figure 2: Disposal methods for various classifications of nuclear waste[29]

These next few subsections will provide greater detail on the current U.S. waste management strategies that are used for Class A, B, and C LLW, GTCC waste, and HLW.

2.3.1. Class A, B and C Low-Level Waste

Class A, B, and C LLW is typically stored on-site where it is generated, either until it has decayed and can be disposed of as non-nuclear waste or until it is shipped to a LLW disposal site.[30] As of 2024, there are four LLW disposal sites in the United States, as shown in the Figure *3* below. Each site is licensed to accept certain classifications of LLW (based on a site-specific license) and these sites are responsible for managing and storing Class A, B, and C

[29] Recreated based on an IAEA image: International Atomic Energy Association | No. GSG-1
[30] U.S. Nuclear Regulatory Commission | Low-Level Waste

LLW inventories across the United States until they have decayed to levels that are no longer harmful to the public of environment.

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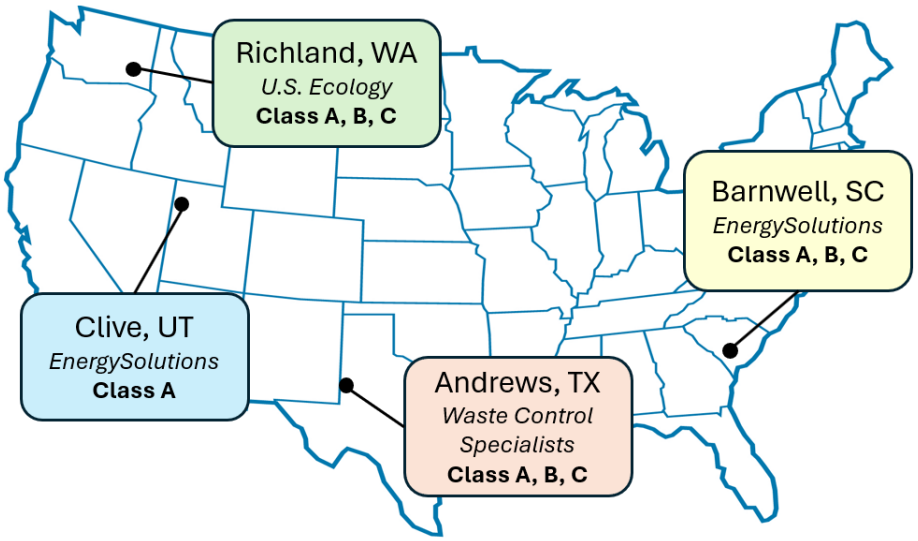

Figure 3: LLW Disposal Sites in the United States

The management and disposal of Class A, B, and C nuclear waste is generally less complex than that of HLW and GTCC waste due to its lower radioactivity and shorter required containment periods. Therefore, advanced reactor Class A, B, and C waste should be relatively easily integrated into existing LLW disposal pathways, even if it is different from the Class A, B, and C generated by existing reactors.

Management and disposal of HLW and GTCC waste from both advanced and conventional reactors, however, require more long-term oversight, planning, and resources for monitoring. Therefore, the remainder of this report will focus mainly on the management and disposal of HLW and GTCC waste associated with advanced reactor operation and decommissioning.

2.3.2. GTCC Waste

The United States currently has no disposal capability for GTCC nuclear waste generated by conventional or advanced reactors. Most GTCC waste is currently stored onsite at the nuclear power plant where it was generated. Since the largest source of GTCC nuclear waste comes from nuclear reactors reaching the end of their operating lifetimes, this GTCC inventory is generated by the reactor decommissioning process, and it resides in temporary (i.e., interim) storage onsite at shutdown nuclear power reactors that have ceased operations.[31]

[31] Bowen et al. (2024). "Revisiting GTCC and GTCC-Like Nuclear Waste Disposal in the United States".

In 2016, DOE published a final environmental impact statement on potential disposal options for GTCC waste that identified several approaches that could be taken for GTCC permanent disposal, including: above-grade vaults, enhanced near-surface trenches, intermediate depth boreholes, and a deep geologic repository at the Waste Isolation Pilot Plant (WIPP)[32] in New Mexico.[33] DOE also evaluated several sites to host a GTCC waste disposal facility and identified the LLW disposal facility in Texas and the WIPP as preferred candidates. Figures 4 and 5 below show the different disposal methods and sites that were evaluated.

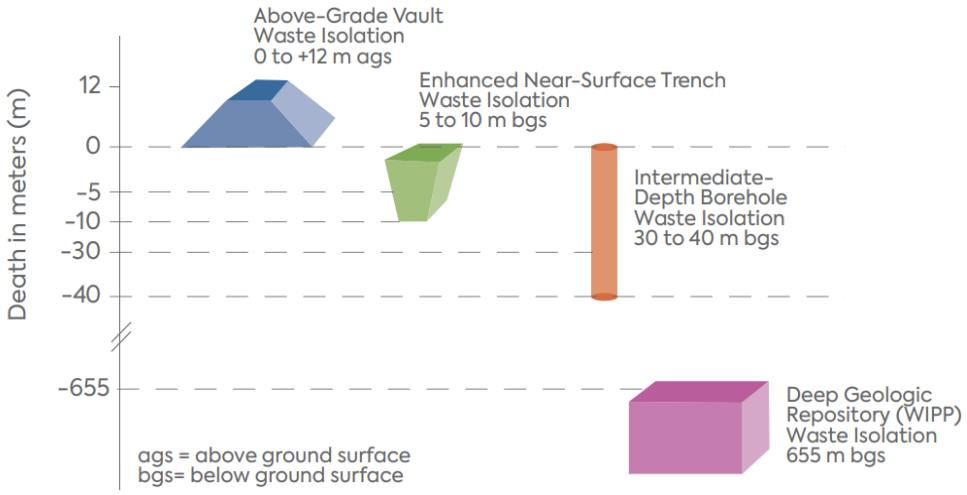

Figure 4: DOE illustration of waste isolation depths for proposed GTCC waste disposal methods[34]

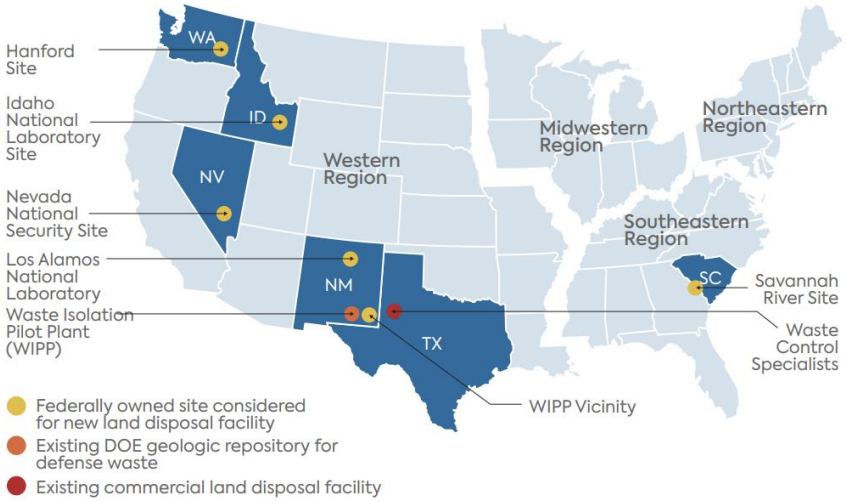

Figure 5: Locations the DOE evaluated for GTCC nuclear waste disposal[35]

[32] WIPP is a deep geological repository located in New Mexico designed to safely store defense-related transuranic (TRU) waste.
[33] U.S. Department of Energy | EIS-0375, Final Environmental Impact Statement
[34] Bowen et al. (2024). "Revisiting GTCC and GTCC-Like Nuclear Waste Disposal in the United States".
[35] Bowen et al. (2024). "Revisiting GTCC and GTCC-Like Nuclear Waste Disposal in the United States".

In May 2024, the NRC issued a proposed rule that would authorize the near-surface disposal of certain GTCC waste which is a major step towards constructing a GTCC disposal facility.[36] However, political opposition to constructing a facility, particularly from the governors of Texas and New Mexico, remains strong.[37]

2.3.3. High-Level Waste

The United States has a long history of managing HLW, which includes SNF and HLW generated from reprocessing SNF. SNF from conventional light water reactors (LWRs) consists of spent fuel pellets that are found within the reactor fuel assemblies, as shown in Figure 6 below.

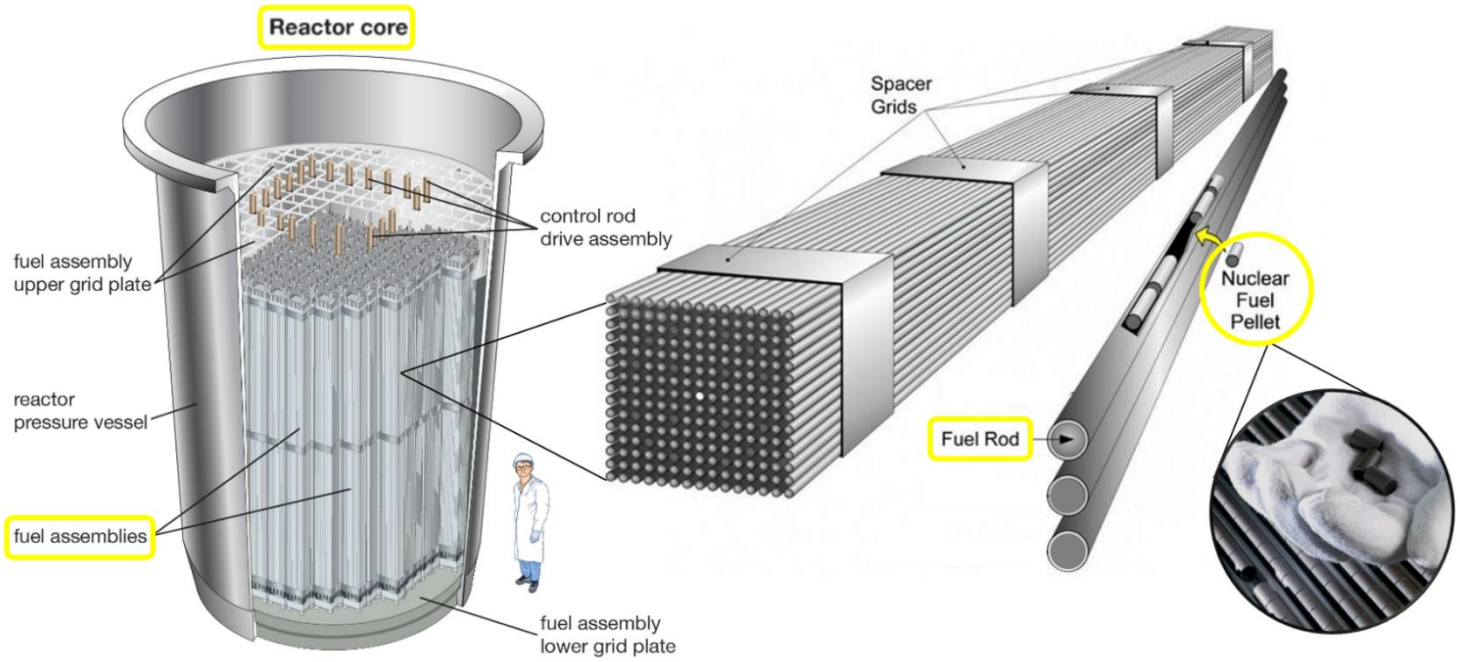

Figure 6: Diagram of a reactor core, fuel assemblies, fuel rods, and fuel pellets[38]

Once SNF assemblies are removed from the reactor, they are placed into "wet storage". This involves storing fuel assemblies in spent fuel pools filled with water, which acts as a coolant

[36] U.S. Nuclear Regulatory Commission | SECY-24-0045: Proposed Rule - Integrated Low-Level Radioactive Waste Disposal

[37] Bowen et al. (2024). "Revisiting GTCC and GTCC-Like Nuclear Waste Disposal in the United States".

[38] Image derived from the following sources: Deep Isolation | What is spent nuclear fuel?; Nuclear Regulatory Commission | Fuel Pellet; Britannica | Thermal, Intermediate, and Fast Reactors

and radiation shield, allowing the SNF to cool down and reduce its radioactivity as highly radioactive fission products decay over time.

After several years of cooling, the SNF can be transferred to "dry cask storage". In dry cask storage, the fuel assemblies are encapsulated in robust, corrosion-resistant dry casks. These casks provide external shielding from radiation and protect the SNF from external hazards, enabling the safe and secure storage for extended periods. Figure *7* below depicts wet and dry cask stroage.

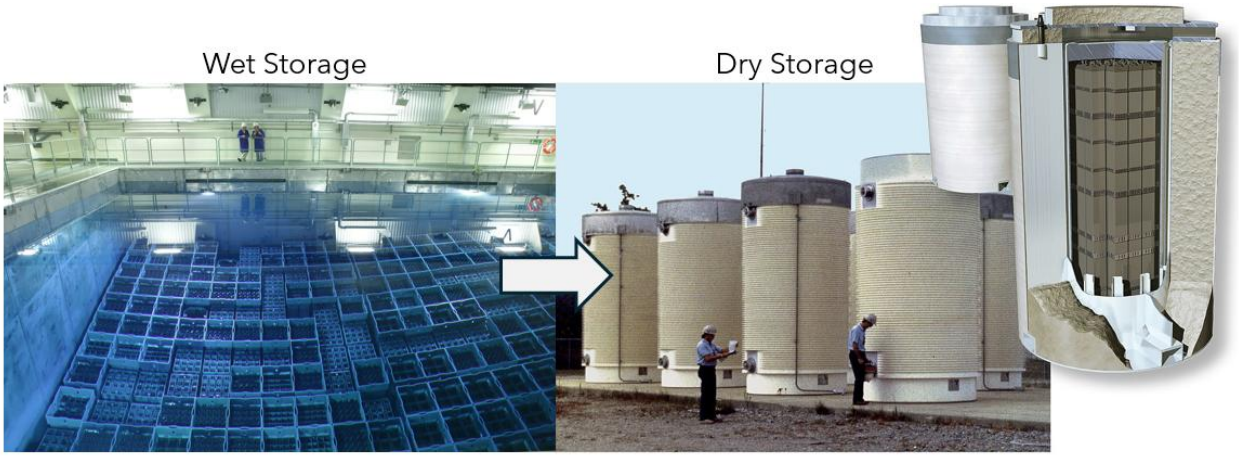

Figure 7: Photos of wet and dry cask storage of SNF[39]

As of 2021, the United States had a SNF inventory of roughly 89,000 metric tons of heavy metal (MTHM), stored across the country in 35 different states.[40] The amount of SNF stored in each state, and the sites within each state that store SNF, can be seen in Figure *8* below:

[39] Stimson | Geological Disposal of Spent Nuclear Fuel; and U.S. Nuclear Regulatory Commission
[40] Peters et al. (2022). "Spent Nuclear Fuel and Reprocessing Waste Inventory"

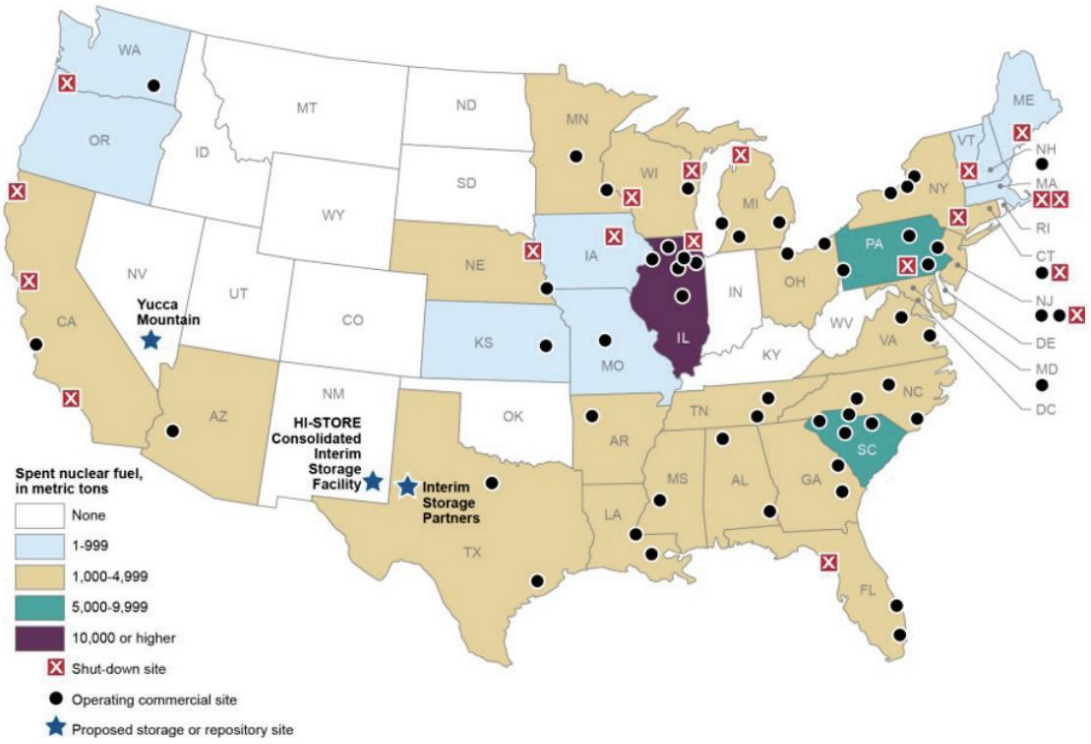

Figure 8: U.S. SNF Inventory[41]

While current onsite dry cask storage methods are safe and effective, such practices are only meant to be a temporary solution until a permanent disposal facility is constructed and operational. In the Nuclear Waste Policy Act of 1982, Congress gave the U.S. Department of Energy (DOE) statutory responsibility for taking SNF from commercial reactor sites and placing it into a permanent geological repository. However, DOE has yet to fulfill this responsibility.

The United States has explored mined deep geological repositories as a long-term solution for isolating SNF from the environment. Yucca Mountain in Nevada was selected in 1987 under the Nuclear Waste Policy Act as the nation's designated repository site due to its location and geological features. However, the project has faced substantial political and public opposition, particularly from Nevada residents and state officials who raised concerns about transportation risks, potential groundwater contamination, and the adequacy of Yucca Mountain's geology for long-term waste isolation.[42] As a result, the project has effectively been terminated, leaving the U.S. without a permanent repository for HLW and it is highly unlikely that Yucca Mountain will ever become operational given its long history of opposition and the current political climate.

[41] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 144
[42] Navada Attorney General | The Fight Against Yucca Mountain

HLW generated from reprocessing SNF also lacks a permanent disposal solution in the United States. This waste is produced from the chemical separation of reusable isotopes from SNF, resulting in a highly radioactive byproduct with long-lived radionuclides. HLW generated from reprocessing, like SNF itself, requires containment in a stable, isolated environment to prevent radioactive release over thousands of years, making a deep geological repository the most suitable disposal method. The total volume of this HLW is significantly less than SNF. A future deep geological repository for SNF would therefore also likely be used to dispose of HLW generated from reprocessing SNF.

While the development of a permanent U.S. deep geological repository has stalled, several innovative approaches are under consideration to address long-term nuclear waste management. One promising method is deep borehole technology, which involves drilling narrow, deep holes into stable geological formations to dispose of HLW. This technique, as shown in Figure 9 below, could provide an alternative permanent deep geological disposal method that has its own unique advantages.

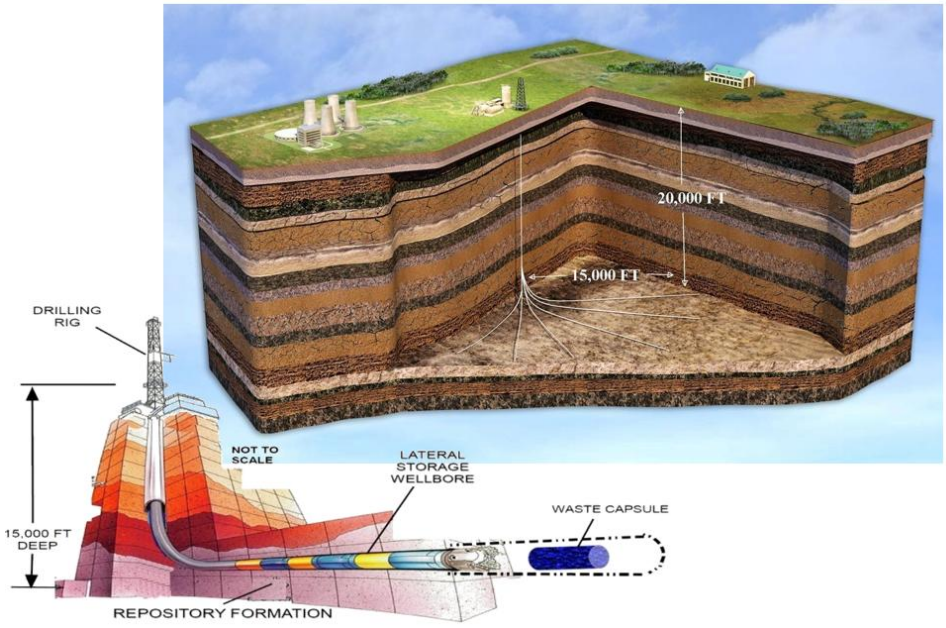

Figure 9: Deep Borehole Technologies for Permanent Disposal of SNF[43]

Certain challenges exist that complicate the use of deep boreholes. For example, the currently proposed diameter of these boreholes does not accommodate the casks currently used for dry storage, and repackaging spent nuclear fuel into new smaller diameter casks may be costly. Despite these challenges, deep boreholes remain an innovative solution and could prove to be a viable option if these challenges can be overcome.

[43] Crichlow. (2021). "Disposing of High-Level Nuclear Waste, Safer, Cheaper, Quicker and Retrievable."; and

Consent-based siting can also play a crucial role in the creation of nuclear waste storage and disposal facilities. Consent-based siting involves working closely with communities to site a new facility. This ensures that the community is informed about a potential facility and siting process, that they understand the risks and benefits of hosting a facility, and that they are able to provide consent to host the facility. This process not only builds public trust but also helps identify the most suitable sites for storage and disposal based on both geological suitability and community acceptance.

Countries like Finland and Sweden have made significant strides toward constructing and operating geological repositories using a similar approach to consent-based siting. Finland's Onkalo repository, shown in Figure 10 below, is expected to begin accepting spent nuclear fuel in 2025 or early 2026, making it the world's first operational permanent repository for nuclear waste.[44] Sweden has also issued an environmental permit for the construction of a deep geological repository and is hoping to start its 10 year construction period later this decade.[45]

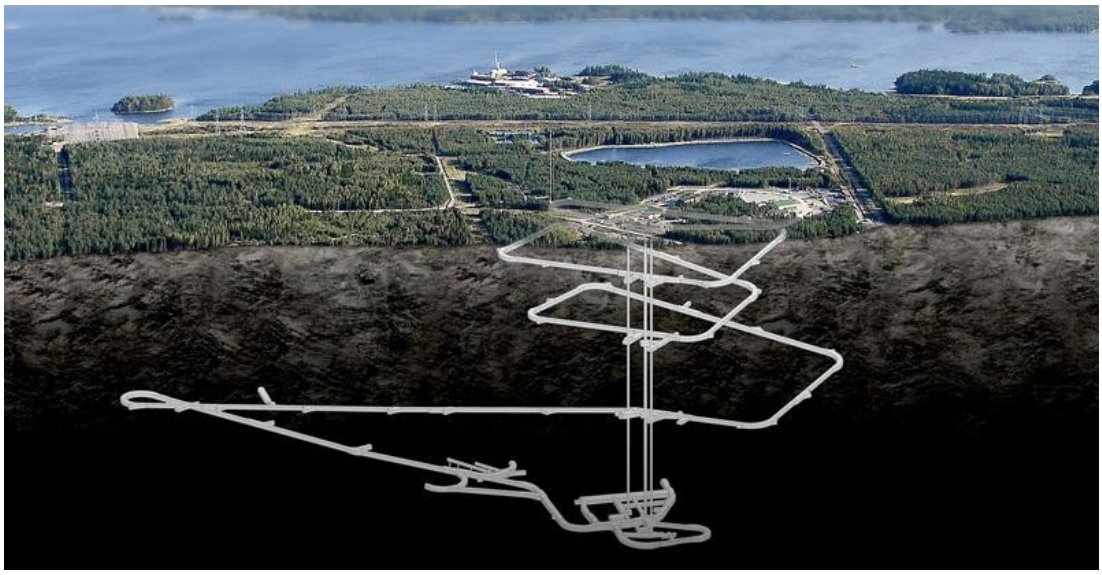

Figure 10: Finland's Onkalo deep geological repository[46]

DOE is adopting a consent-based siting approach to identify communities willing to host a consolidated interim storage facility.[47] Consolidated interim storage is a strategy that involves temporarily storing SNF at a centralized facility until a permanent disposal solution is established. This concept can be viewed in the rendering shown in Figure *11*. The primary benefit of this approach is that it simplifies the management of SNF, which is currently dispersed across various sites throughout the country, including decommissioned reactor

[44] CNBC | Finland will soon bury nuclear waste in a geological tomb that's built to last for 100,000 years
[45] World Nuclear News | Environmental permit granted for Swedish repository
[46] DEMM | Safe nuclear waste disposal aided by Kiwi technology
[47] U.S. Department of Energy | Consent-Based Siting

sites. By consolidating the waste in one location, it becomes easier to monitor, secure, and manage the material. Additionally, a consolidated interim storage facility can help bridge the gap between current storage practices and the development of a permanent repository. However, it is critical that consolidated interim storage does not become a de facto permanent storage solution, underscoring the need for continued progress toward establishing a permanent repository.

Figure 11: Rendering of a consolidated interim storage facility[48]

Despite innovative approaches to permanent disposal, it is clear that interim storage will continue to play a leading role in the near and mid-term management of U.S. HLW. This means that waste generated by advanced reactors, like waste from conventional reactors, must be safely managed in the interim, prior to final disposal while long-term efforts to develop a permanent repository move forward.

3. Characterizing Advanced Nuclear Reactor Waste Streams

A wide variety of advanced nuclear reactor technologies are currently being developed by private companies. These advanced reactors offer numerous improvements in safety, efficiency, and operational flexibility largely due to the innovative fuel forms and coolants utilized in their design. These new fuel forms and coolants offer unique benefits in comparison to those used in conventional LWRs, including greater fuel efficiency, increased thermal conductivity, more robust physical safety characteristics, enhanced stability at high temperatures, and inherent safety features that significantly reduce the risk of accidents.

[48] U.S. Department of Energy | Department of Energy Moves Forward with Consolidated Interim Storage Facility Project for Spent Nuclear Fuel

These new reactor designs will also generate new waste streams that may differ significantly from those generated by conventional reactors, depending on the specific reactor design. A comprehensive overview of each reactor's waste streams is needed to understand their specific waste characteristics and any unique management and disposal methods that are needed.

These advanced nuclear reactors will generate a wide range of HLW and LLW. As discussed in section 2.3.1, Class A, B, and C waste generated by advanced reactors should be relatively easily integrated into existing LLW management and disposal pathways. GTCC and SNF present greater challenges to waste management due to the lack of established disposal solutions. Therefore, this chapter will focus on SNF and certain reactor-specific GTCC waste that could be generated by five promising advanced reactor designs: the Advanced Light Water Reactor (ALWR), High-Temperature Gas Reactor (HTGR), Sodium-cooled Fast Reactor (SFR), liquid-fueled Molten Salt Reactor (MSR), and solid-fueled MSR.

Figure *12* below summarizes the various SNF forms that will be produced by these advanced reactor technologies.

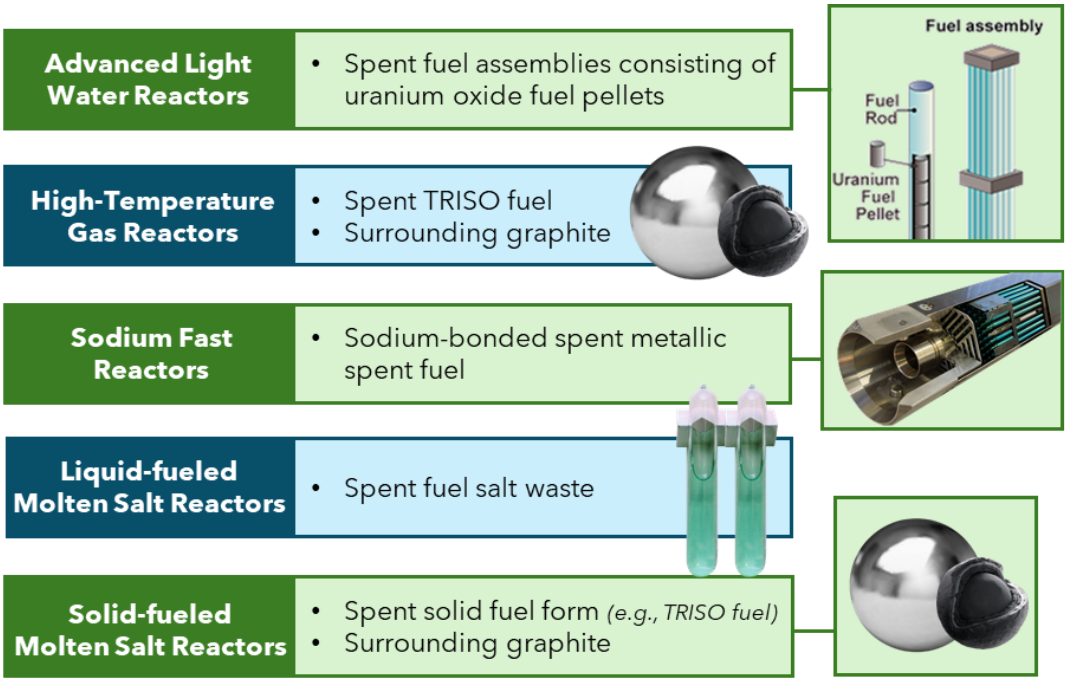

Figure 12: SNF generated by several advanced reactor technologies

3.1. Advanced Light Water Reactors

An ALWR is a light water-cooled advanced reactor design that implements design changes or operational modifications to improve performance compared with existing LWRs. The basic ALWR technology is similar to a conventional LWR since both use water as a coolant and moderator, along with uranium oxide fuel pellets as their fuel form (see Figure 6). However, ALWRs differ from conventional LWRs because they incorporate advanced safety features into their design, increase efficiency through higher enriched fuels or new fuel forms, and enable more flexible operation. Specific ALWR designs vary from company to company, but in general, ALWR advancements include passive and inherent safety systems that rely on natural forces such as gravity and natural convection of air for cooling, reducing the need for active mechanical systems.

Additionally, ALWRs typically feature a lower total power output, smaller footprint, and more compact and/or modular design in comparison to LWRs. They also have lower upfront capital costs than their much larger conventional LWR counterparts and offer greater flexibility in deployment. While both use standard LWR fuel assemblies, those used in ALWRs will be slightly shorter in length because of their smaller reactor pressure vessels.

Examples of ALWRs under development include NuScale's Voyagr, GE Hitachi's BWRX-300, Holtec's SMR-300, and Westinghouse's AP300.

Due to the general similarities between ALWR and LWR technologies, both will produce SNF assemblies consisting of uranium oxide fuel pellets (see Figure 6) nearly identical to existing reactors. Both technologies will also produce comparable spent nuclear fuel inventories on a per thermal power basis, and similar levels of fuel burnup (a measure of how much energy is extracted from nuclear fuel), as shown in Table *3* below.

Technology	Design	Power Output (MWe)[1]	Burnup (GWd/MTHM)[1]	SNF Inventory (MTHM/GWe-yr)[1]
Light Water Reactor	Pressurized Water Reactor	1000	50	22
Advanced Light Water Reactor	NuScale VOYGR	924[2]	41-60	20-29

[1] Megawatts electric, MWe | Gigawatt-days per metric ton of heavy metal fuel, GWd/MTHM | Metric tons of heavy metal per gigawatt electrical year, MTHM/GWe-yr
[2] Twelve 77MWe reactors in a single NuScale VOYGR power plant

Table 3: Example of SNF inventories from light water reactors and advanced light water reactors[49]

One challenge for ALWRs is that the smaller reactor core of the ALWR could result in greater levels of exposure to neutron radiation for certain reactor components as compared with

[49] Table was derived from information provided in the National Academies report: National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 158-159

conventional LWRs.[50] Different ALWR reactor components such as the reactor pressure vessel, baffles, and reflectors that are close to the reactor fuel could become activated to the point that they would be classified as GTCC waste. This would result in ALWRs producing more GTCC waste than conventional LWRs per unit of energy produced. However, ALWRs designs can also utilize optimized core power distributions[51] that reduce neutron leakage outside of the core into these components, effectively reducing the amount of GTCC waste that would otherwise be produced. NuScale has stated that their design has fewer components that can become GTCC waste from neutron activation compared with currently operating boiling water reactors and PWRs.[52] Therefore, despite having smaller reactor cores that in principle could lead to more GTCC waste, the use of innovative engineering and design principles in ALWRs can be used to reduce the amount of GTCC waste produced.

Overall, the characteristics of waste generated by ALWRs will be comparable to that produced by conventional LWRs. Both reactor types will produce similar waste streams, and while the specific quantities (masses and volumes) and specific levels of contamination or activation produced may differ, current methods for handling nuclear waste can be readily applied to ALWR waste. As a result, given the robust nature of existing LWR waste management practices, ALWR waste is not expected to introduce any significant new challenges with respect to the overall waste management strategy that is needed to safely and effectively prevent its exposure to the public and environment. Therefore, the interim and permanent disposal methods for this waste are expected to align with those already established for conventional LWR waste. Consequently, this report will not discuss new management or disposal methods for ALWR waste in Chapters 4 and 5.

3.2. High-Temperature Gas Reactors

HTGRs are graphite-moderated and gas-cooled reactors. Helium is typically used as the coolant and heat-transfer medium because it is an inert gas and therefore does not react with other materials or degrade components within the reactor. HTGRs use TRI-structural ISOtropic (TRISO) fuel particles as their fuel. These TRISO fuel particles, which are less than one millimeter in diameter, contain a uranium fuel "kernel" in their center and are surrounded by several layers of protective coatings. These layers, as shown in Figure *13*, include a porous carbon buffer, an inner pyrolytic carbon[53] layer, a silicon carbide barrier, and an outer pyrolytic carbon layer. Together, these layers provide exceptional structural integrity and

[50] Brown et al. (2017). "Impact of thermal spectrum small modular reactors on performance of once-through nuclear fuel cycles with low-enriched uranium".

[51] Core power distribution refers to the spatial variation of power generation within a nuclear reactor core. It is influenced by factors such as fuel composition, geometry, and neutron flux, and plays a critical role in reactor efficiency, stability, and safety management.

[52] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 159

[53] Pyrolytic carbon is carbon is a specific structure rather than a unique compound. It is similar to graphite, but includes some covalent bonds between its graphene sheets.

containment, allowing the TRISO particles to retain fission products and remain stable under the high temperature and radiation environments typical of HTGRs. This multi-layered design enhances the safety and durability of the fuel during both reactor operation and waste disposal. For a more detailed description of each layer, along with the unique purpose it serves, see Appendix B.

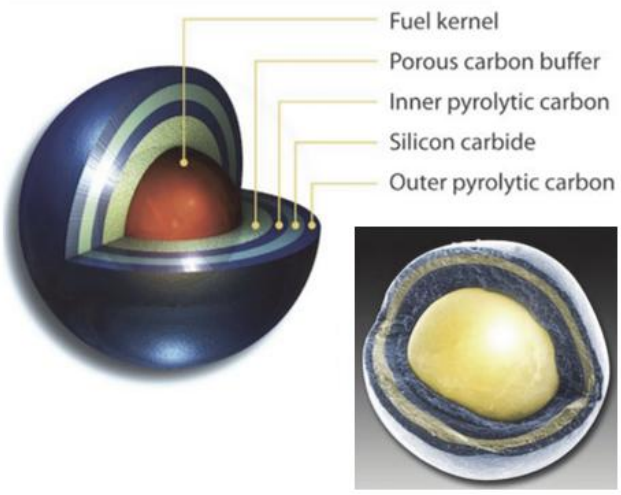

Figure 13: Illustration of a TRISO fuel particle (left); Scanning electron microscope image of a TRISO fuel particle with false coloring (right)

There are two types of HTGRs: pebble bed reactors and prismatic block reactors. In pebble bed reactors, the TRISO fuel particles are dispersed within graphite fuel pebbles, each roughly the size of a tennis ball. Each pebble contains approximately 18,000 TRISO fuel particles, and a single reactor core can contain over 200,000 fuel pebbles.[54] These pebbles are circulated through the reactor core, and the reactor operates continuously by adding fresh fuel pebbles and removing spent ones as they pass through the core.

In prismatic block reactors, the TRISO fuel particles are embedded in a carbon matrix,[55] forming composite cylindrical units known as fuel compacts (see Figure *14*). These fuel compacts (sometimes referred to as fuel sticks because of their shape) are then placed within hexagonal graphite blocks, known as prismatic block fuel elements. These graphite blocks are designed with fuel holes to house the fuel sticks and coolant channels, which are drilled into the graphite to allow the flow of coolant.

[54] X-energy | Xe-100 Technology Explainer

[55] Ultra Safe Nuclear Company are also exploring depositing their TRISO fuel particles in silicon carbide instead of graphite. (Source: USNC). However, for this report, graphite will be the assumed material of choice to simplify this discussion for the reader. The implications of graphite vs. silicon carbide as the material that surrounds the TRISO fuel particles, and it's impacts to nuclear waste disposal, is a topic for future investigation.

Despite having very different fuel designs, as shown in Figure *14* below, both types of HTGR are graphite-moderated, gas-cooled, thermal reactors that use many of the same materials. As a result, both prismatic block and pebble bed HTGRs produce similar waste streams.[56]

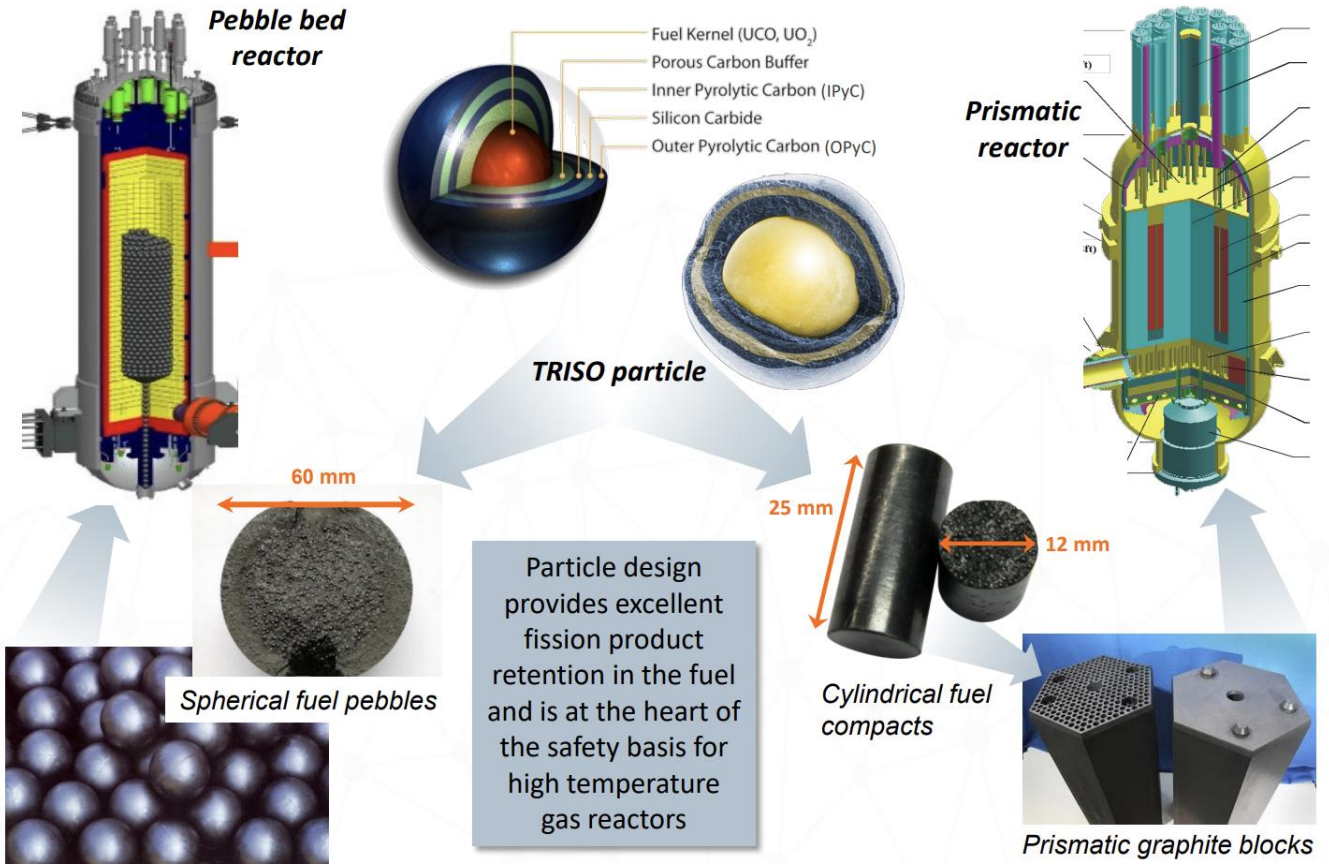

Figure 14: Depiction of both pebble beds and prismatic block HTGRs, and each reactors fuel form. [57]

Examples of HTGRs that use TRISO fuel pebbles and prismatic blocks include X-energy's Xe-100 reactor and BWXT's Advanced Nuclear Reactor (BANR), respectively.

In pebble bed reactors, the TRISO spent fuel consists of the spent fuel pebbles. In prismatic block reactors, the TRISO spent fuel consists of spent prismatic graphite blocks. The graphite surrounding the TRISO fuel particles will likely be treated as SNF because separating the fuel particles from the graphite is currently not a viable or economical practice (this is discussed

[56] Kitcher. (2020). "Disposition Options for a High-Temperature Gas-Cooled Reactor". *National Reactor Innovation Center*
[57] Demkowicz. (2019). "TRISO Fuel: Design, Manufacturing, and Performance". *Idaho National Laboratory*

further in Section Appendix C), in which case, the entire fuel pebble and prismatic block would be disposed of as SNF.

It should be noted that SNF from HTGRs provides a unique advantage for long-term disposal due to the self-containment of fission products within each TRISO fuel particle and the surrounding graphite structure. Each TRISO fuel particle acts as its own self-containment system, because of the multiple layers of carbon-based materials that surround the uranium kernel. This robust, multi-layered encapsulation effectively traps fission products within the fuel particle, significantly reducing the likelihood of radionuclide release, even under extreme conditions such as high temperatures or mechanical stress. This inherent containment capability can enhance the long-term safety of geological disposal by providing an additional barrier to radionuclide migration, helping to ensure the stability and security of the spent fuel over extended periods.[58]

HTGRs with TRISO fuel pebbles present a unique challenge that arises from the production of radioactive graphite dust. This dust is primarily generated by the friction between the graphite pebbles as they move against each other within the reactor core. The dust comprises fine particles of graphite and radionuclides, which pose additional complications related to waste management during decommissioning.

While the helium coolant used in pebble-bed reactors does not become radioactive itself, it can carry radioactive graphite dust particles throughout the reactor system. This contamination can affect various components, complicating their maintenance and potentially classifying them as higher-level radioactive waste. This contamination is particularly problematic when it can infiltrate pores on components within the reactor core, turning them into GTCC waste. The production of radioactive carbonaceous dust is estimated to range from 15 kg to 100 kg per year, depending on the specific HTGR design and operating conditions.[59]

The challenges of dealing with radioactive dust are greatest during decommissioning pebble-bed reactors. These challenges were seen when the German Arbeitsgemeinschaft Versuchsreaktor (AVR) pebble-bed test reactor, which operated from 1967 to 1988, was decommissioned. The accumulation of radioactive dust and the contamination of reactor components necessitated filling the entire core with concrete to stabilize the materials before transporting it to a storage facility.[60] This example highlights the complexity and cost that can occur because of the need to manage radioactive graphite dust during decommissioning. However, with the right decommissioning strategy and dust management practices (e.g.,

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[58] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 142

[59] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 160

[60] Moormann. 2008. "A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts"

filtration), the approach used to decommission the German AVR test reactor would likely not be necessary for future commercial HTGRs. It's also important to note that, as a test reactor, the AVR operated under conditions outside the normal parameters of commercial reactors, which can exacerbate dust management challenges, and are unlikely to occur in a commercial HTGR setting.

3.3. Sodium-Cooled Fast Reactors

SFRs are advanced nuclear reactors that utilize liquid sodium as a coolant to transfer heat from the reactor core to power generating systems. Unlike conventional reactors, which rely on water as a coolant and moderator, SFRs use liquid sodium due to its excellent heat transfer properties and its ability to remain in a liquid phase at high temperatures without boiling. SFRs operate using fast neutrons, meaning they do not have a neutron moderator in the core.

Examples of SFRs currently under development include TerraPower's Natrium reactor and ARC Clean Technology's ARC-100 reactor.

The fuel used in SFRs is a metallic fuel form consisting of uranium that is alloyed with various other metals and designed to withstand the high-temperature environments of fast reactors. These fuel forms include uranium-zirconium (U-Zr) and uranium-plutonium-zirconium (U-Pu-Zr) metallic fuel. In the reactor core, this fuel is encased in metal cladding, typically stainless steel, and packaged into fuel assemblies, similar to a LWR design. In contrast to LWR designs which operate with a small helium-filled gap between the fuel and the cladding, the solid fuel and fuel cladding are bonded by liquid sodium to improve heat transfer while accommodating the differences in thermal expansion between the metallic fuel and cladding material.

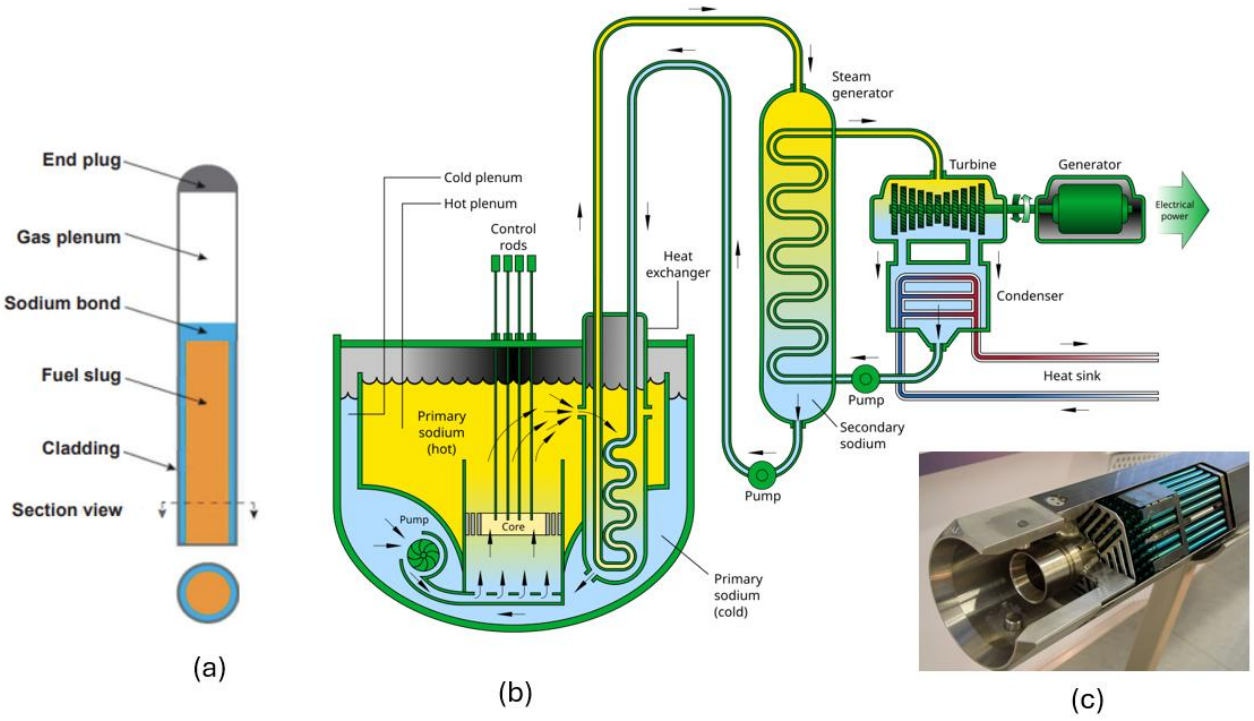

Figure 15: Diagram of a SFR fuel rod (a); Schematic of a SFR (b); Photo of a SFR fuel assembly (c)[61]

The SNF generated by SFRs is sodium-bonded spent metallic fuel. Sodium-bonded spent metallic fuel refers to the spent fuel that has been bonded with liquid sodium during reactor operation. During operation, fission gases such as xenon and krypton are generated and produce micro-pores inside the irradiated fuel matrix. At higher burnups, these micro-pores begin to connect, forming larger pores and pathways for fission gas to move upward to escape. As a result, liquid sodium can enter the interconnected pores within the fuel and fuse with the fuel.

This sodium-bonded spent metallic fuel creates challenges for disposal, as the sodium is highly reactive and cannot be easily separated from the fuel elements, necessitating specialized treatment to prevent chemical reactions and safely isolate the radioactive material (see chapters 4 and 5 for a more detailed discussion of disposal considerations). However, not all SFR designs will produce sodium-bonded spent metallic fuel. TerraPower's Natrium reactor will ultimately use an advanced metallic fuel, called Type 1B fuel, that does not use sodium to bond the fuel to the cladding and instead comes into direct contact with the fuel cladding. Although TerraPower plans to use this novel fuel form in the future, it is still under development, and therefore their early mover reactors will generate sodium-bonded spent metallic fuel.[62]

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[61] Image sources for each image: National Academies, sodium-cooled fast reactor wiki, jetseal.
[62] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 164

In addition to spent metallic fuel, the large quantity of sodium coolant used to transfer heat in the reactor must be disposed of. This "bulk sodium" contains several sources of radioactivity, including sodium activation products and contamination from actinides and fission products. The main radionuclides include sodium-22, tritium, cobalt-60, and cesium-137. In contrast to the sodium-bonded spent metallic fuel, bulk sodium will have a much lower level of radioactivity and will subsequently be classified as LLW. The specific LLW classification will vary based on reactor design, but it is unlikely that it will be classified as GTCC waste. While the focus of this report is on SNF and GTCC waste produced by advanced reactors, the relatively large volume of bulk sodium that will be produced from each reactor necessitates identifying it as a critical waste form that must be managed. The exact volume of bulk sodium produced will vary depending on the specific reactor design, but for example, one of TerraPower's Natrium reactors is expected to produce roughly 800m^3 of bulk sodium.[63]

Of note, bulk sodium must only be drained from the reactor and managed upon decommissioning, since it is designed to remain in the reactor throughout the reactor's entire lifetime, unlike SNF which will be generated on a periodic basis whenever the reactor is being refueled. It is possible that this bulk sodium could be reused, which would delay the need to properly dispose of it. However, the feasibility of this reuse is currently unclear and would depend on the development and deployment of future SFRs, as well as several economic and technical considerations.

3.4. Molten Salt Reactors

MSRs are advanced nuclear reactors that use molten salt as either the primary coolant or as both the primary coolant and fuel carrier. There are generally two types of MSRs: liquid-fueled and solid-fueled MSRs. Liquid-fueled MSRs dissolve their fuel directly into the molten salt coolant, allowing the fuel to flow through the reactor core. Solid-fueled MSRs utilize molten salt solely as a coolant and have a separate solid fuel form.

An example of a solid-fueled MSR is Kairos Power's KP-FHR reactor design that uses TRISO fuel pebbles as its fuel form. Examples of liquid-fueled MSR include TerraPower's Molten Chloride Fast Reactor and Terrestrial Energy's Integral Molten Salt Reactor.

Both liquid and solid-fueled reactors can use either a fluoride- or chloride-based salt. Most advanced reactor designs, however, favor using a fluoride-based salt, specifically a fluoride-lithium-beryllium (FLiBe) salt. At room temperature, FLiBe is a solid crystal, but it has a melting point below MSR operating temperatures. Therefore, FLiBe circulates the reactor as a liquid.

3.4.1. Liquid-Fueled Molten Salt Reactors

[63] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 163

In liquid-fueled MSRs, the molten salt is the spent nuclear fuel form because the uranium is dissolved directly into the FLiBe salt, as shown in Figure *16*.

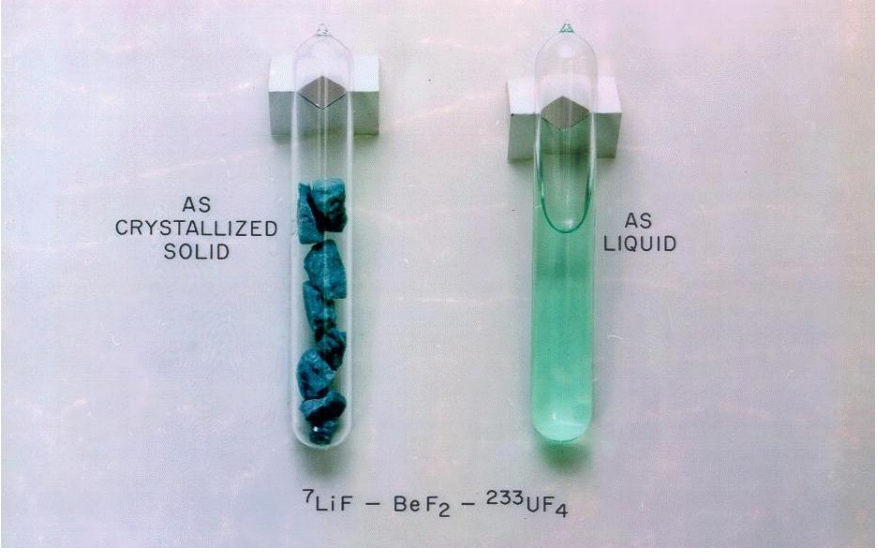

Figure 16: FLiBe salt with uranium hexafluoride

Since the nuclear fuel is circulated throughout the reactor during operation, the entire volume of the salt becomes highly radioactive. This continuous circulation means that fission products, actinides, and other radioactive isotopes are spread throughout the reactor's salt system. Unlike solid-fueled reactors, where spent fuel is localized as a solid material (e.g., pebbles, blocks, or assemblies) in the core, liquid-fueled MSRs waste management requires the handling of the entire volume of the fuel-salt mixture as SNF. Therefore, liquid-fueled MSRs require comprehensive strategies to handle the dispersed and highly radioactive spent fuel salt, complicating the overall waste management process. These complications have been seen during the decommissioning of the molten salt reactor experiment (MSRE), as discussed in Appendix D.

Liquid-fueled MSRs also produce an off-gas waste stream. Volatile radionuclides produced during fission are not confined to the fuel form and are released into the reactor, requiring an off-gas system to capture noble gas fission products (such as xenon and krypton), reactive gases (I_2, Cl_2, F_2), tritium, and other gases.[64] This capture is necessary to comply with EPA regulation 40 CFR 190, which governs fission gas releases across the uranium fuel cycle.[65]

3.4.2. Solid-Fueled Molten Salt Reactors

Solid-fueled MSRs can simplify waste management when compared to liquid-fueled MSRs because the nuclear fuel remains contained within a solid-fuel form, such as TRISO fuel

[64] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 166
[65] 10 CFR 190

pebbles, which are not dispersed throughout the reactor coolant system. The characteristics of solid-fueled MSR SNF will be similar to HTGR SNF (discussed in section 3.2) because they use the same fuel forms. Solid-fueled MSRs will also produce bulk sodium, as discussed in section 3.4.1.

The discrete fuel elements in solid-fueled MSRs make it easier to handle and remove spent fuel in contrast to liquid-fueled MSRs. Since the molten salt in solid-fueled MSRs is used only as a coolant, it is typically classified as LLW rather than HLW based on its radioactivity. This distinction reduces the complexity of waste disposal, as solid fuel retains most of the radioactive isotopes, making both the fuel and coolant easier to manage and dispose of. Additionally, TRISO fuel can trap off-gases, acting as an inherent containment system for fission products, further simplifying waste management.

4. Interim Storage

Interim storage[66] will play a critical role in the management of nuclear waste generated by advanced reactors because the United States currently lacks a permanent repository. Fortunately, the United States already has significant experience managing the SNF that will be generated by advanced reactors seeking to be deployed by the end of the decade.

SNF from ALWRs can utilize existing interim storage strategies, given it is nearly identical to the SNF generated by conventional LWRs. The United States and several other countries have also previously operated a total of five HTGRs and eleven SFRs, which have since ceased operation and undergone varying degrees of decommissioning involving the management of their SNF. Among these reactors, the United States has decommissioned two prismatic block HTGRs and five SFRs. Details regarding each of these reactors can be seen in Table *4* below.

[66] For the purposes of this report, "interim storage" generally refers to onsite temporary storage of HLW. This should not be confused with *consolidated* interim storage, which involves the transportation of HLW from multiple sites across the country to a single location. Current efforts at DOE to establish a consolidated interim storage facility using a consent-based siting process are separate from the onsite interim storage methods described in this report.

Reactor Type	Reactor	Country	Power, (MW_{th})	Years of Operation
HTGR Prismatic Block	Peach Bottom	United States	115	1966–1974
	Fort Saint Vrain	United States	842	1976–1989
	Dragon	England	21.5	1964–1975
HTGR Pebble Bed	THTR	Germany	750	1985–1991
	AVR	Germany	46	1967–1988
SFR	EBR-1	United States	1.4	1950–1964
	SRE	United States	20	1957–1964
	Fermi 1	United States	200	1963–1975
	EBR-2	United States	62.5	1965–1994
	FFTF	United States	400	1980– 2003
	Rapsodie	France	40	1967–1983
	Phénix	France	590	1973–2010
	Superphénix	France	3000	1986– 1997
	Monju	Japan	714	1995–2010
	BN-350	Soviet Union	~850	1973–1999
	PFR	United Kingdom	500	1974–1994

Table 4: List of decommissioned HTGRs and SFRs[67]

The majority of the SNF generated by these U.S. reactors is currently safely stored at the Idaho National Laboratory (INL), and the methods used to manage them vary. For example, sodium-bonded spent metallic fuel from EBR-II was initially placed in wet storage before being transferred to dry cask storage, while sodium-bonded spent metallic fuel from Fermi 1 was placed directly into dry cask storage. Additionally, the designs of the storage containers and the facilities used to house them differed. However, the interim storage methods used were robust, technically mature, and provided a means to safely store the waste generated by these reactors. Consequently, the experience gained handling the SNF generated by these HTGRs and SFRs provides a strong foundation for the future management of advanced reactor wastes.

While the United States has no direct experience with managing pebble-bed HTGR waste, both prismatic block and pebble bed HTGRs generate similar waste streams. As discussed in section 3.2, despite having very different fuel designs, both types of HTGR are graphite-moderated, gas-cooled, thermal reactors that use many of the same materials.[68] Additionally, the waste management strategies used in Germany for their pebble bed reactors demonstrate that this waste can be safely and effectively placed into interim storage,

[67] Table derived from the following reports: Kitcher. (2020). "Disposition Options for a High-Temperature Gas-Cooled Reactor". *National Reactor Innovation Center.*; and Kitcher. (2020). "A White Paper: Disposition Options for Sodium-Cooled Fast Reactors". *Idaho National Laboratory*

[68] Kitcher. (2020). "A White Paper: Disposition Options for a High-Temperature Gas-Cooled Reactor". *National Reactor Innovation Center.*

providing valuable insights that can inform future management of U.S. pebble bed reactor waste.

It should be noted that experience with decommissioning MSRs is lacking. This presents a potential challenge for liquid-fueled MSRs, so more technical research on decommissioning methods and waste handling protocols may help refine specific interim storage methods for SNF generated by these reactor designs. However, solid-fueled MSRs, which are more likely to be deployed in the near-term compared to their liquid-fueled counterparts, do not present such issues because their solid fuel can be easily separated from the liquid molten salt coolant.

Advanced reactor developers also already have robust plans for onsite interim storage. Interim storage strategies proposed by both TerraPower and X-energy for their Natrium and Xe-100 reactors, respectively, give insights into the kinds of methods that will be used to manage advanced reactor spent nuclear fuel. Both companies have developed designs for interim storage facilities that have been submitted to the U.S. Nuclear Regulatory Commission, demonstrating their plans to safely manage and store SNF until a permanent disposal solution is available.[69] These interim storage facilities include sophisticated systems and advanced engineering and safety measures, such as passive cooling and secure containment systems, to ensure the safe containment and management of spent fuel over extended periods.

In summary, the United States is well positioned to safely and effectively manage wastes generated by advanced nuclear reactors. Interim storage is expected to be a viable strategy to manage advanced reactor wastes streams prior to the need to permanently dispose of such waste, and it is unlikely that implementing future interim storage strategies will pose any significant challenges.

The following sections provide more details on these key takeaways and specific descriptions of the interim storage methods proposed by two advanced reactor developers, along with those that were employed to manage waste produced by several legacy HTGRs and SFRs. These will offer additional context, provide a working knowledge for the specific methods being proposed, and illustrate what real-world interim storage looks like in practice, highlighting its effectiveness and reliability in managing advanced reactor waste.

4.1. Natrium and Xe-100 Interim Storage Strategies

TerraPower submitted a construction permit application to the Nuclear Regulatory Commission in March of 2024 for the Natrium reactor they are building in Kemmerer, Wyoming. As part of this application, TerraPower included a Preliminary Safety Analysis

[69] Natrium: Preliminary Safety Analysis Report (ML24088A065); X-energy: Spent Fuel Management White Paper (ML23011A324)

Report, which includes details regarding the management and interim storage of Natrium's SNF.[70]

The Natrium Preliminary Safety Analysis Report specifies that its SNF will be placed into wet storage using a "Water Pool Fuel Handling System", which cools the SNF, controls SNF reactivity, contains SNF fission products, and provides radiation shielding. Schematics of this system can be seen in Figure 17 and Figure *18* below. It consists of the following subsystems, and:

- **The Spent Fuel Pool (SFP):** The SFP is a below-grade water-filled concrete pool that includes a stainless-steel liner. This is where SNF is placed in interim wet storage.
- **The Pool Immersion Cell (PIC):** The PIC is the system that transitions core assemblies being stored in a sodium environment to a water-filled environment. It provides a controlled environment to chemically clean and remove residual sodium coolant, before they transfer SNF to the spent fuel pool. SNF assemblies are transferred into the PIC from the bottom loading transfer cask (BLTC).
- **Fuel Pool Cooling (FPC):** The FPC subsystem is the cooling system for the SFP that is designed to maintain the SFP water temperature within prescribed limits. The FPC consists of two independent cooling elements. Each cooling element is sized to remove the entire heat load of the SFP.
- **Fuel Pool Purification (FPP):** The FPP is a filtration and purification subsystem designed to purify SFP water directly from the SFP, and to process the PIC effluent water prior to the return to the SFP. The FPP consists of a single purification element that uses a combination of anion and cation exchange resins to maintain pool chemistry within a specified range.

[70] Natrium Preliminary Safety Analysis Report (ML24088A065)

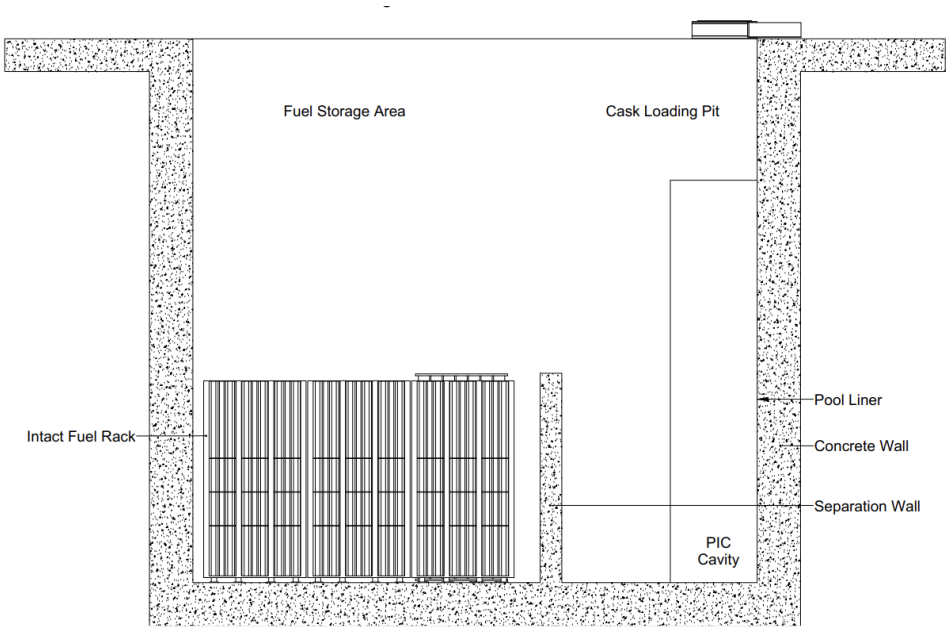

Figure 17: Natrium Spent Fuel Pool, Cross Section View [71]

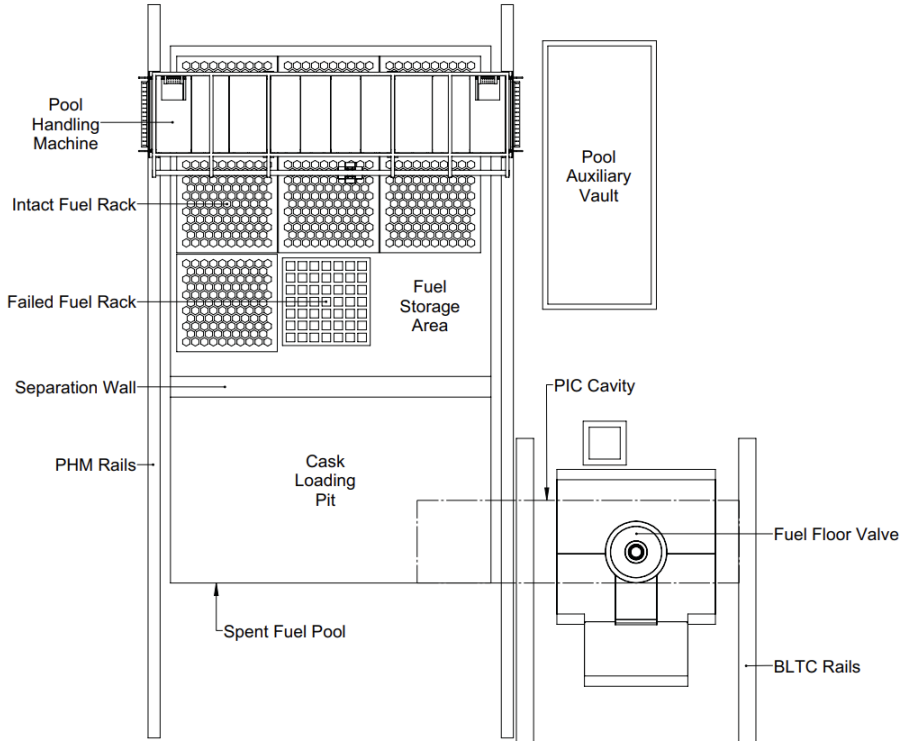

Figure 18: Natrium Spent Fuel Pool, Top-Down View [72]

[71] Natrium Preliminary Safety Analysis Report (ML24088A065)
[72] Natrium Preliminary Safety Analysis Report (ML24088A065)

This facility ensures the safe interim storage of Natrium spent nuclear fuel by providing an environment in which the SNF assemblies can be securely stored, continuously monitored, and managed to prevent any release of radioactive materials. The comprehensive design prioritizes stability and safety, ensuring the containment integrity of the SNF throughout the storage period.

X-energy plans to place their SNF into dry cask storage, using Spent Fuel Intermediate Storage Facilities (SFISF), as shown in Figure *19* below. A SFISF is a robust and highly engineered structure designed for the safe and long-term storage of spent fuel canisters (SFCs) from Xe-100 reactors. Each SFC can hold up to 6,000 spent fuel pebbles and have lids that are welded onto the canisters. The SFISF, measuring approximately 32 meters by 25 meters with 1-meter-thick reinforced concrete walls and roof, can house up to 640 SFCs in its storage racks, ensuring ample capacity for the facility's 80-year service life. The facility's design includes a shielded floor to minimize radiation exposure, allowing personnel safe access, while an overhead crane enables efficient movement of SFCs. The SFISF's ventilation design ensures passive cooling of SFCs through natural air convection. Ventilation is managed through wall-mounted inlets and roof-mounted outlets to maintain a controlled environment. Additionally, all SFC movements between the reactor building and the SFISF occur below-grade via the Inter-Unit Access Tunnel (IUAT), which minimizes radiation doses to surrounding areas. Importantly, the SFISF ensures that all spent fuel from the reactor's operational life can be safely stored on-site, demonstrating flexibility and resilience in nuclear waste management.

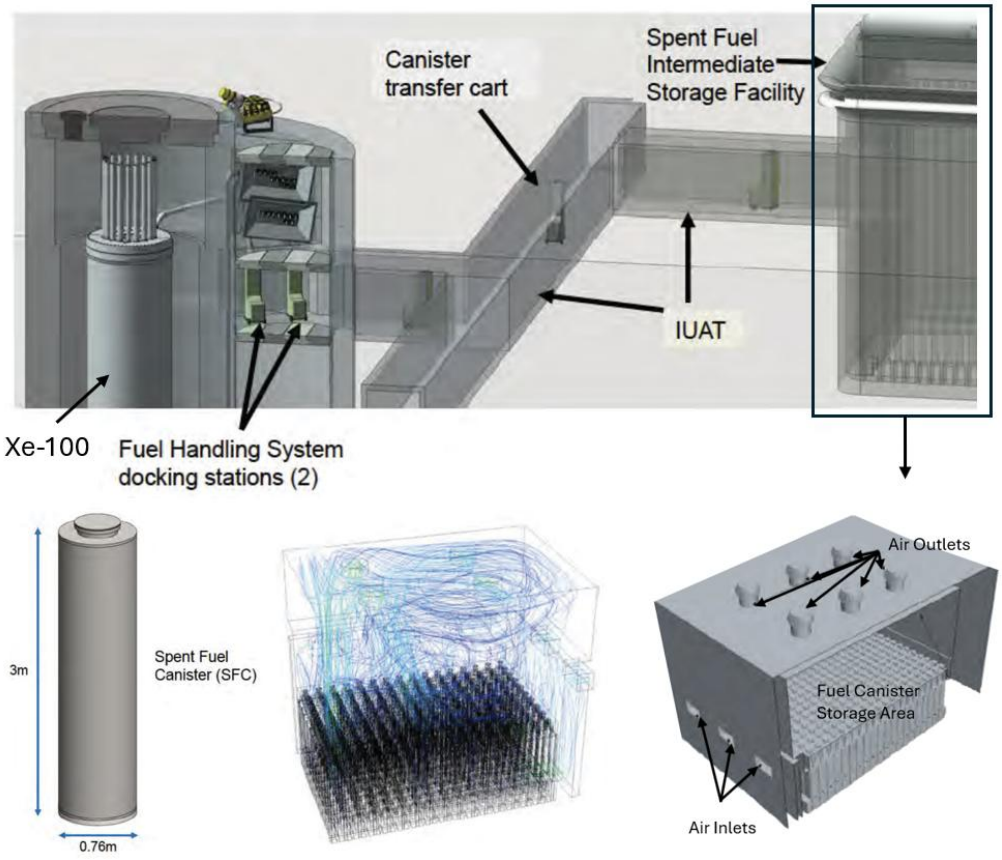

Figure 19: SNF interim storage facility for an Xe-100[73]

Like the Natrium interim storage facility, the Xe-100 facility provides a robust and secure environment for the safe storage and monitoring of spent nuclear fuel. Together, these approaches demonstrate that comprehensive interim storage strategies have been developed. Therefore, interim storage of SNF can be effectively managed, ensuring that the storage process is reliable, efficient, and fully capable of protecting public health and the environment during the interim period before permanent disposal solutions are implemented.

4.2. Legacy High Temperature Gas Reactors

4.2.1. Fort Saint Vrain

The Fort Saint Vrain (FSV) reactor operated from 1976 to 1989, and decommissioning took place between 1992 and 1996. It was located in northern Colorado, and produced approximately 23 metric tons of SNF. Upon decommissioning, the hexagonal spent fuel

[73] X-energy | ML23011A324

elements were placed within cylindrical carbon steel storage canisters that could each hold up to six graphite SNF elements.

Roughly one third of these canisters were packaged within specialized transportation casks and shipped to the Idaho National Laboratory where they were placed in dry cask storage at the convection-cooled CPP-603 Irradiated Fuel Storage Facility.[74] Images of these transportation casks and the CPP-603 storage facility can be seen in Figure *20*.

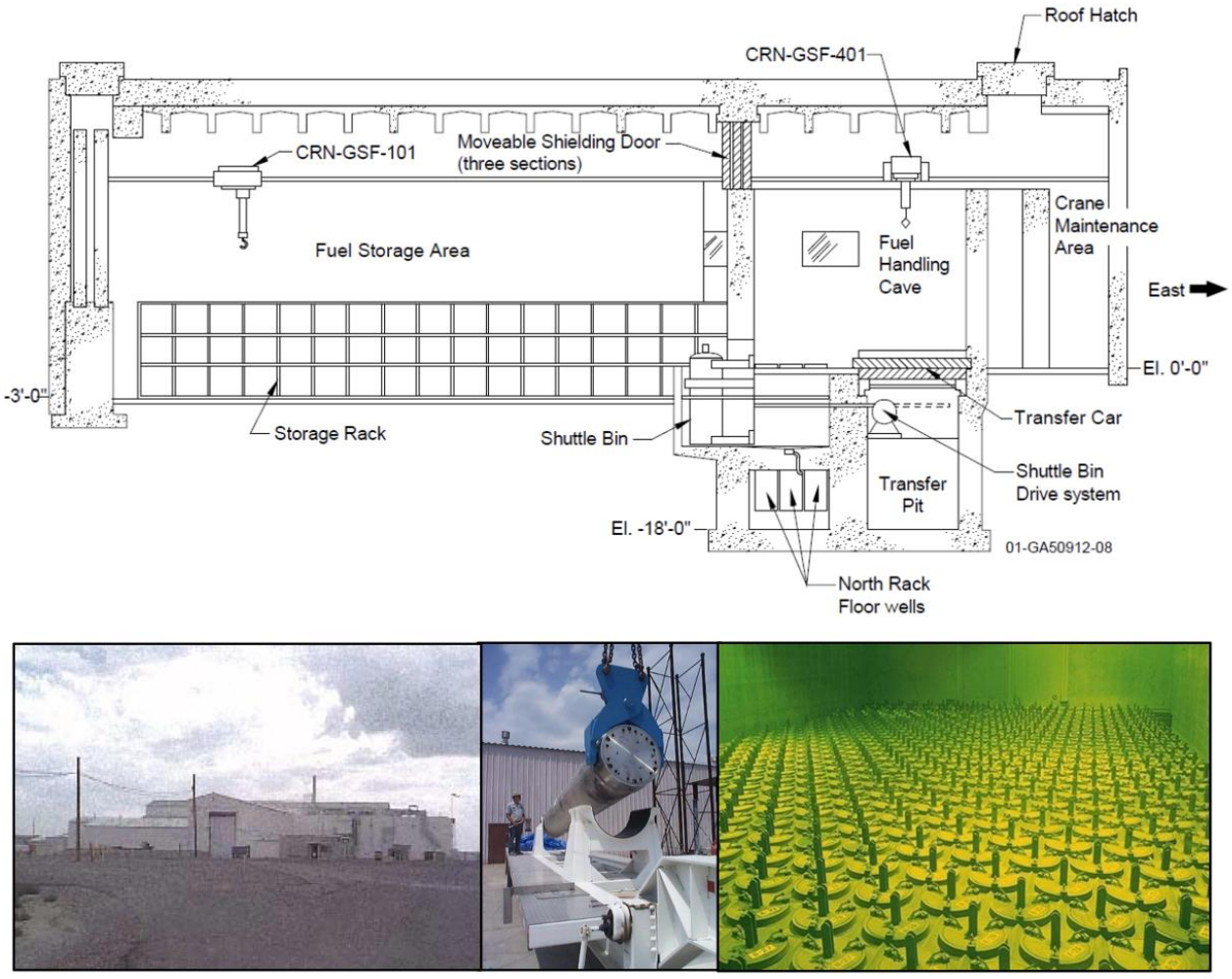

Figure 20: Cross-section view of CPP-603 facility (top); TN-FSV transport cask used for FSV spent fuel (bottom middle); Photo of the exterior of the CPP-603 facility (bottom left); Photo of the CPP-603 facility's fuel dry storage area (bottom right) [75]

[74] U.S. Nuclear Waste Technical Review Board | Department of Energy – Managed Spent Nuclear Fuel at the Idaho National Laboratory

[75] (1) Thomas. (2019). "Preliminary Evaluation of Loading DOE Standardized Canisters in the CPP-603 Irradiated Fuel Storage Facility". *Idaho National Laboratory*; (2) U.S. DOE | DOE-Managed Spent Nuclear Fuel; (3) U.S. DOE | Department of Energy – Managed Spent Nuclear Fuel at Fort St. Vrain; (4) U.S. DOE | Idaho Site Spent Nuclear Fuel Management;

The remaining spent fuel from the FSV reactor was never moved from the FSV site in northern Colorado. It is currently stored on-site in a specialized Independent Spent Fuel Storage Installation (ISFSI) that is designed to store FSV's SNF in a concrete structure, as shown in Figure *21* below. This concrete structure has vertical storage positions within the concrete that can each store one fuel storage container, and these storage positions are cooled by natural air circulation. Currently, 244 fuel storage containers have been placed in this facility, meaning a total of 1,464 spent fuel elements (244*6) are stored onsite. There are no records of TRISO-coated particle fuel failure under the storage conditions at FSV.[76]

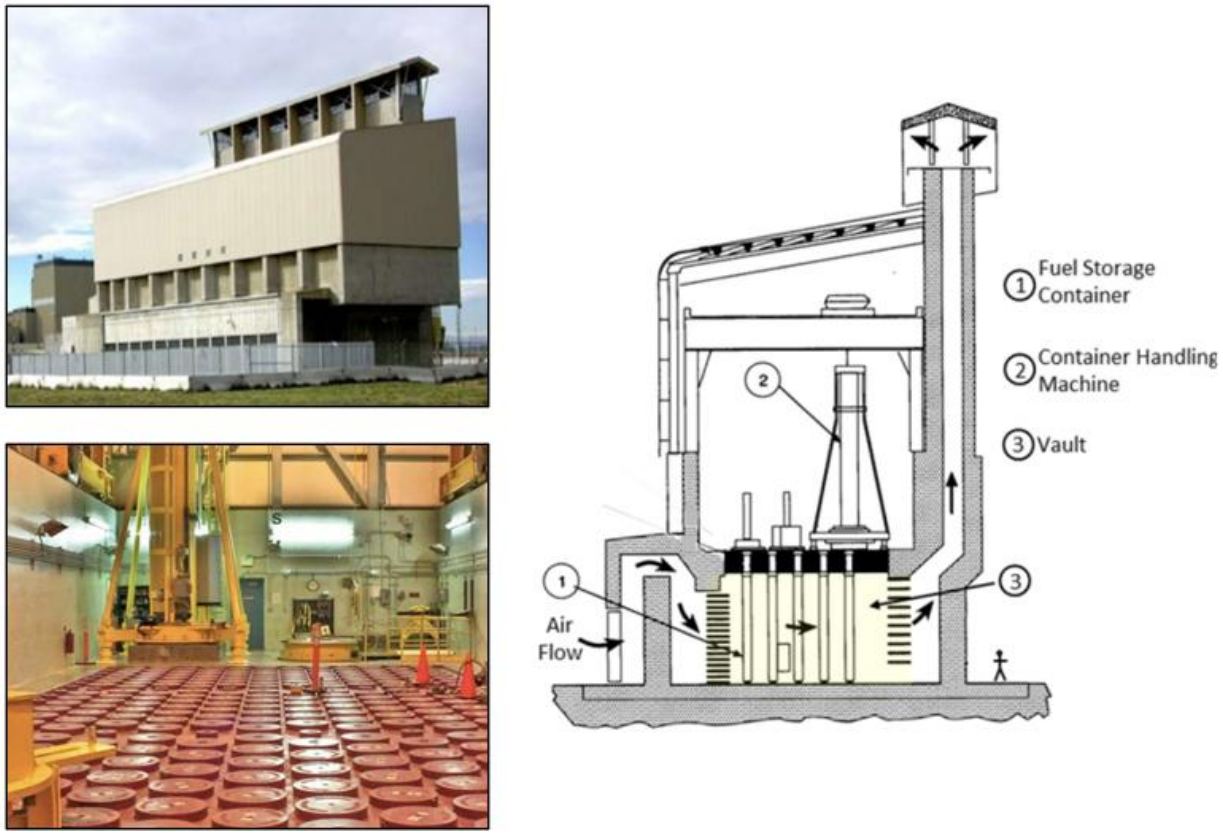

Figure 21: Exterior Photo of the FSV ISFSI (top left); Photo inside the FSV ISFSI (bottom left; Diagram of the FSV ISFSI (right)[77]

4.2.2. German AVR

The German Arbeitsgemeinschaft Versuchsreaktor (AVR) HTGR was a pebble bed reactor that was commissioned in 1969 and decommissioned in 1988. The management of its spent fuel involved packaging its spent fuel pebbles into storage canisters and casks that were placed in

[76] Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*
[77] U.S. Nuclear Waste Technical Review Board | Department of Energy – Managed Spent Nuclear Fuel at Fort St. Vrain

wet storage, then dry cask storage. The entire TRISO spent fuel pebble interim storage process can be seen in Figure *22*.

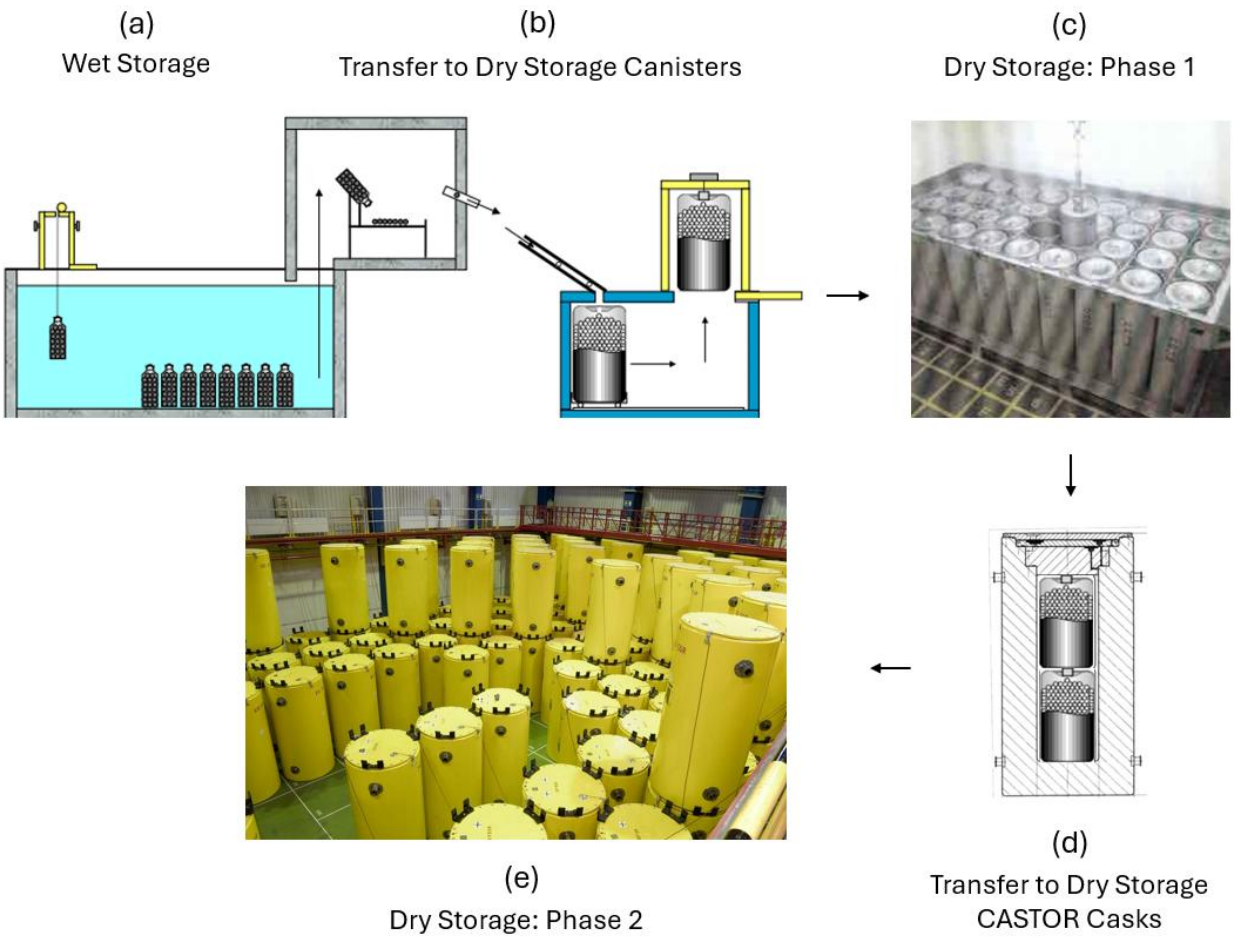

Figure 22: AVR spent fuel management.[78]

The above image shows that spent fuel pebbles were gravity fed from the reactor into stainless steel cans, which held 50 fuel pebbles each. These steel cans were then transferred to Germany's Forschungszentrum Jülich (FZJ) site where they were stored in a pool of water for approximately two years to allow for heat dissipation (see (a) in Figure 22). From wet storage, the cans were opened, and the fuel pebbles were transferred to stainless steel dry storage canisters (see (b) and (c) in Figure 22). These dry storage canisters could each hold 950 pebbles and were filled with helium. Eventually, these dry storage canisters were packaged into larger and more robust cast iron CASTOR-THTR/AVR casks (see (d) in Figure 22). Each CASTOR cask weighs about 25 metric tons, is roughly 9 feet tall and 4.5 feet in diameter, has a specialized double-barrier lid system to prevent leaks, and contains two

[78] IAEA. (2012). "Advances in High Temperature Gas Cooled Reactor Fuel Technology"; and Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*; EWN | Interim Storage

vertically oriented dry storage canisters. In total, each CASTOR cask holds 1900 (950*2) total fuel pebbles. These CASTOR casks were then moved to an interim dry storage facility at FZJ (see (e) in Figure 22). This dry storage facility passively cools the CASTOR casks through the use of natural convection and contains 153 CASTOR casks that hold roughly 290,000 TRISO pebbles in total. Investigations into the performance of these casks have found the chances of a cask leaking and releasing radioactive material to be negligibly low.[79]

4.3. Legacy Sodium-Cooled Fast Reactors

4.3.1. Experimental Breeder Reactor-II

The sodium-bonded spent metallic fuel generated by Experimental Breeder Reactor-II (EBR-II) was originally stored in roughly 3,600 stainless steel containers that were placed in wet storage at INL.[80] Between 2011 and March 2023, DOE completed the process of transferring all this spent fuel from wet storage to below-grade dry cask storage at the Radioactive Scrap and Waste Facility at INL. This involved the transfer of more than 100 shipments of spent fuel from wet storage to dry cask storage, as part of an Idaho Settlement Agreement that was signed in 1995 between DOE, the state of Idaho, and the U.S. Navy.[81]

To prepare for dry cask storage, the spent fuel was transferred into containers that have an inner layer of carbon steel, a middle layer of stainless steel, and an outer layer of carbon steel, all designed to prevent release of radioactive materials or intrusion of water. The outer carbon steel layer is cathodically protected from corrosion, and it has a 30-inch concrete shield plug at the top to shield radiation and prevent water intrusion. Images of the wet and dry cask storage can be in Figure *23*.

[79] IAEA. (2012). "Advances in High Temperature Gas Cooled Reactor Fuel Technology"; Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*

[80] Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*

[81] World Nuclear News | Final fuel transfer from storage basin at INL

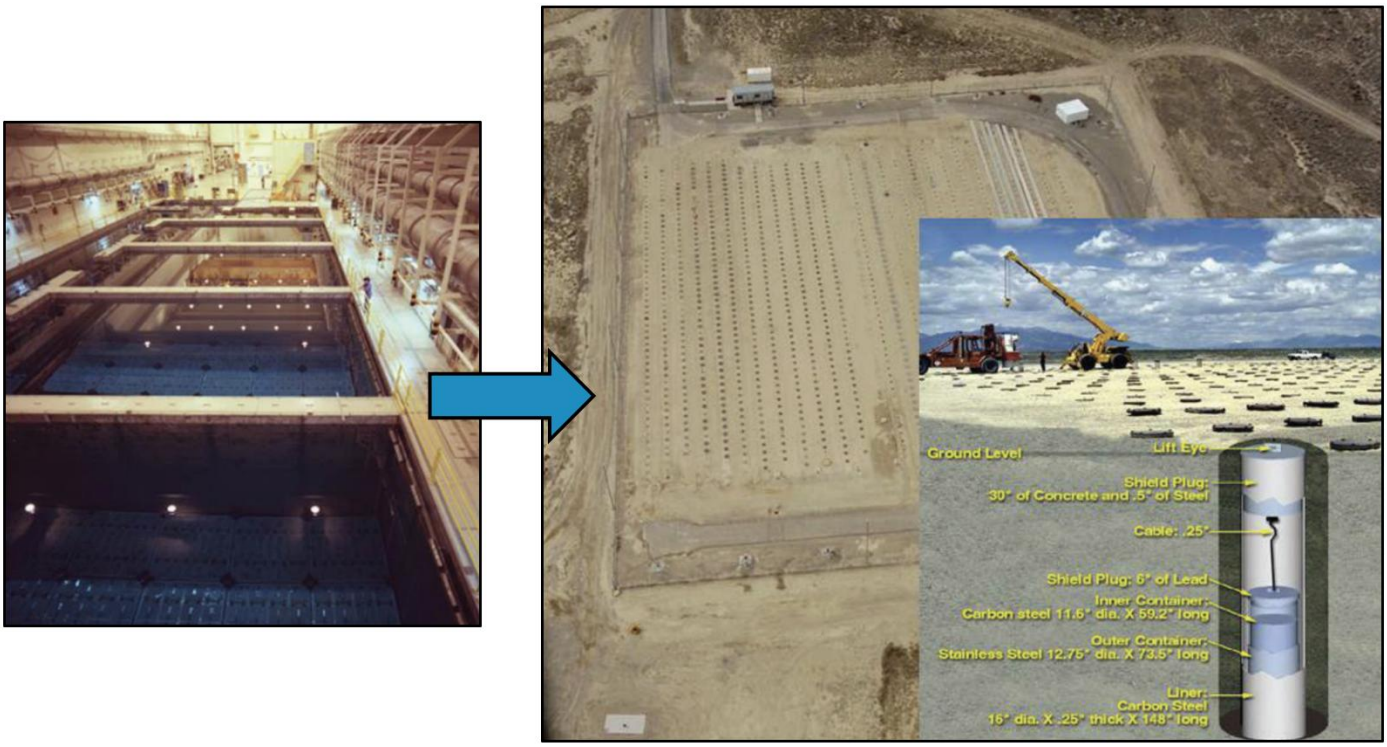

Figure 23: Wet storage (left) and dry cask storage (right) of EBR-II SNF at INL[82]

DOE is currently working to systematically take the SNF that is in dry cask storage at this site and process it for eventual permanent storage (see section 5.2 and Appendix C for more information on SFR processing).

4.3.2. Fermi 1

Sodium-bonded spent metallic fuel from Fermi 1 is currently in dry cask storage at INL's Idaho Nuclear Technology and Engineering Center. This facility features two primary storage areas that include vertical vaults made of carbon steel pipes with shield plugs. There are two generations of vault designs, as shown in Figure *24* below. The first-generation design is entirely below grade and the second-generation design extends above grade to prevent surface water from entering the vault. Although detailed public information on the vaults' construction is limited, the second-generation design offers improved protection against water intrusion. These vaults are routinely monitored for hydrogen levels, corrosion, and overall condition. Due to the unique characteristics of the Fermi-1 spent fuel, DOE is currently exploring alternative treatment methods for these stored fuels. [83]

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[82] Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*

[83] Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*

Figure 24: Dry cask storage of Fermi-1 SNF at the INL's Idaho Nuclear Technology and Engineering Center. 1st-generation underground vaults built in 1971 (left) and 2nd-generation underground vaults built in 1984 and 1985 (right). [84]

5. Permanent Disposal

The permanent disposal methods used for advanced reactor SNF will generally align with those needed to dispose of SNF generated by conventional reactors, given the need to ensure the long-term safety and security of these materials. SNF from both conventional reactors and advanced reactors contain radioactive elements that remain hazardous for thousands to millions of years and must be isolated from the public and the environment to prevent any potential exposure. This long-term physical isolation can be achieved by using deep geological repositories, where waste can be securely stored far beneath the earth's surface in stable geological formations. The design of these repositories focuses on preventing the release of radioactivity through multiple engineered and natural barriers, ensuring that the waste remains contained and does not pose a threat to human health or the environment.

The following subsections will discuss the potential impacts of advanced reactor waste streams on repository performance, design, and cost, along with the necessary processing steps that could be used to help prepare these wastes for permanent disposal. By addressing these considerations, we can better understand what is needed to effectively manage the diverse waste streams generated by advanced reactors over the long term, while maintaining the stringent safety standards required for permanent disposal.

5.1. Performance of an Advanced Reactor Waste Repository

[84] Hall et al. (2019). "Storage Experience with Spent (Irradiated) Advanced Reactor Fuel Types". *Center for Nuclear Waste Regulatory Analyses*

The waste streams generated by advanced reactors raise important questions about their management and permanent disposal. These unique waste forms, due to variations in fuel types and reactor materials, necessitate careful consideration to ensure safe and effective disposal methods.

A primary concern is whether these new waste forms will impact the performance of geological repositories, which are designed to isolate radioactive materials from the environment. However, advanced reactor waste streams are expected to have little to no impact on the long-term performance of geological repositories provided they meet the repository's waste acceptance criteria.[85]

At first glance, this conclusion may seem counterintuitive because, as discussed in previous sections, advanced reactors produce a wide range of waste streams with varying radionuclide inventories, chemical compositions, and physical forms due to variations in fuel types and materials used within the reactor. However, the long-term performance of a geological repository is dominated by the robustness of the engineered and geological barriers designed to contain the waste, not the characteristics of the waste, as long as the waste has been properly processed and packaged prior to disposal.

5.2. Processing Advanced Reactor Wastes

Some advanced reactor waste streams may need to be processed prior to permanent disposal to reduce the waste's overall volume, render it chemically inert, or stabilize it to ensure long-term safety in a geological repository. Other advanced reactor waste streams may be directly disposed of (i.e., direct disposal) without any major processing steps.

The term "processing" encompasses all activities that are designed to produce a waste form that is acceptable for disposal. Nuclear waste processing can generally be separated into three categories:[86]

- **Pre-Treatment:** Pre-treatment prepares the waste for subsequent processing stages. It involves activities such as collection, segregation, and decontamination to separate contaminated materials from non-contaminated materials.
- **Treatment:** Treatment focuses on altering the waste's properties, often by changing the waste's composition or reducing the volume of waste by separating different radioactive components by their level of radioactivity or contamination.
- **Conditioning:** Conditioning involves stabilizing the waste to prepare it for transport, storage, and disposal. This step ensures that the waste is encapsulated or solidified in

[85] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 11

[86] IAEA | Processing

For more detailed information on processing techniques used to handle existing nuclear waste streams, including defense related waste streams, see Appendix C of this report.

stable matrices, such as cement, bitumen, or glass, to prevent the release of radionuclides into the environment. The conditioned waste can then be packed into special containers to provide additional containment and shielding, ensuring long-term safety.

These three categories often overlap and therefore the terminology is not always clearly defined, but they generally refer to distinct activities, often emphasizing different aspects of waste handling as shown in Figure *25*.

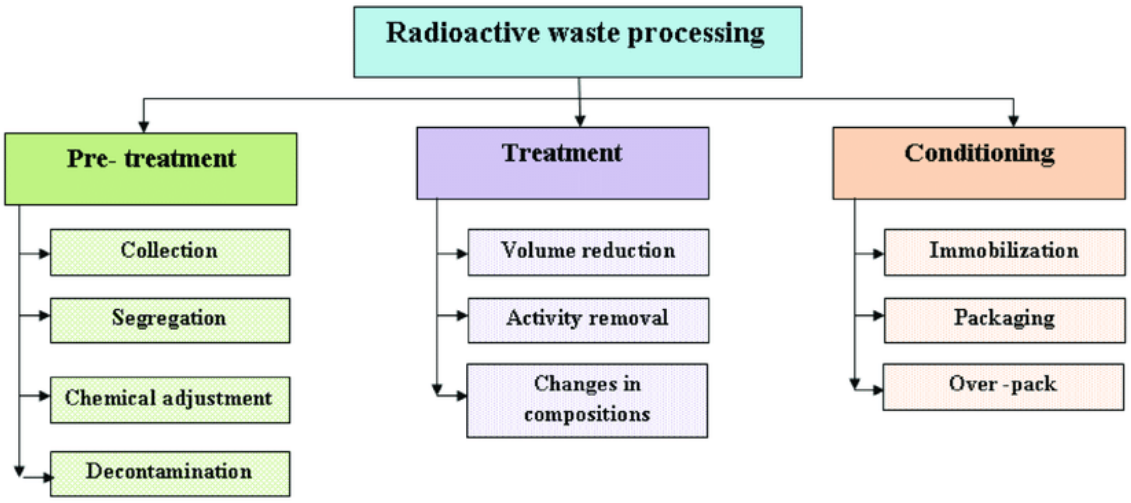

Figure 25: The various aspects of nuclear waste processing[87]

The specific processing methods that can be used depend on the waste form in question. As discussed in Chapter 3, there will be a wide range of wastes produced by advanced reactors that correspond to the specific reactor technology being used. The processing methods that could be used for several advanced reactor technologies are summarized in Table *5* below and are discussed in greater detail in Appendix C.

[87] Abdel Rahman, Ojovan. (2021). "Toward Sustainable Cementitious Radioactive Waste Forms: Immobilization of Problematic Operational Wastes". *Sustainability*

	ALWRs	HTGRs	SFRs	Solid-fueled MSRs	Liquid-fueled MSRs
SNF Waste Form:	UO_2 fuel pellets	TRISO pebbles or prismatic blocks	Sodium-bonded spent metallic fuel	TRISO pebbles or prismatic blocks	Spent fuel salt
Stable SNF Waste Form?[1]	Yes	Yes	No	Yes	No
Disposal Pathway	Direct Disposal	Direct Disposal	Direct Disposal or Processing[2]	Direct Disposal	Direct Disposal or Processing[2]
Processing Method	n/a	n/a[3]	Varies	n/a[3]	Varies
Processing Method Maturity	n/a	n/a[3]	Lab scale demonstrations	n/a[3]	Has not been demonstrated

1. Chemical stability prior to any potential processing
2. Dependent on the design and waste acceptance criteria of the final repository, and laws and regulations that govern. For example, the Resource Conservation and Recovery Act would not currently allow for the direct disposal of sodium bonded spent metallic fuel.
3. Incineration and mechanical separation processes have been proposed but are not generally considered preferable to direct storage

Table 5: Advanced Reactor Permanent Disposal Pathways and Processing Methods

SNF from ALWRs, HTGRs, and solid-fueled MSRs could be directly disposed of, but certain challenges to direct disposal exist. For example, direct disposal of the entire TRISO fuel pebble would involve disposing of the TRISO fuel particles and the large volumes of graphite surrounding them. While this graphite is radioactive, it will likely not be classified as HLW, so it would increase the total volume of material that is placed in permanent disposal facility. Certain processes have been proposed to separate the TRISO fuel particles from the surrounding graphite to lower the total volume in need of permanent disposal, but they are generally not thought of as feasible or economic.

The processing methods that could be used for SFR and liquid-fueled MSRs will require additional research, development, and deployment initiatives to ensure they can be used at scale due to their complexity and the maturity of the technology. For example, processing methods used to separate the sodium from spent metallic fuel produced by SFRs have been performed at INL on legacy EBR-II SNF. However, this can currently only be performed on relatively small quantities of SNF because the current technology being used is at a "lab-scale". SNF generated by liquid-fueled MSRs may even require processing methods that have not yet been demonstrated.

Despite these challenges, their solutions are known to be technically feasible. Given the availability and robustness of interim storage, it is not critical to develop permanent solutions in the near term. However, it is important to begin planning and identifying pathways to implement these solutions now. DOE should conduct additional technical evaluations to determine viable disposal pathways for spent nuclear fuel from advanced reactors, particularly SNF from SFRs and liquid-fueled MSRs. These evaluations should assess the

feasibility of direct disposal or if additional processing steps are needed to increase the safety and stability of SNF prior to permanent disposal in a geological repository. These evaluations should also consider how various geological repository environments and their conditions can impact this determination.

5.3. Design and Costs of an Advanced Reactor Waste Repository

Outside of a permanent repository's primary performance metric, (i.e., its ability to limit radionuclides from migrating and reaching the surrounding environment and public) advanced reactor wastes will impact certain aspects of geological repository design. This is partly due to the higher thermal loads associated with advanced reactor waste packages, which result from the higher burnup of fuels used in advanced reactor environments compared to conventional reactor SNF.

Burnup is a measure of how much energy is extracted from nuclear fuel and is typically expressed in gigawatt-days per metric ton of uranium (GWd/MTU). Higher burnup means that the fuel has been used more efficiently, extracting more energy per unit of nuclear fuel before being removed from the reactor. Advanced reactors often operate at higher burnup levels compared to traditional reactors, leading to more efficient fuel use. However, higher burnup also results in greater accumulation of fission products, which contributes to increased heat generation and higher thermal loads in the spent fuel.

Any increased thermal load from advanced reactor wastes necessitates careful consideration in the design of geological repositories. Specifically, the spacing between HLW packages must be increased to manage the additional heat generated, ensuring that temperatures remain within safe limits to prevent degradation of the geological and engineered barriers. This increased spacing requirement can expand the total footprint of the repository, necessitating more space for storage.[88]

The volume of a permanent repository is not, however, dictated solely by the total volume of material that must be disposed of. Spent fuel that is placed in interim storage for long periods has more time to cool and reduce the heat load, effectively decreasing the volume needed within a repository to handle the fuel. The more time SNF spends in interim storage, the smaller a repository needs to be. However, additional safeguards at an interim storage facility that contains high burnup may be needed. The relationship between time spent in interim storage, SNF heat load, and impacts to a repository size and cost are therefore all interrelated.

Additionally, the physical characteristics of advanced reactor waste and their disposal packages, such as their volume, mass, and the wastes final waste form, will influence various

[88] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 11

aspects of a repository's design, including the size and configuration of storage areas, the layout of waste packages, and overall volume of a repository. Each of these will, in turn, impact the costs associated with constructing a repository. Many of the processing steps that may or may not be taken prior to final waste disposal will impact the volume and other physical characteristics of advanced reactor wastes. These steps will be crucial in determining the final form and volume of the waste, directly influencing repository design and costs. Optimizing these processes can reduce the repository's footprint and complexity, leading to more efficient use of space and resources while ensuring safety.

Conclusion

As the global shift toward cleaner, more reliable energy continues, advanced nuclear energy technologies stand at the forefront of meeting the growing demand for sustainable solutions. These reactors offer the promise of firm, zero-carbon power, but efforts to deploy these technologies can depend on addressing the questions surrounding nuclear waste management.

A comprehensive understanding of the unique waste streams generated by advanced nuclear reactors, the readiness of our waste management systems, and the innovations needed for future permanent disposal is critical. This knowledge can help empower policymakers to create the conditions for success, ensuring that advanced nuclear energy can play a central role in addressing climate change and meeting future energy demands. The path forward lies in informed decision-making, continued innovation, and a clear commitment to the safe and responsible management of nuclear waste, positioning advanced nuclear technology as a vital component of our clean energy future.

Advanced nuclear reactors will generate a wide range of waste streams that require different management strategies. While final disposal pathways for advanced nuclear reactor wastes are still being considered, current strategies for interim storage are well-developed and equipped to safely manage waste until a permanent repository is constructed. The absence of a permanent repository in the U.S. underscores the need for future action, but interim storage solutions will ensure that advanced nuclear reactor waste can be handled securely in the meantime. Additionally, permanent disposal pathways for advanced reactor wastes are known to be technically feasible and are currently being explored. With continued innovation and planning, waste streams from advanced reactors are not expected to be a barrier to continued deployment supported by strong waste management frameworks that ensure their long-term sustainability and safety.

Appendix A: Classifying Low-Level Waste with Long-Lived and Short-Lived Isotopes

Table 1 and Table 2 in Section 2.2.2 of this report specify which LLW classification (i.e., Class A, Class B, Class C, or GTCC) is applicable to LLW materials that contain long-lived or short-lived isotopes shown in the tables. This appendix presents the provisions in 10 CFR 61.55 that specify how to classify LLW that contains multiple long-lived or short-lived isotopes that each fall within a different class, or a mixture of both long-lived and short-lived isotopes.

For wastes containing mixtures of only short-lived radionuclides, or only long-lived radionuclides, waste classification is determined by the "sum of fractions rule", where the sum fraction is calculated as follows:

$$\text{Sum Fraction} = \sum \frac{C_R}{C_L}$$

Where:

- C_R is the concentration of the individual radionuclide; and
- C_L is the concentration limit for the radionuclide mixture. This value is set equal to the upper bound of the range presented within the table for each respective radionuclide.[89]

If the sum fraction is less than one for that particular waste classification, then that is the appropriate waste classification. For example, if a particular waste material contains 50 Ci/m^3 of Sr-90 and 22 Ci/m^3 of Cs-137, both of which are short-lived radionuclides, the calculation would be as follows:

$$\text{Sum Fraction} = \frac{50}{150} + \frac{22}{44} = 0.33 + 0.5 = 0.83$$

Given the sum fraction for this example, which used the Class B upper bounds in the denominator, was less than 1, this mixture would be considered Class B nuclear waste. Should the calculation have resulted in a sum fraction that is greater than 1, the same calculation would have been performed using the Class C upper bound.

For nuclear waste that contains a mixture of both long-lived and short-lived radionuclides, if its long-lived radionuclides have concentrations that only fall within the Class A column of Table 1, the class is determined by the short-lived nuclide concentrations. If its long-lived radionuclides have concentrations that fall within the Class C of Table 2, it is Class C waste, unless it contains GTCC short-lived radionuclides, in which case the mixture would be GTCC

[89] For mixtures, the appropriate concentration limits must all be taken from the same column of the table. For example, a mixture containing 50 Ci/m^3 of Sr-90 and 45 Ci/m^3 of Cs-137 would have C_L values set equal to 7,000 for Sr-90 and 4,600 for Cs-137 (not 44 for Cs-137).

nuclear waste. If the LLW does not contain any radionuclides listed in either Table 1 or Table 2, and is not HLW, it is class A waste.

Appendix B: TRISO Fuel Particle Layers

TRISO fuel particles are multi layered composite materials. Each layer, including the fuel kernel, serves a unique purpose.

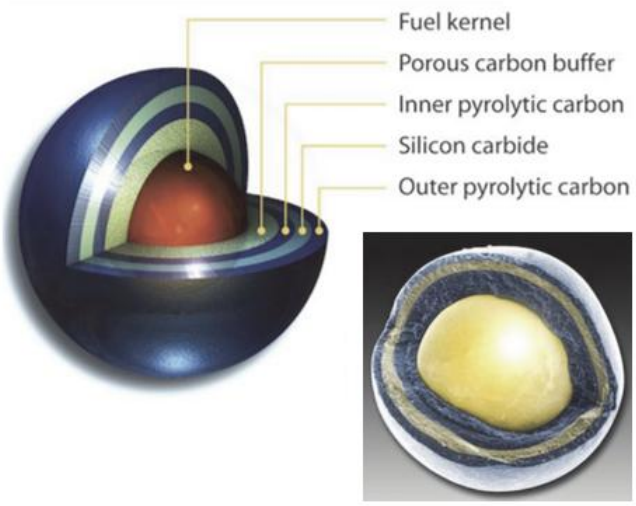

Figure 26: Illustration (left) and false color image (right) of a TRISO fuel particle

A description of each layer, and the unique purpose it provides, is as follows:[90]

- **Kernel:** The kernel is the spherical fissionable fuel at the center of the TRISO particle. In addition to being fissionable fuel, the kernel also serves as a barrier to radionuclide release by immobilizing and/or delaying fission products. A broad range of fissionable fuels are used to make kernels and include: UO_2, $(U,Th)O_2$, UC_2, (U,Th)C, PuO_2, and UCO. UCO is a conglomerate of UO_2, UC, and UC_2 chemical compounds. The primary difference between the UO_2 and UCO kernels is that the UCO kernels limit oxygen activity. Reducing oxygen activity reduces the generation of CO and CO_2, which has benefits for kernel migration and reducing gas pressure in the particle, allowing for higher burnup limits and thermal gradients.
- **Porous carbon buffer:** The fuel kernel is surrounded by a porous carbon buffer that provides void space to accommodate fission gas release. The purpose of the buffer is to absorb the kinetic energy of fission fragments ejected from the fuel kernel surface and to provide space for the accumulation of gaseous fission products and carbon monoxide. It functions by mechanically decoupling the kernel from the inner pyrolytic carbon layer to accommodate kernel swelling.

[90] Wells et al. (2021). "TRISO Fuel: Properties and Failure Modes". *PNNL*

- **Inner pyrolytic carbon:** The inner pyrolytic carbon (often referred to as "IPyC") is a dense layer of carbon with approximately 85% porosity. The IPyC serves several purposes. It protects the kernel from corrosive gases (HCL, CL2) liberated during the silicon carbide coating process. The IPyC layer is also the first load-bearing barrier and provides structural support for the silicon carbide layer. The IPyC layer also protects the silicon carbide layer from fission products during operation by retaining gaseous fission products.
- **Silicon carbide:** The silicon carbide layer is a high-density, high-strength layer of silicon carbide whose primary function is to provide structural stability to the particle, and to act as a pressure vessel for internal fission products.
- **Outer pyrolytic carbon:** The outer pyrolytic carbon layer (often referred to as "OPyC") is another layer of high-density carbon. The OPyC acts to protect the fuel particle as it is being deposited in the final fuel form (i.e., a fuel pebble or prismatic block matrix). The OPyC layer also provides structural support for the silicon carbide layer and acts as an additional barrier to the release of gaseous fission products in the event of silicon carbide failure. Because the matrix material of the fuel compact will not bond to the SiC layer, the OPyC layer is necessary to provide a bonding surface between the TRISO particles and the carbon used for the final fuel form.

Appendix C: Processing Methods for Advanced Reactor Wastes

F.1 High-Temperature Gas Reactors

The use of graphite as a moderator and structural component in TRISO fuel particles in pebbles or prismatic blocks provides benefits but also presents challenges. More specifically, the disposal of large volumes of radioactive graphite surrounding the TRISO fuel pebbles increases the total volume of HLW that must be placed in permanent disposal.[91]

Certain processes have been proposed to separate the fuel particles from the surrounding graphite to lower the total volume in need of permanent disposal. These methods include combustion and mechanical separation. Combustion involves oxidizing the graphite off of the TRISO fuel pebbles. Mechanical separation involves crushing the TRISO fuel pebble or prismatic block to physically separate the graphite from the TRISO particles. Both processes, however, come with their own set of challenges. For example, the combustion process will produce radioactive carbon dioxide gas as a biproduct, and mechanical separation requires

[91] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 162

accounting for tens of thousands of less than one-millimeter TRISO particles per fuel pebble. Therefore, isolating the TRISO fuel particles from the fuel pebbles or prismatic blocks is generally not thought of as feasible. Instead, direct disposal of the TRISO fuel into a permanent repository is considered the preferred method.

Direct disposal involves placing TRISO fuel particles or prismatic blocks directly into disposal packages and disposing of those packages within a permanent repository. Not only does the graphite of the TRISO fuel increase the total volume that must be disposed of, but the packing factor of the fuel pebbles does too because of the empty space between spherical fuel pebbles. This can impact the total volume, and cost, of a permanent repository. However, as discussed in section 5.3, the volume of a permanent repository is not dictated solely by total volume of material that must be disposed of, but also by the heat load of the waste.

It should be noted that the United Kingdom leads the world in research and development for irradiated graphite disposal, due to its extensive inventory of irradiated graphite that was generated by the UK's use of graphite-moderated Magnox and Advanced Gas Reactors throughout its history.[92] The UK, along with many other countries, participates in several international initiatives aimed at advancing methods to dispose of irradiated graphite. These initiatives include the International Atomic Energy Agency's (IAEA's) IMMONET program, which is a repository for data and reports on irradiated graphite, and GRAPA (GRAphite Processing Approaches), which is a program that seeks to build a comprehensive knowledge base on irradiated graphite waste. Additionally, the European Commission has developed a network of experts through the CARBOWASTE program to evaluate technologies for the characterization, retrieval, treatment, recycling, and disposal of irradiated graphite.

F.2 Solid-Fueled Molten Salt Reactors

Management of solid-fueled MSR spent fuel will be similar to HTGRs spent fuel because both reactor designs use TRISO fuel pebbles. The same considerations regarding TRISO fuel processing will also apply to solid-fueled molten salt reactors. Certain differences will exist, for example residual molten salt that remains on the TRISO fuel pebbles surface may need to be removed prior to interim storage, but after this step the management of the spent fuel from a permanent disposal perspective closely mirrors that of HTGRs.

It is worth noting that solid-fueled MSRs, such as the Kairos KP-FHR reactor design and Hermes demonstration reactor, will result in a molten salt waste stream that has relatively low levels of radioactivity. This is because the salt does not come into direct contact with the fuel, preventing the transfer of fission products and other radionuclides into the coolant. The specific waste classification for the molten salt coolant will vary depending on the level of contamination and activation but will generally be classified as LLW. Consequently, the management and disposal of this coolant salt becomes less complex and hazardous

[92] IAEA | History of Graphite in the UK Nuclear Industry

compared to MSRs where the fuel is dissolved in the salt, reducing the overall radioactive waste burden.

F.3 Liquid-Fueled Molten Salt Reactors

Spent Fuel Salt

Direct disposal of spent fuel salt from liquid-fueled MSRs in a deep geologic repository is feasible, given the considerations discussed in section 5.1. However, if the repository were breached, these salts are soluble in water and could mobilize. Therefore, another option is to immobilize the salt in a waste form that is more chemically resistant towards dissolution and transportation to the environment.

There is currently no technically mature approach to processing spent fuel salt, although several options exist, and a number of R&D efforts are needed to advance current liquid-fueled MSR waste management practices. These options, which vary for either fluoride- or chloride-based MSRs, generally attempt to process the spent fuel salt into a stable glass, ceramic, or ceramic-metal composite waste form.[93] The majority of these stable waste forms and the processes to generate them, however, are only theoretical, as shown in Figure *27*. Of note, there are no waste forms experimentally proven for MSR fluoride salt, and only some have been proven for MSR chloride salt.[94] This presents a large gap in our current understanding of how to process spent fluoride fuel salt. In fact, it has been stated in recent literature that "very little (if any) research has been done to evaluate methods for immobilizing fluoride salt wastes from MSRs, so this field remains wide open for new research."[95]

[93] Riley et al. (2019). "Molten salt reactor waste and effluent management strategies: A review"
[94] Arm et al. (2020). "Status of Fast Spectrum Molten Salt Reactor Waste Management Practice"
[95] McFarlane et al. 2020. "Molten Salt Reactor Engineering Study for Off-Gas Management"

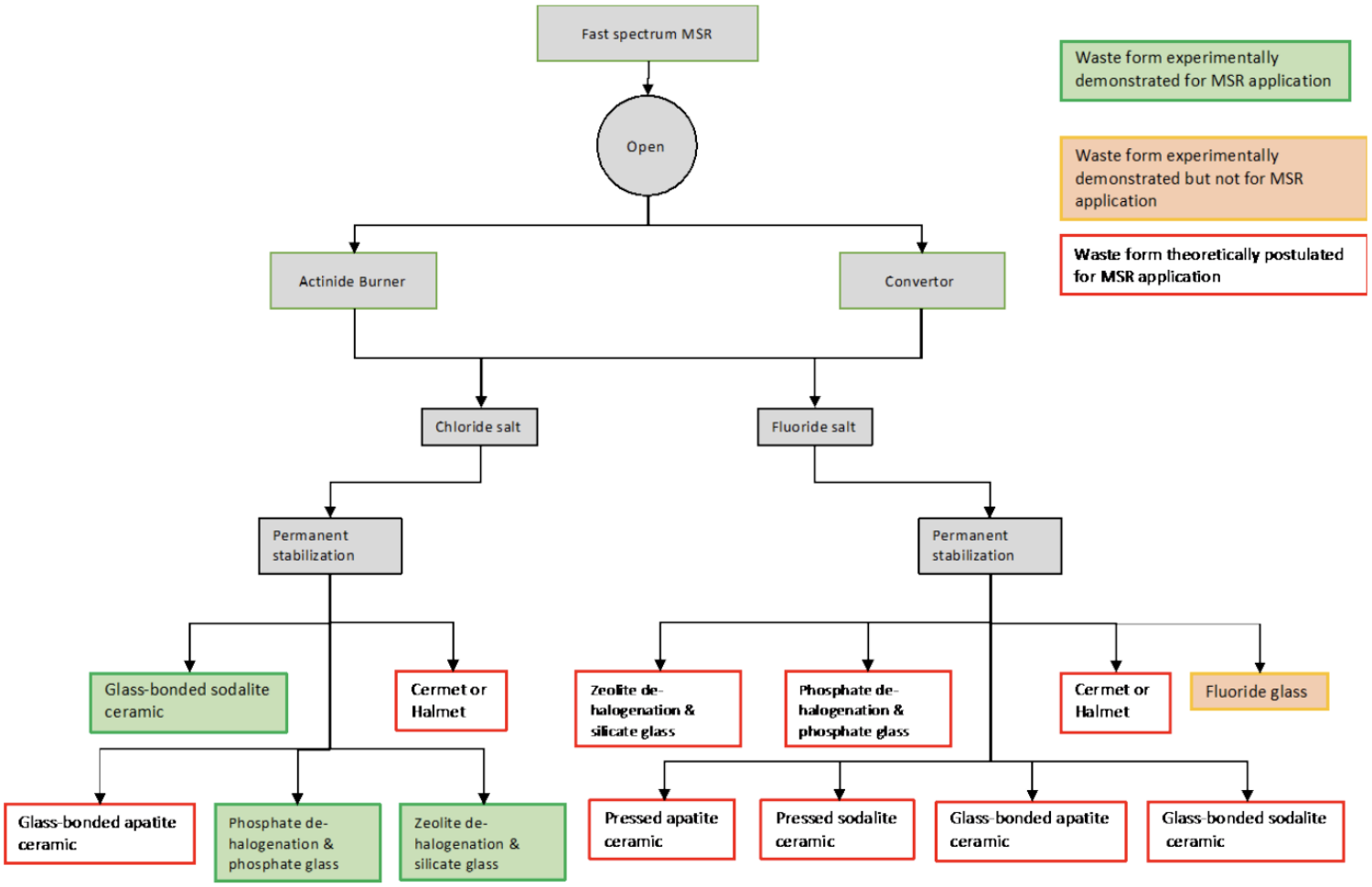

Figure 27 Waste Forms Applicable to Permanent Stabilization of Used Fuel Salt in an MSR[96]

Off-Gas Wastes

The many different species of radioisotopes found within off-gases produced by liquid-fueled MSR necessitate various processing methods to manage them effectively. There are currently several proposed processes for managing these off-gases, as shown in Figure *28*.

[96] Arm et al. (2020). "Status of Fast Spectrum Molten Salt Reactor Waste Management Practice"

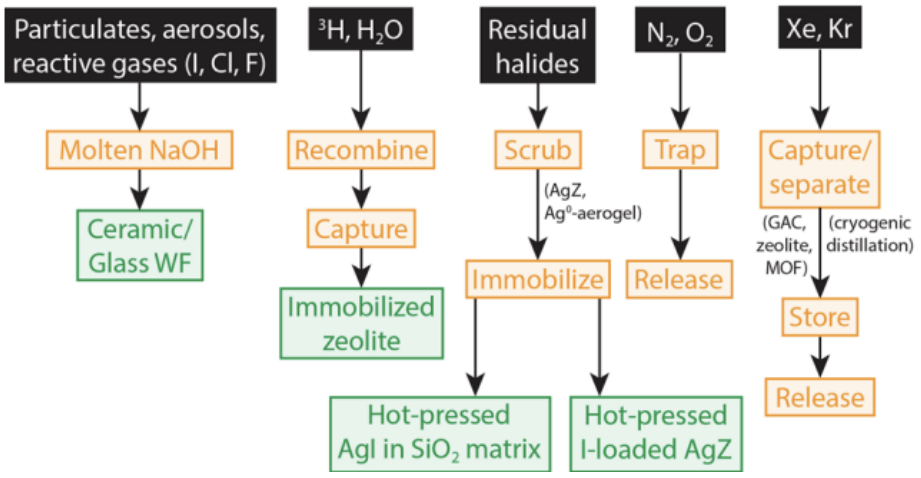

Figure 28: MSR off-gas constituents and planned path forward for capture, immobilization, disposal, and/or release.[97]

Certain particulates, aerosols, and reactive gases like iodine (I_2), chlorine (Cl_2), and fluorine (F_2) could be immobilized in ceramic waste forms. This immobilization process would involve incorporating the radioactive isotopes into stable ceramic matrices to contain the radionuclides and prevent their release into the environment.

Other off-gases, such as tritium, will be more difficult to treat.[98] Several methods to control tritium produced by liquid-fueled MSRs appear viable, but limited experimental data is the primary constraint for designing efficient cost-effective methods of tritium control.[99] Lab scale experiments have been conducted to investigate methods to manage tritium produced in liquid-fueled MSRs,[100] but such experiments show no indication that these methods are developed enough to be used at scale for treatment of tritium produced in commercial MSRs.

Noble gases, such as xenon and krypton, can be stored in an off-gas system and be allowed to decay. However, management practices of noble gas fission products are relatively immature.

All the above processes for dealing with liquid-fueled MSR off-gases necessitate a sophisticated off-gas system capable of capturing, containing, and handling these gases. While such a system is technically feasible, it remains largely conceptual at this point in time,

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[97] McFarlane et al. 2020. "Molten Salt Reactor Engineering Study for Off-Gas Management"
Note: "WF," "AgZ," and "Ag-aerogel" denote waste form, silver mordenite, and silver-functionalized silica aerogel, respectively
[98] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 166
[99] Oak Ridge National Laboratory | Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors
[100] Harrison et al. "Preliminary Tritium Management Design Activities at ORNL"

and initial designs have only been relatively recently considered. Such a design can be seen in Figure *29*.

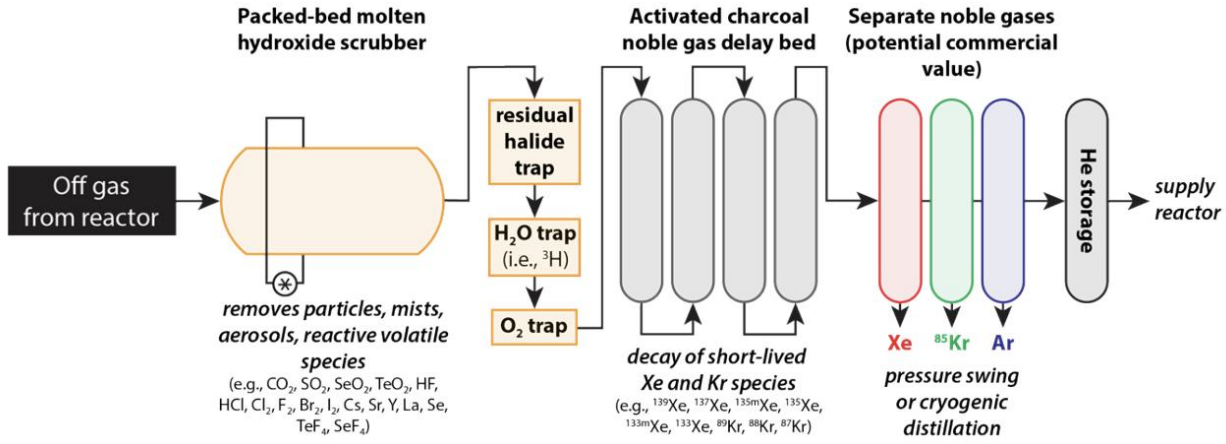

Figure 29: Conceptual design of an off-gas system for MSRs[101]

F.4 Sodium-Cooled Fast Reactors

Sodium-bonded spent metallic fuel

The liquid sodium used as a coolant can bond with metallic fuel in a SFR, becoming an inseparable part of the metallic fuel matrix. Furthermore, interdiffusion between the sodium-bonded metallic fuel and the fuel cladding can produce additional quantities of HLW containing sodium.[102] This poses a unique challenge for permanent disposal due to the reactive nature of sodium.

When liquid sodium comes into contact with water, it reacts exothermically, producing heat and hydrogen gas. This reaction can lead to explosive hazards, which may make direct disposal of sodium-bonded spent metallic fuel in geological repositories difficult. As a result, such waste can be processed before disposal to create a more stable waste form.

The primary method for processing this waste is electrometallurgical treatment, also known as "pyroprocessing", which produces three main types of HLW: (1) uranium; (2) highly radioactive metallic waste; and (3) a highly radioactive salt mixture that can be converted into a ceramic HLW form (i.e., glass-bonded sodalite).[103] This process can be viewed in Figure *30* below.

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[101] McFarlane et al. 2020. "Molten Salt Reactor Engineering Study for Off-Gas Management"

[102] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 164

[103] This treatment method was chosen by DOE for processing wastes from the Experimental Breeder Reactor-II (EBR-II) (Source: 65 FR 56565)

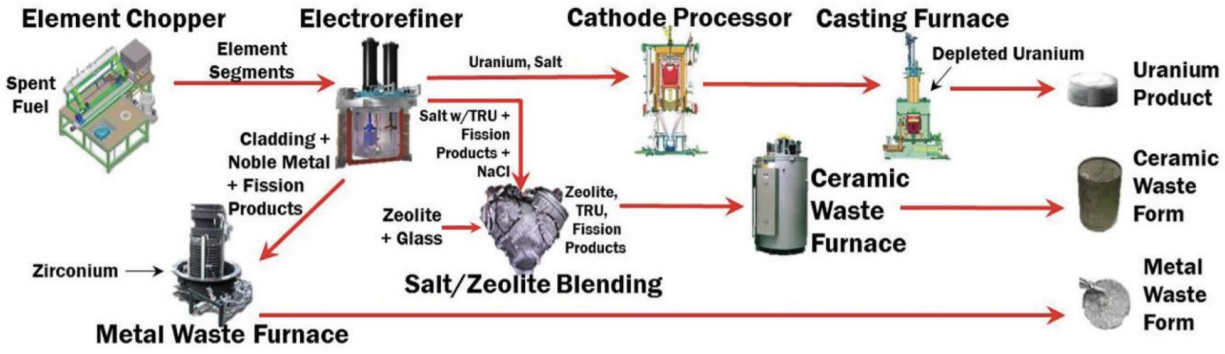

Figure 30: Pyroprocessing Method for Sodium-Bonded spent metallic fuel

A description of this process is as follows:

> "The chopped spent metallic fuels are placed in an anode metal basket and immersed in a 500°C molten LiCl and KCl salt. When current is passed through the metal baskets, fission products and actinides are oxidized and dissolved in the salt bath. The U is reduced to its metallic form and accumulates on the cathode. Cladding and noble metal fission products remain in the anode and can be cast into metal ingots and become metal high-level waste forms. Fission products in the salt bath are first passed through zeolite columns, then mixed with glass and pressed into a glass-bonded sodalite, a ceramic form of high-level waste." [104]

Currently, pyroprocessing can be performed at INL. However, it can only be performed on relatively small quantities of materials because the current technology being used is at "lab-scale".

To pyroprocess the large quantity of sodium-bonded spent metallic fuel that is expected to be generated by just a small number of SFRs, a larger scale pyroprocessing facility will need to be developed. The final design of such a facility and the costs to build and operate it are largely unknown. However, direct disposal of sodium-bonded spent metallic fuel into a permanent repository may be a viable alternative.

While pyroprocessing can help produce a more stable waste form, it may be an unnecessary step in the permanent disposal of sodium-bonded spent metallic fuel because, as discussed in section 5.1, the performance of a geological repository is mostly dependent on the engineered and natural barriers in the geological environment. While direct disposal of sodium-bonded spent metallic fuel without pyroprocessing can increase the probability of interactions between sodium and water in a geological environment (assuming the engineered barrier fails), such interaction could be safely contained and isolated within the

[104] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 165

repository, and therefore may not necessitate pyroprocessing in the first place. That said, further investigation into this topic is needed, with an emphasis on the balance of engineered and environmental containment structures in a repository environment.

Bulk Sodium

Upon decommissioning of a SFR, the sodium coolant, also known as bulk sodium, must be managed. It is possible that this bulk sodium could be reused in subsequent reactors, which would delay the need to properly dispose of it. However, the feasibility of this reuse is currently unclear and would depend on the development and deployment of future SFRs, as well as several economic and technical considerations. If reuse is not feasible, the sodium will likely need to be processed prior to disposal. This bulk sodium, however, would be classified as LLW, and therefore will not require as stringent management strategies as, for example, the sodium-bonded spent metallic fuel.

The processing of bulk sodium from SFR involves several key steps to prepare it for disposal in geological repositories. These processing methods generally focus on neutralizing the highly reactive sodium to create stable waste forms that minimize the risk of chemical reactions and environmental contamination. The primary methods are the "NOAH" process (NOAH being an anagram of the chemical formula for sodium hydroxide, NaOH) and the Argonne process. Prior to undergoing treatment, the bulk sodium is typically pretreated using mechanical filtration to remove any impurities because both the NOAH and Argonne processes are sensitive to such impurities.[105]

The NOAH process involves a highly controlled reaction between bulk sodium and water. Small amounts of liquid sodium are injected into a large flow of water in a closed vessel, resulting in the production of sodium hydroxide and hydrogen gas. This reaction can be represented using the following chemical formula:

$$2\mathrm{Na} + 2\mathrm{H_2O} = 2\mathrm{NaOH} + \mathrm{H_2}$$

Since the quantity of sodium reacting each time is low, the chemical reaction is moderate and continuously controllable. This reaction produces liquid sodium hydroxide, a less reactive substance compared to metallic sodium. The sodium hydroxide is then further neutralized, typically by reacting it with an acid to produce a stable salt solution. Additionally, contaminated radioactive hydrogen gas can be processed further and released in accordance with the applicable radioactive gaseous waste release limits.[106]

The Argonne process, developed and utilized at facilities such as EBR-II and Fermi-1, involves a caustic reaction of bulk sodium with aqueous sodium hydroxide solutions. This method results in the formation of sodium hydroxide monohydrate crystals, which have a concrete-like consistency. The waste streams from this process include sodium hydroxide

[105] IAEA | Radioactive Sodium Waste Treatment and Conditioning
[106] IAEA | Radioactive Sodium Waste Treatment and Conditioning

monohydrate, a solid stable form of sodium hydroxide that can be packed in drums for disposal.[107]

Appendix D: Molten Salt Reactor Experiment

The Molten Salt Reactor Experiment (MSRE) was an 8 MWth DOE test reactor that operated at Oak Ridge National Laboratory from 1965 to 1969 to demonstrate a liquid-fueled molten salt breeder reactor technology, and it is currently awaiting final decommissioning. Unlike traditional reactors, the MSRE used liquid fuel formed by dissolving UF_4 in a carrier salt composed of LiF, BeF_2, and ZrF_4. This fuel salt was circulated through graphite channels within the reactor vessel, providing the necessary geometry and moderation to sustain a nuclear chain reaction. After the reactor was shut down, the salts cooled and solidified into a monolithic mass, with beta and gamma radiation continuously generating fluorine gas.[108]

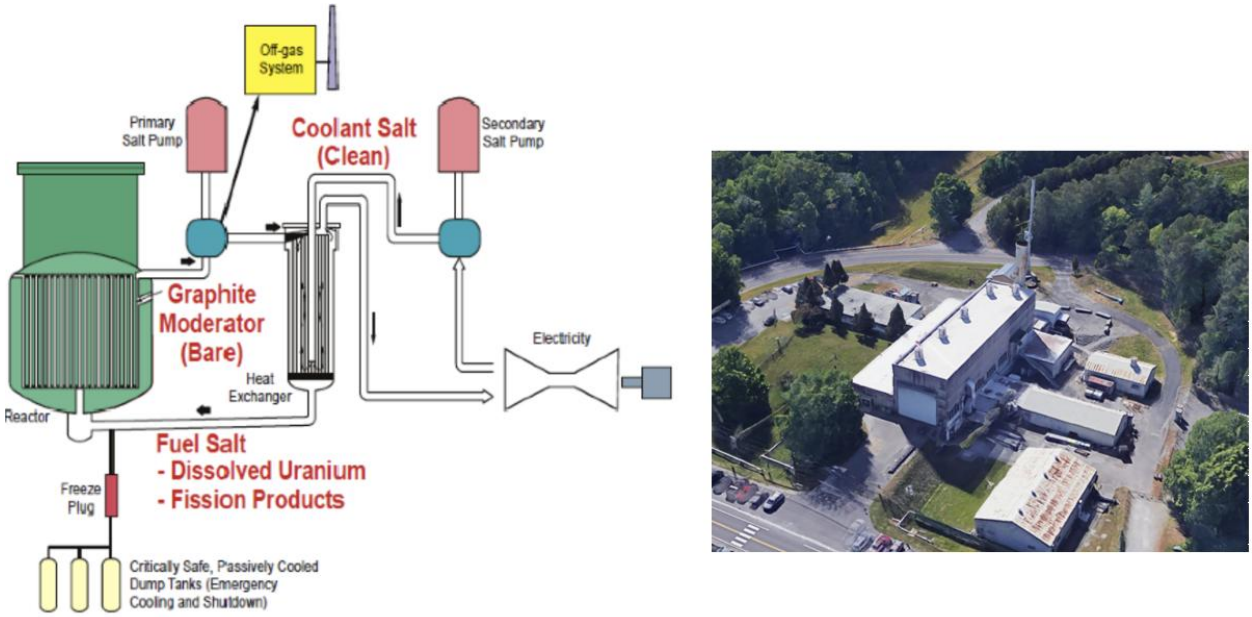

Figure 31: Schematic of the MSRE (left) and aerial photo of the site (right).

The decommissioning of the MSRE site presents significant challenges due to its aging infrastructure and hazardous residual materials.[109] Concerns about uranium migration and fluorine gas buildup have made the decontamination process technically demanding. The site remains in a SAFSTOR state, a strategy where the facility is maintained in a safe and stable condition for an extended period before final decommissioning. This approach provides time

[107] National Academies. (2023). "Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors". pg. 164
[108] Abelquist & Morgan. (2021). "Decommissioning Challenges at the Molten Salt Reactor Experiment Site". *Presentation by UCOR*
[109] Notz. (1988). "Decommissioning of the Molten Salt Reactor Experiment – a Technical Evaluation". *ORNL*.

for radioactive materials to decay, which enhances safety during the eventual decommissioning process. In the meantime, ongoing monitoring is in place to manage risks,

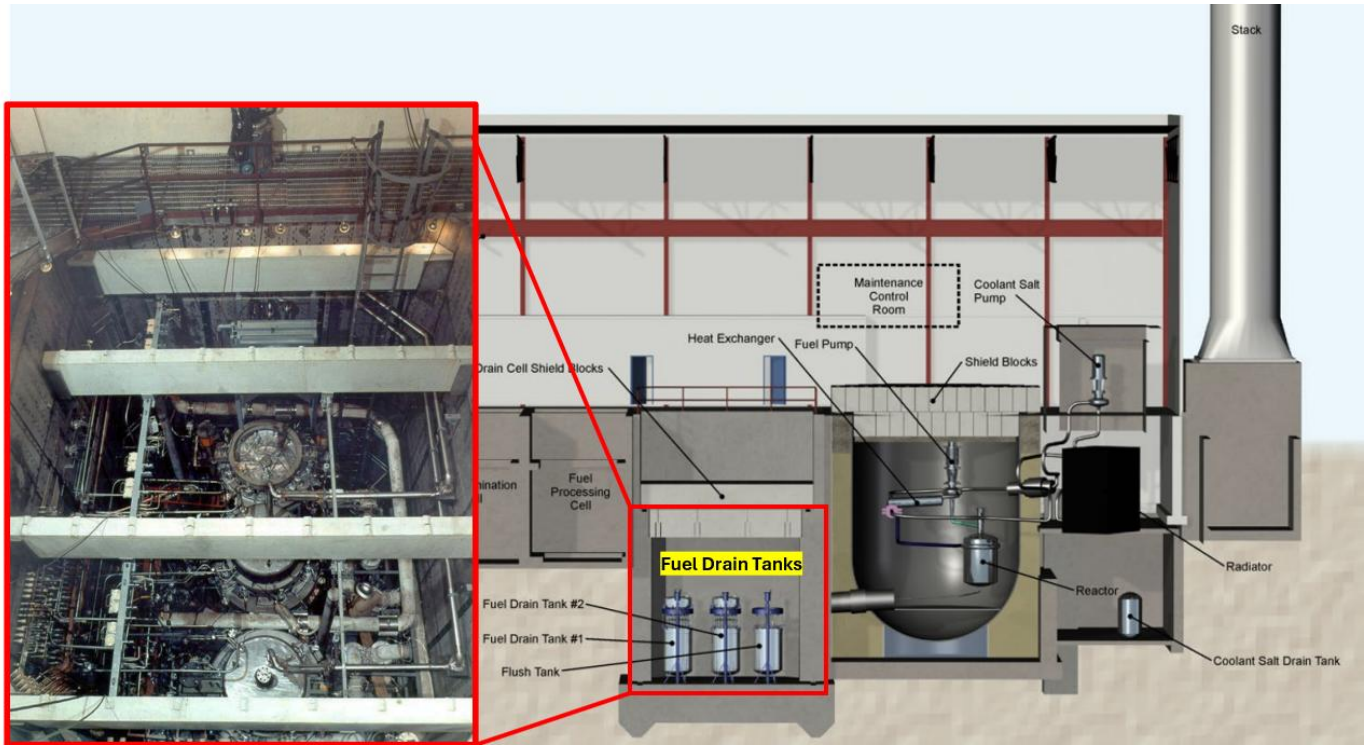

Figure 32: Schematic of the MSRE, including a photo of the fuel drain tanks.

including the potential breach of the three fuel drain tanks that currently hold the high-level waste (HLW). These tanks, located beneath the reactor, were used to drain the molten fuel salt, and now serve as the storage for this HLW. These tanks can be seen in the figure above.

In 1994, unexpected discoveries of uranium hexafluoride and fluorine gas in the reactor's process lines revealed a highly hazardous situation, prompting an evacuation and initiating a complex remediation project. Engineers and chemists have since worked meticulously to remove these materials from the piping and manage the highly radioactive and chemically unstable uranium-233 collected in the charcoal-bed filters. The remediation project has progressed significantly, with most of the UF6 removed and preparations underway to convert the remaining UF_6 into a more stable oxide. The focus is now on managing the highly radioactive fuel salt in the drain tanks. [110]

DOE is exploring disposal options but are facing significant challenges due to the complexities of safely handling and disposing of the radioactive salt, which must be resolved for the site's long-term decommissioning.[111]

[110] Oak Ridge National Lab | Ending the MSRE
[111] Abelquist & Morgan. (2021). "Decommissioning Challenges at the Molten Salt Reactor Experiment Site". *Presentation by UCOR*