

STANDARDS RESEARCH

Evaluation of Canadian Requirements for Management of Waste from Small Modular Reactors

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Executive Summary

This research report provides an overview of the radioactive waste forms expected to be generated throughout the lifecycles of small modular reactor (SMR) design types that are planned or being evaluated for deployment in Canada. It also analyzes the current Canadian regulatory and standards-based requirements for radioactive waste management with respect to waste from SMRs and provides recommendations based on this analysis. The report summarizes the radioactive waste forms expected to be generated throughout the lifecycles of 12 different SMRs from the following design types:

- Light water reactor (LWR)
- High-temperature gas-cooled reactor (HTGR)
- Molten salt reactor (MSR)
- Lead-cooled fast reactor (LFR)
- Sodium-cooled fast reactor (SFR)
- Heat pipes/solid core reactor

In addition, the report presents a review of the CSA N292 Series of standards, and identifies gaps and recommendations related to their applicability to radioactive waste from SMRs.

For the purposes of this report, pressurized heavy water reactors (PHWRs) were considered as the baseline for existing standards and regulations in Canada. The findings indicate that LWR designs produce radioactive waste forms similar to those of PHWRs. HTGR designs produce a few novel waste forms and materials, while MSR, LFR, and SFR designs produce multiple novel waste forms, unique radionuclides, requirements for processing, and other radioactive waste considerations. In all cases, the used nuclear fuel is physically different in size, shape, and mass to current fuel bundles used in Canadian reactors.

The findings also indicate that current Canadian standards are generally broad and include radioactive waste from various sources, including SMRs. Some recommendations are provided for future revisions of the standards to ensure that they fully address waste from SMRs. The key waste features that could be addressed by current standards include:

- Enrichment of fuel, which requires non-proliferation measures, criticality measures, and additional security for highly enriched fuel;
- Graphitic waste, which may require additional consideration due to flammability and explosion hazards, and may require new packaging; and
- Additional toxicity and environmental concerns with liquid lead or liquid sodium-cooled reactors.



"With the expected deployment of SMR technologies in Canada, it is important to review the regulations and relevant standards related to radioactive waste"

1 Introduction

Small modular reactors (SMRs) are nuclear reactors that have a power capacity as small as 10 MW(e) and up to 300 MW(e) [1]. Reactors with a power output less than 10 MW(e) are called microreactors [1]. For the purposes of this report, microreactors are considered under the broad definition of SMRs. SMRs have the potential to fulfill the market need for smaller, simpler, and less costly nuclear power reactors while potentially offering improved safety and performance over traditional full-scale nuclear power reactors. Their modular designs also offer fleet deployment benefits over the larger, more traditional reactor designs.

Several utilities in Canada are planning to deploy SMRs in the near future, including:

- Ontario Power Generation [2]
- SaskPower [3]
- NB Power [4]

In addition, the following proponents are considering implementing an SMR among other options:

- Bruce Power in Ontario is considering SMR options as well as larger nuclear power reactors [5];
- Canadian Nuclear Laboratories is considering the deployment of SMRs for demonstration projects at Atomic Energy of Canada Limited's Chalk River and Whiteshell laboratories [6]; and
- In Alberta, Capital Power is conducting an SMR feasibility study [7], and TransAlta is working with X-energy on a study to assess Xe-100 SMR deployment [8].

At the time of writing, all nuclear power reactors deployed and operating in Canada are pressurized heavy water reactors (PHWRs) of the Canada Deuterium Uranium (CANDU) design, as developed by Atomic Energy of Canada Limited. Waste streams from the commercial operation and decommissioning of CANDU reactors have been studied for decades and are well characterized and understood. The deployment of SMR technology would introduce new waste streams that may not be well understood by the Canadian nuclear power industry [9]. The specifics of SMR designs, operations, materials, and required nuclear fuel cycles must be investigated to understand the details of the waste streams and the requirements regarding waste management. With the expected deployment of SMR technologies in Canada, it is important to review the regulations and relevant standards related to radioactive waste, including the CSA N292 Series, and assess their applicability to support the implementation of various SMR technologies.

This research report provides an overview of the radioactive waste forms that would be generated throughout the lifecycles of SMR design types that are planned or being evaluated for deployment in Canada. It also presents an analysis of the current

state of Canadian regulatory and standards-based requirements for radioactive waste management, and provides recommendations based on this analysis. All phases of the lifetime of a nuclear facility are considered, including site selection, site preparation, construction, commissioning, operation, decommissioning or closure, and additional considerations for design and manufacturing phases. It should be noted that design, manufacturing, construction, and commissioning activities do not generate radioactive waste. Steps pertaining to the management of radioactive waste include generation, treatment, storage, transport, disposal, and decommissioning [10].

2 Methods

A review of public literature was conducted to determine the waste forms that are expected to be generated from the various SMR design types and to analyze Canadian standards-based requirements. Most of the SMR design and waste stream information was obtained from the International Atomic Energy Agency (IAEA), the United States Department of Energy, media releases from prospective utility companies and SMR design companies, and other published reports. Specific details of reactor designs and operations from previous reactor prototypes were obtained from publicly available information and were linked to the related design concept SMRs. Namely, the details of the SMR lifecycles were reviewed in the context of the radioactive waste management steps and requirements outlined in the CSA N292 Series. Communication with SMR vendors to request additional design information was not required as a substantial volume of information on SMR designs was publicly available. The reviewed information was then used to create a representative list of SMR waste features, which was mapped in a matrix to identify each waste feature's inclusion within the CSA N292 Series.

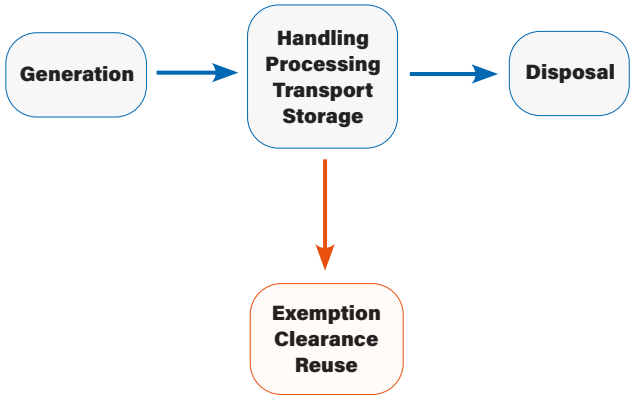
3 Results and Discussion

3.1 Radioactive Waste Management

Radioactive waste is generated through most of the lifecycle of a nuclear facility, such as a nuclear power reactor. The lifecycle includes all phases of a nuclear facility's lifetime: site selection, site preparation, construction, commissioning, operation, and decommissioning or closure. In Canada, all of these phases are subject to the provisions of the *Nuclear Safety and Control Act* [11] and its supporting regulations. It should be noted that while design, construction, and commissioning activities generate negligible radioactive waste, these activities could influence the types and amounts of waste generated during the remainder of the lifecycle. The same lifecycle stages would be applicable to any SMR.

Radioactive waste is managed following the steps shown in Figure 1, and the key steps are generation, handling, processing, transport, storage, and disposal of the waste. The exemption, clearance, and reuse steps refer to the removal of waste from regulatory control, either by virtue of it becoming below exemption quantities or clearance levels (e.g., decay storage, sorting, decontamination, etc.) or by no longer being considered waste.

Figure 1: Steps in the management of radioactive waste [12].



In Canada, generating, handling, processing, transporting, and storing radioactive waste are all the responsibility of the waste generator. In some cases, waste brokers, carriers, processors, or receivers are contracted to manage the respective steps under their jurisdiction. In Canada, disposal of high-level waste (HLW) and intermediate-level waste (ILW) is the responsibility of the waste generator and is managed by

the Nuclear Waste Management Organization (NWMO). Disposal of low-level waste (LLW) is the responsibility of the waste generators and owners. HLW is defined as used nuclear fuel whose owners have declared it as radioactive waste or which generates significant heat through radioactive decay. LLW and ILW refers to all forms of radioactive waste, except used nuclear fuel, limited waste from the production of medical isotopes, and waste from uranium mining and milling [13]. HLW and ILW would be disposed of in suitable geological repositories, to be implemented by the NWMO, and LLW would be disposed of in multiple near surface facilities, to be implemented by the waste generators and waste owners [14].

3.2 Small Modular Reactor Design Types

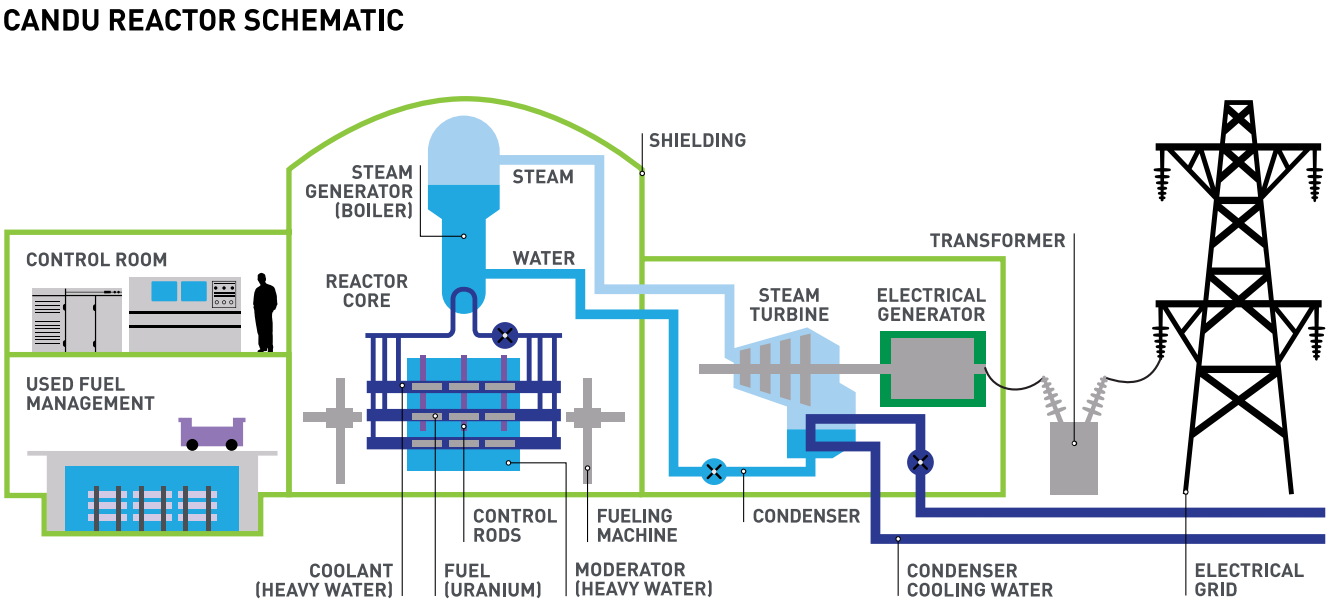
This section provides an overview of the different SMR design types and key definitions related to radioactive waste. Because all of the designs differ, they may produce different waste streams compared to those produced by the CANDU PHWRs currently operating in Canada. The specific SMRs under consideration for deployment in Canada are described in Section 3.3.

The following SMR design types are discussed in this section:

- Light water reactor (LWR)
- High-temperature gas-cooled reactor (HTGR)
- Molten salt reactor (MSR)
- Lead-cooled fast reactor (LFR)
- Sodium-cooled fast reactor (SFR)
- Heat pipes/solid core reactor

An illustration of a CANDU PHWR, which serves as the baseline design against which SMR designs are compared, is shown in Figure 2. Important characteristics of this reactor type are the use of low- or non-enriched uranium fuel, and the use of heavy water as the moderator (activated to tritiated products).

Figure 2: Diagram of a Canada deuterium uranium (CANDU) reactor. Reproduced from [15], © 2020 Canadian Nuclear Association.



LWRs include pressurized water reactors (PWRs) and boiling water reactors (BWRs). A PWR is shown in Figure 3. This reactor type is essentially the same as the CANDU PHWR except light water is used as the moderator. A BWR, shown in Figure 4, is another light water moderated reactor that uses boiling water to drive the turbines without a secondary heat transfer circuit. An HTGR, shown in Figure 5, is a gas-cooled reactor that uses a solid graphite moderator instead of a liquid moderator, a gas coolant, and a secondary heat transfer circuit to provide a hot fluid to drive the turbine. An MSR, shown in Figure 6, can have the fuel dissolved in a molten salt coolant, which passes through a graphite moderator where the fission reaction occurs. A secondary system is used to take the heat from the reactor to drive an electricity generator. An MSR can also be operated using distinct fuel elements isolated from the molten salt, where the molten salt is used to achieve high operating temperatures. Similar to an MSR is a molten metal fast reactor, which uses a molten metal as the primary heat transfer fluid. Two common examples of this type of reactor are the LFR, shown in Figure 7, and the SFR, shown in Figure 8.

Figure 3: Diagram of a pressurized water reactor (PWR). Reproduced from [16] | Office of Nuclear Energy, United States Department of Energy.

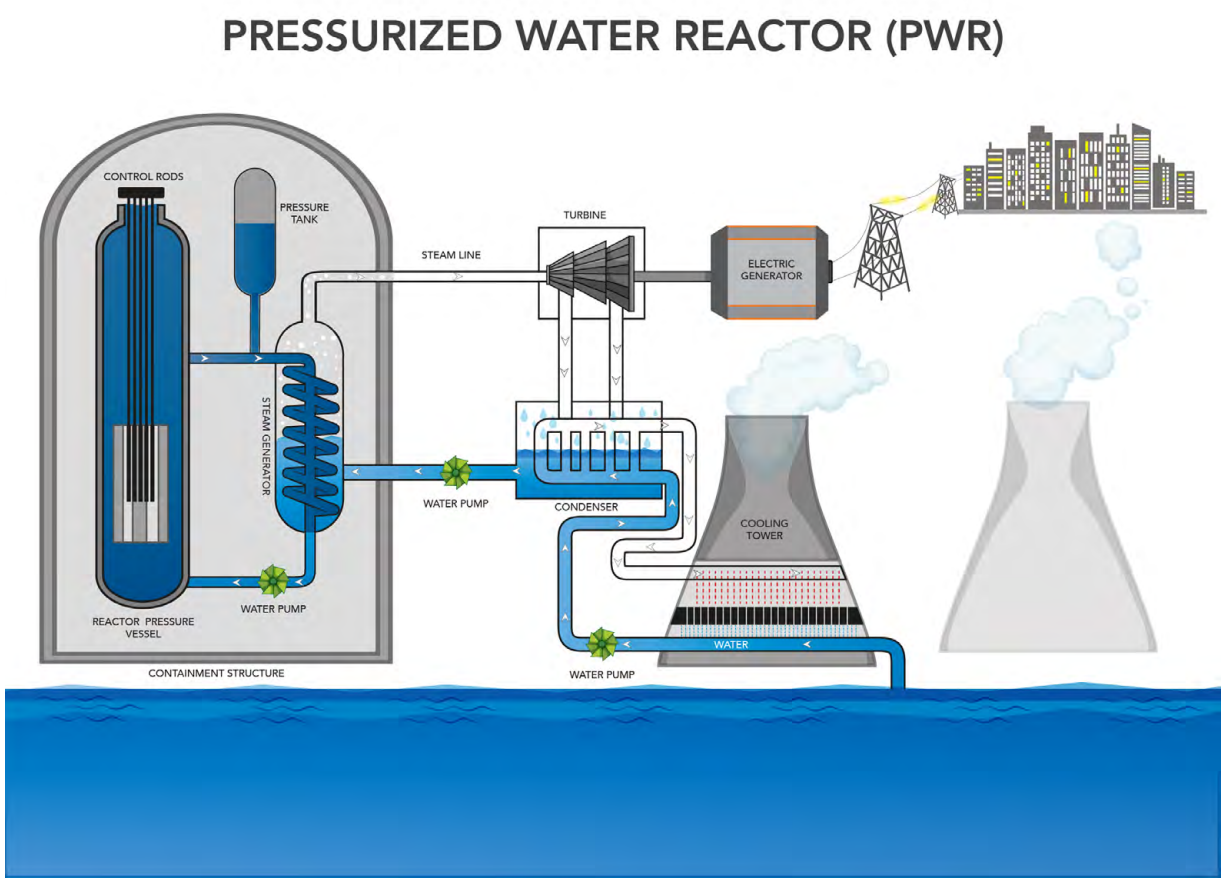


Figure 4: Diagram of a boiling water reactor (BWR). Reproduced from [16], Office of Nuclear Energy | United States Department of Energy.

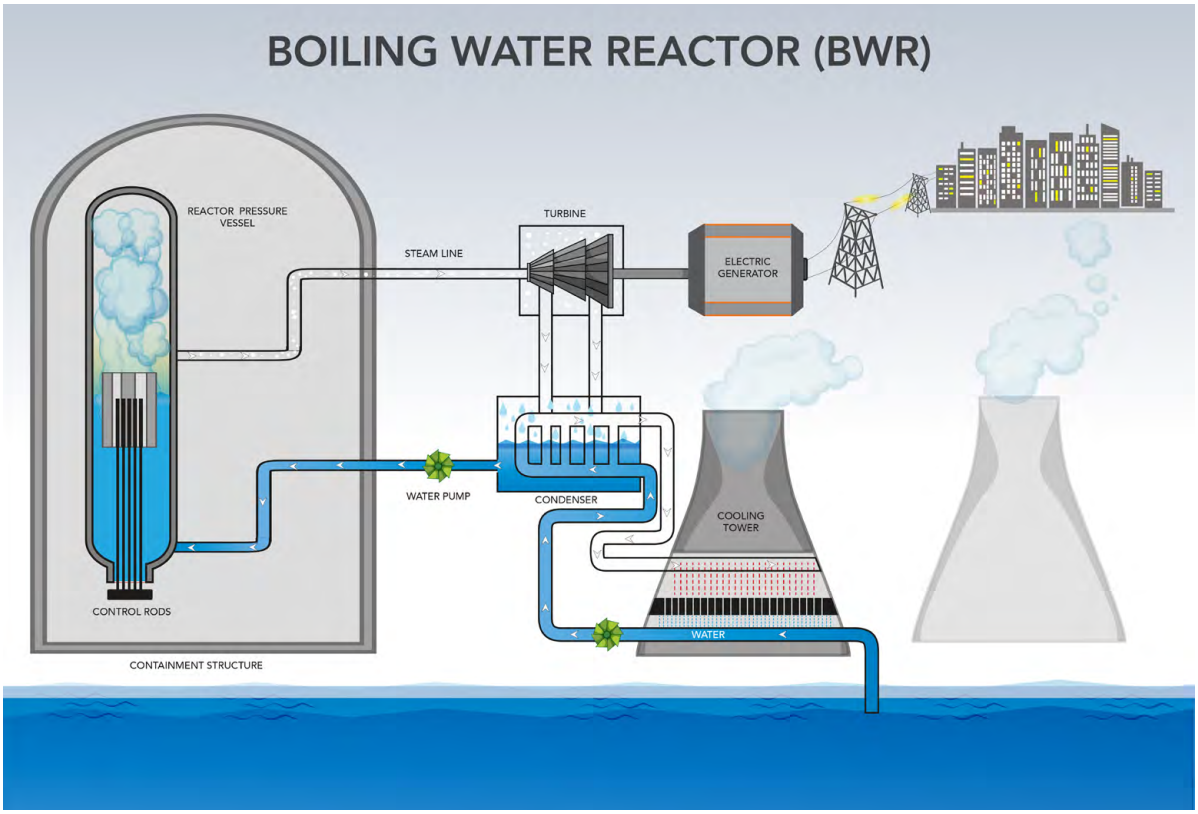


Figure 5: Diagram of a high-temperature gas-cooled reactor (HTGR). Reproduced from [17], EnergyEncyclopedia.com.

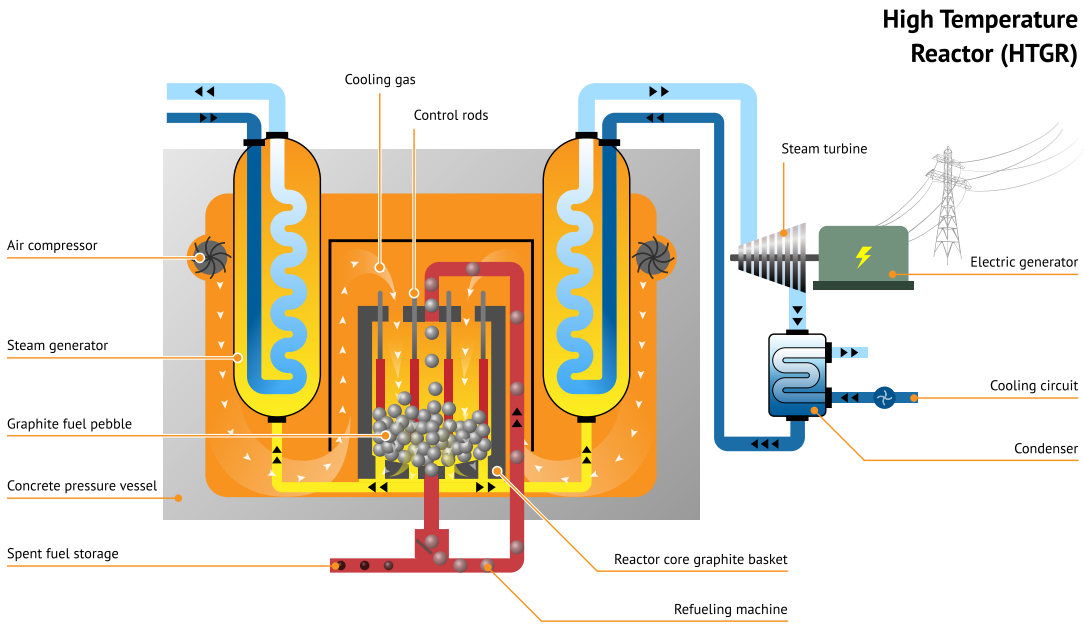


Figure 6: Diagram of a molten salt reactor (MSR). Reproduced from [18], Generation IV International Forum.

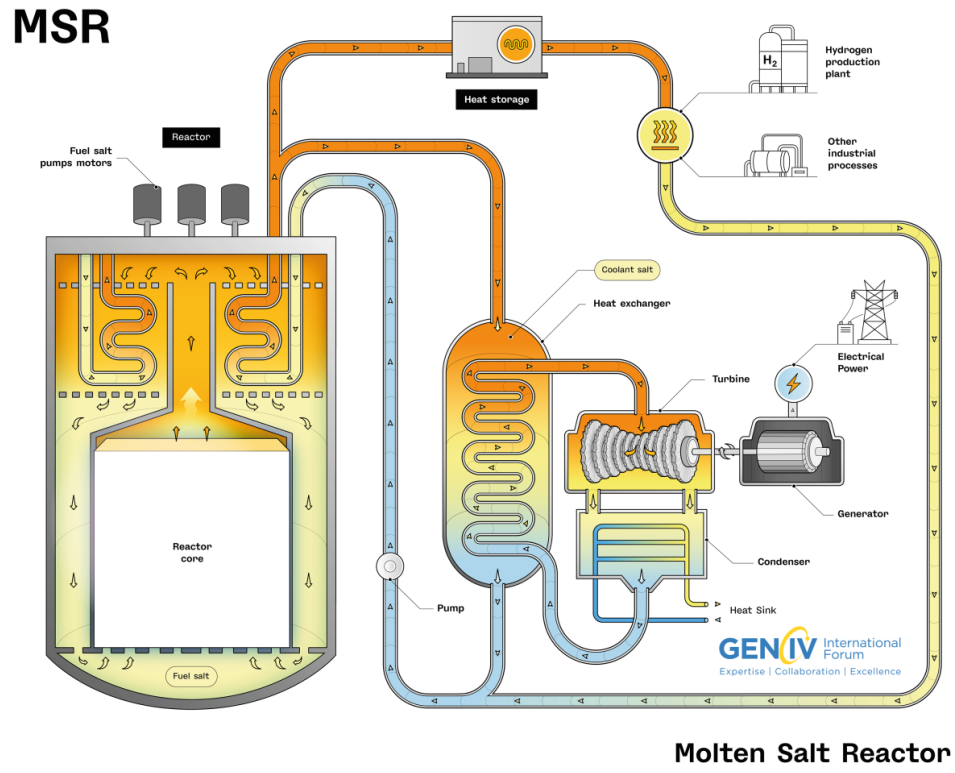


Figure 7: Diagram of a lead-cooled fast reactor (LFR). Reproduced from [19], Generation IV International Forum.

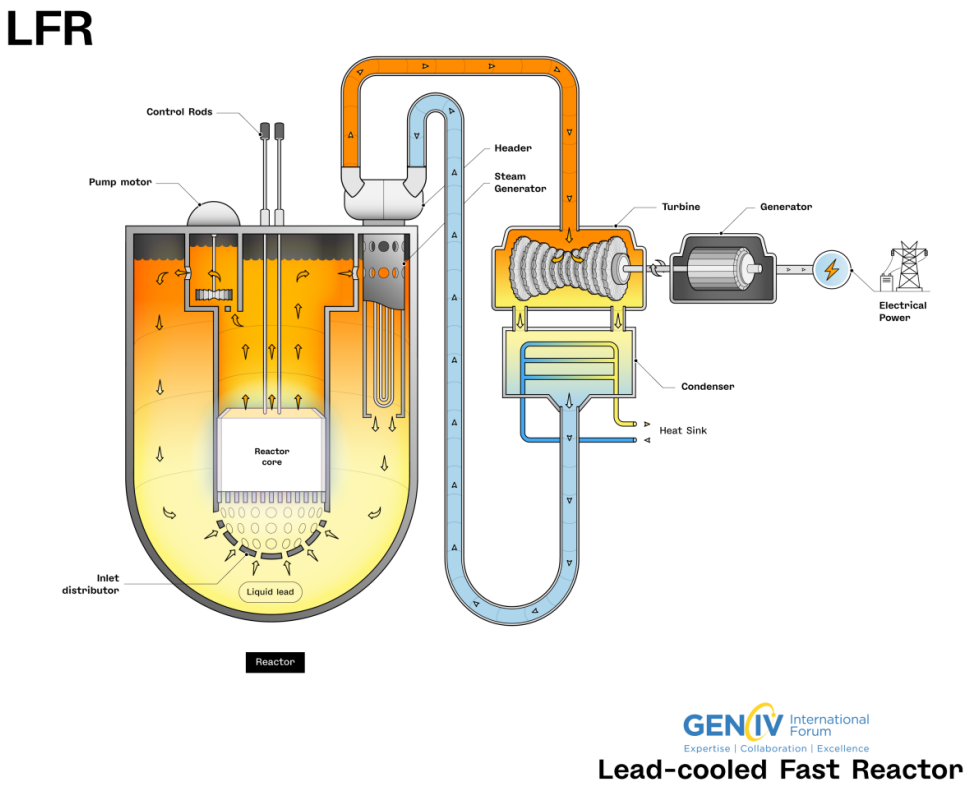
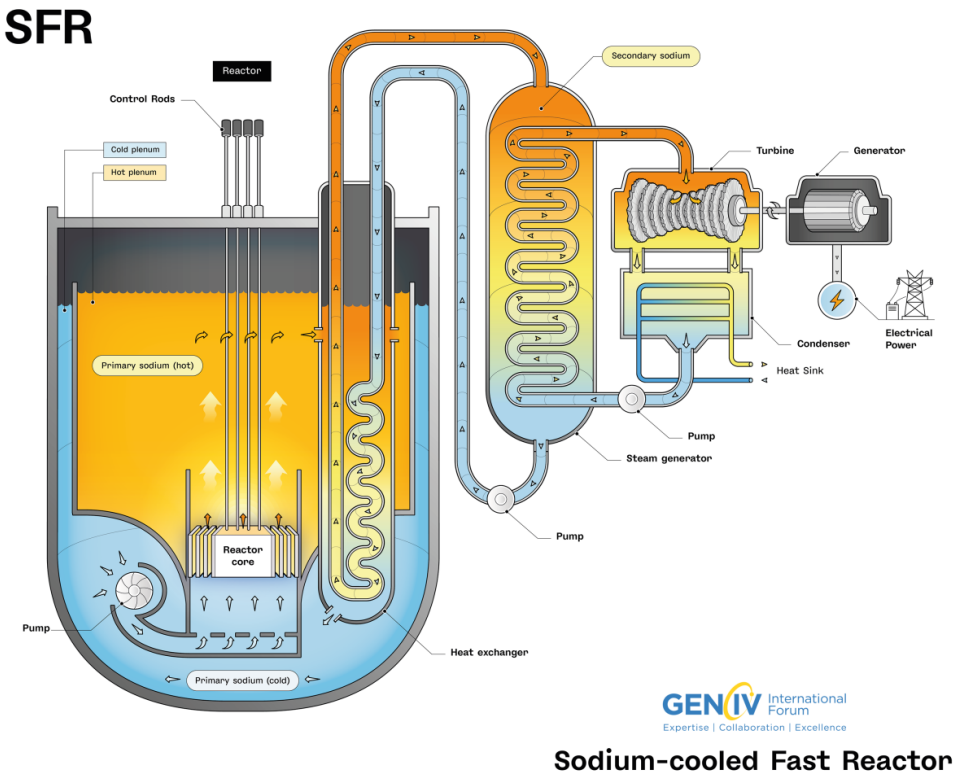


Figure 8: Diagram of a sodium-cooled fast reactor (SFR). Reproduced from [20], Generation IV International Forum.



3.2.1 Terminology and Definitions

This section provides a list of key nuclear reactor components and other important definitions pertaining to radioactive waste generation that are used in this report [21], [22].

Activity: The number of nuclear transformations occurring per unit of time, as measured in becquerels (Bq).

Back-end steps of the nuclear fuel cycle: Steps to ensure that the used nuclear fuel is safely managed, recycled, or disposed of. These steps include fuel storage, recycling, and waste disposal [24].

Closed fuel cycle: Process includes mining, processing, conversion, enrichment of uranium, nuclear fuel fabrication, reactor operation, electrical generation or other energy products, reprocessing to recover fissile material, storage of reprocessed fissile material, reuse of reprocessed fissile material, disposal of highly radioactive (unusable) fission products, and final end states for all waste.

Coolant: The liquid or gas used to transfer heat from the reactor core to the steam generators or directly to the turbines.

Core: The central part of a nuclear reactor containing the fuel and any moderator.

Criticality: A self-sustaining nuclear chain reaction of fissionable nuclear material.

Enrichment: Physical process of increasing the proportion of a particular isotope in a sample of an element, for example, increasing uranium-235 (U-235) concentration in uranium-238 (U-238).

Front-end steps of the nuclear fuel cycle: Steps required to prepare uranium for use as a fuel in a nuclear reactor. These steps may include mining, milling, conversion, enrichment, and fuel fabrication [24].

Fuel: Fissionable nuclear material. In CANDU reactors, this is in the form of fabricated elements for loading into the reactor core of a nuclear power plant or research reactor.

Half-life: For a radionuclide, the time required for the activity to decrease, by radioactive decay, by half.

Isotope: A variation in the form of a chemical element, which is distinguished by the number of neutrons in the nucleus. The number of protons remains the same, but the number of neutrons differ.

Moderator: A material used to slow down or "moderate" neutrons produced by the nuclear fission. The moderator reduces neutron energy by scattering without appreciable capture. Materials of prime concern are those containing light nuclei with large scattering cross-sections and relatively low absorption cross-sections.

Open fuel cycle: Process includes mining, processing, conversion, enrichment of uranium, nuclear fuel fabrication, reactor operation, electrical generation or other energy products, storage of spent fuel, disposal, and final end states for all waste.

Primary side: The primary heat transport system [22]; the reactor side of the system, consisting of the reactor core, primary cooling loop, and steam generators [25].

Radioactivity: The spontaneous transformation of an atom's nucleus by expulsion of particles. Radioactivity can be accompanied by electromagnetic radiation. Solids, liquids, or gases can be radioactive.

Secondary side: The system that transports heat from the primary heat transport system to the ultimate heat sink.

Steam generator: A heat exchanger that transfers heat from primary coolant in order to boil water. Also called a boiler.

Used fuel: Used fuel removed from a reactor after several years of use and treated as waste. Also called irradiated fuel or spent fuel.

3.3 Small Modular Reactors Under Consideration for Deployment

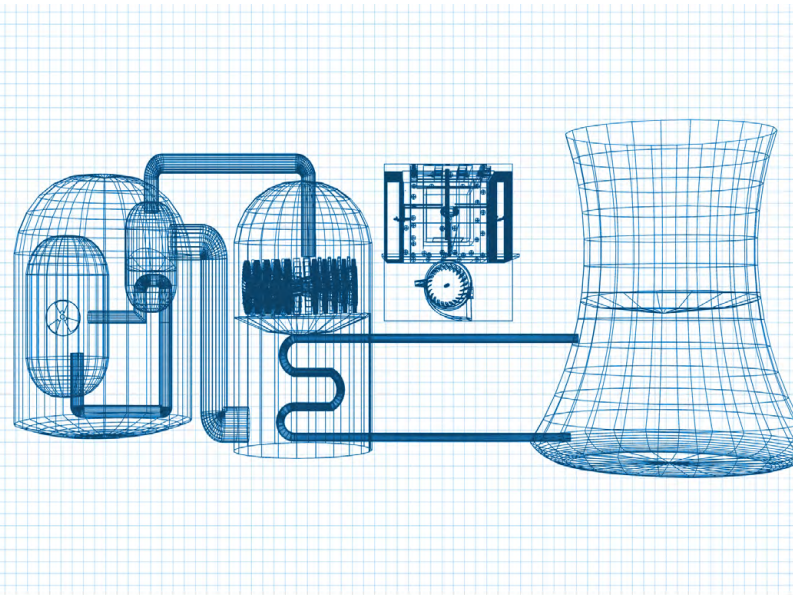
Two SMR designs have been proposed by three utilities for deployment in Canada. One is the BWRX-300, proposed to be built by both Ontario Power Generation [26] and SaskPower [27], and the other is the ARC-100, proposed to be built by NB Power [28]:

- **BWRX-300:** a 300 MW_e BWR design by GE Hitachi Nuclear Energy; and
- **ARC-100:** a 100 MW_e liquid SFR design by ARC Clean Technologies.

In April 2025, a licence to construct a BWRX-300 was awarded to Ontario Power Generation by the Canadian Nuclear Safety Commission (CNSC) [29]. At the time of writing, the following SMR designs, which include microreactor designs, have a vendor design review in place with the CNSC [30]:

- **Micro-Modular Reactor (MMR):** a 3.3 MW_e to 15 MW_e capacity HTGR by Ultra Safe Nuclear;
- **Xe-100:** an 80 MW_e capacity HTGR by X-energy;
- eVinci Microreactor: a 5 MW_e capacity heat pipes/ solid core reactor by Westinghouse Electric Company;
- **SSR-W300:** a 300 MW_e capacity MSR by Moltex Energy;
- **SMR-160:** a 160 MW_e capacity PWR by Holtec International SMR; and
- **IMSR400**: a 200 MW_e capacity MSR by Terrestrial Energy.

The CNSC vendor design review process is an optional feedback mechanism that provides vendors with early feedback on their reactor technology. It is initiated by the vendor prior to submitting a licence application and can help to ensure compliance with Canadian regulatory requirements, expectations, and standards while also alerting the CNSC to any barriers to licensing for new reactor technology that may need to be addressed. There are three phases in a vendor design review [30], [31]:



“Regulatory decisions and licensing are not based on the vendor design reviews, and the two processes are conducted independently.”

- **Phase 1:** Pre-licensing assessment of compliance with regulatory requirements, which involves a high-level assessment of the reactor design against current CNSC design requirements;
- **Phase 2:** Pre-licensing assessment of any potential fundamental barriers to licensing, which identifies any fundamental barriers to licensing the reactor design in Canada; and
- **Phase 3:** Follow-up, which provides vendors with additional information regarding the results from Phase 2.

Regulatory decisions and licensing are not based on the vendor design reviews, and the two processes are conducted independently. During the licensing process, a more detailed assessment of the entire reactor design is conducted through the site preparation, construction, operation, and decommissioning licensing process, which is led by the proponent, typically an operating utility [31].

Although they do not have a vendor design review in place at the time of writing, the following SMR/microreactor designs have been mentioned in previous reports [31], and are included in this report at a higher level:

- **NuScale Power Module (NPM):** a 60 MW_e capacity PWR by NuScale Power;
- **U-Battery:** a 4 MW_e capacity HTGR by U-Battery Canada (no longer an active project);
- **StarCore:** a 10 MW_e capacity HTGR by StarCore Nuclear; and
- **SEALER:** a 3 MW_e capacity LFR by Blykalla (previously known as Lead Cold Nuclear).

It should be noted that the MMR and Xe-100 SMR designs have been given additional attention in this report due to their connections to the Canadian nuclear market, which include:

- A proposed MMR HTGR project by Global First Power, a company previously jointly owned by Ontario Power Generation and Ultra Safe Nuclear, was advancing the micro SMR design for deployment at the Chalk River Laboratories site [32]. At the time of writing, the project is paused.
- Alberta, although still early in the consideration process, is conducting feasibility studies and investing in the Xe-100 [7], [8], [33].
- Ontario Power Generation is working with X-energy to pursue opportunities to deploy the Xe-100 for industrial applications in Canada [34].
- StarCore Nuclear is partnering with the local government district of Pinawa to demonstrate the first StarCore SMR at the Whiteshell Laboratories site [35].
- An eVinci demonstration microreactor has been announced by the Saskatchewan Research Council [36].

3.4 Waste Features of Small Modular Reactors

A summary of the different SMR design types, key waste features, and differences from the baseline CANDU reactors are summarized in Table 1. Different reactor design types have different waste features compared to the baseline, and many of these differences have an impact on the design, operation, and environmental assessments for waste management facilities, or require additional processing, but are covered by existing regulations and standards. Further details on the impacts of these waste differences on existing regulations and standards are discussed in Section 3.7.

Although only some SMR designs may possess a listed waste feature, they can be grouped into the following three categories:

1. **CANDU waste features**
   - Heavy water coolant and moderator;
   - Natural uranium; and
   - Well known and studied waste streams and characteristics.
2. **Slight difference from CANDU waste features (qualitative or quantitative)**
   - Greater nickel-59 (Ni-59) and nickel-62 (Ni-62) activities;
   - Lower tritium (H-3) and carbon-14 (C-14) activities;
   - Different size/shape fuel assemblies;
   - More gaseous waste;
   - More decommissioning waste per Mwe;
   - Some SMR designs include installation of the core below grade; and
   - May not require wet storage of fuel.

Table 1: Radioactive waste from SMR design types. **CANDU** = Canada deuterium uranium, **PHWR** = Pressurized heavy water reactor, **LWR** = Light water reactor, **PWR** = Pressurized water reactor, **BWR** = Boiling water reactor, **HTGR** = High-temperature gas-cooled reactor, **MSR** = Molten salt reactor, **LFR** = Lead-cooled fast reactor, **SFR** = Sodium-cooled fast reactor, **ILW** = intermediate-level waste, **LLW** = low-level waste, **TRISO** = tristructural isotropic.

Reactor Design Type	Reactors	Main Waste Differences from CANDU Reactors
Baseline design: **PHWR**	*Baseline reactor:* CANDU	*Baseline waste features:* ▪ Heavy water coolant and moderator ▪ Natural (non-enriched) uranium fuel as uranium dioxide (UO_2), in cylindrical bundles [37]
LWR Includes PWRs and BWRs	BWRX-300 SMR-160 NPM	▪ Light water coolant/moderator ▪ Boron waste stream in some designs [38] ▪ PWR waste generation is consistent with that from PHWRs; BWR low and intermediate waste generation is about 2–3 times greater for both volume and total activity per one GWe [39] ▪ Greater amount of radioactive decommissioning waste in BWRs because steam production occurs on the primary side (i.e., within the reactor vessel) ▪ PWRs contain more Inconel/Incoloy materials, resulting in greater activities of Ni-59 and Ni-63 in PWR waste streams [40] ▪ Lower H-3 and C-14 ▪ Enriched uranium fuel with higher burnup than CANDU fuel, and therefore hotter (more radioactive and increased decay heat) used fuel ▪ Fuel assemblies are much larger (longer and wider) and heavier than CANDU fuel bundles

Reactor Design Type	Reactors	Main Waste Differences from CANDU Reactors
HTGR	MMR Xe-100 U-Battery StarCore	▪ Graphite moderator will generate large volumes of graphite waste as ILW and LLW [41] ▪ Graphite in fuel particles/pebbles/blocks will generate large volumes of graphite waste as HLW (high level waste). ▪ Inert gas coolant ▪ TRISO fuel kernels are physically very different from CANDU fuel bundles [42] ▪ HTGRs can be prismatic fuel or pebble bed fuel designs ▪ Requires higher enrichment of uranium (up to 20%) [42] ▪ Fuel has higher burnup, resulting in increased decay heat of used fuel
MSR	SSR-W300 IMSR400	▪ Chloride or fluoride salt coolant, both as a primary and secondary (and in some cases a tertiary) coolant [43] ▪ Elevated coolant temperatures will generate gaseous waste ▪ Molten uranium salt fuel is physically very different from CANDU fuel bundles, requiring a new strategy to deal with a homogeneously dissolved fissile material in a large volume of liquid ▪ The molten salt fuel presents different reactivity considerations because it is different from contained fuel bundles/assemblies ▪ More corrosion products [42] ▪ Chemically reactive coolant ▪ Graphite moderator ▪ Requirement to process and stabilize the molten salt ▪ New radionuclides, such as F-18, Na-24, and Na-22 [44]
LFR	SEALER	▪ Liquid lead coolant; chemically hazardous coolant ▪ Decommissioning would result in metallic reactor and lead coolant waste ▪ Fast neutron spectrum will result in increased activation of core components [44] ▪ Greater potential for production of radionucleotides such as Po-210 from neutron activation of the lead [45]; Po-210 is highly toxic due to its alpha decay [46] ▪ No moderator ▪ Uranium nitride fuel, which has different dimensions and configuration from CANDU fuel bundles ▪ Fuel has higher burnup, resulting in increased decay heat of used fuel

Reactor Design Type	Reactors	Main Waste Differences from CANDU Reactors
SFR	ARC-100	▪ Liquid sodium (elemental sodium) coolant ▪ Chemically reactive coolant [44] ▪ Presence of cover gas to prevent contact of sodium with atmospheric air [45], but the cover gas itself can become radioactively contaminated ▪ Uranium-zirconium alloy fuel with enriched uranium (up to 15%) [44] ▪ Fuel has different dimensions and configuration from CANDU fuel bundles ▪ Fuel has higher burnup, resulting in increased decay heat of used fuel ▪ Fuel assembly contains elemental sodium ▪ New radionuclides, such as Na-24 and Na-22 [44]
Heat Pipes/Solid Core Reactor	eVinci	▪ TRISO fuel kernels are physically very different from CANDU fuel bundles [38] ▪ Requires higher enrichment of uranium (up to 20%) ▪ Fuel has higher burnup, resulting in increased decay heat of used fuel ▪ Graphite core block [47] ▪ No liquid coolant

3. **Completely new SMR waste features**
   - Higher burnup and therefore more radioactive, higher temperature fuel, and longer cooling times required;
   - Enriched fuel;
   - Light water coolant/moderator;
   - Boron waste;
   - Graphite waste;
   - Gas coolant;
   - Uranium nitride fuel;
   - Metallic fuel;
   - Tristructural isotropic (TRISO) fuel;
   - Molten uranium chloride/fluoride salt coolant and fuel;
   - Liquid lead or liquid sodium coolant;
   - Chemically reactive coolant;
   - Cover gas waste;
   - New radionuclides, including sodium-22 (Na-22), sodium-24 (Na-24), fluorine-18 (F-18), and polonium-210 (Po-210); and
   - Combustible fuel elements or moderator.

3.5 Small Modular Reactor Waste Streams by Design Type

This section reviews the specific LLW and ILW streams that would be generated by the different reactor design types and the 12 specific SMR designs being considered for deployment. The review considers waste generated during operation, maintenance/refurbishment, and decommissioning, as well as throughout the associated nuclear fuel cycles. The differences from existing CANDU waste streams and waste features that may require special attention are highlighted. Following the discussion of the LLW and ILW streams for each design type is a discussion of the HLW streams (i.e., used fuel waste or fuel processing waste). Each subsection considers fuel materials, structures, any needs for processing the fuel, recycling/reuse, cladding materials, and fuel coatings.

3.5.1 Light Water Reactor Waste Streams

The waste streams generated by LWRs (including PWRs and BWRs) are similar to those generated by CANDU reactors. In terms of reactor design, the two most notable differentiating features of LWRs are their use of enriched fuel and light water coolant. LWR waste generated during operation consists of dry active wastes, and water treatment filters and media, which may consist of ion exchange resins, charcoal, activated carbon, zeolite, sand, gravel, molecular sieves, ceramic filters, and membrane filter elements [48]. The light water coolant/moderator is typically cleaned up so it can be discharged into the environment. LWR waste will typically have lower H-3 and C-14 levels than CANDU waste [48]. Most LWRs use boron, in the form of boric acids, for reactivity control, both in the reactor and in the used fuel pool. This will affect the characteristics of ion exchange resins used for processing the used fuel pool water as well those used for the purification of coolant and moderator water. This results in a boric acid concentrate waste stream that is not present in CANDU waste. As smaller core sizes increase neutron leakage, the maximum burnup of the used fuel is less for small modular LWRs compared to Generation III+ LWRs (although still considerably higher than for CANDU used fuel), and the resulting used fuel masses are higher per unit energy [44].

LLW and ILW from decommissioning consists of reactor component steel, zirconium alloy, concrete, and dry active waste consistent with CANDU waste generated during decommissioning. It should be noted that PWRs contain more Inconel/Incoloy (i.e., nickel-based alloy) materials, resulting in greater activities of Ni-59 and Ni-63 in PWR waste streams relative to CANDU waste [40]. BWRs have a direct cycle, meaning that all the power generation components (including the turbines) are on the primary side and therefore result in larger volumes of activated components such as steel and concrete. However, BWRs do not have steam generators, which account for a significant portion of PWR and CANDU metallic refurbishment and decommissioning LLW. LWR SMRs under consideration require fuel enrichment of approximately 5%, which introduces additional considerations because the baseline CANDU reactors do not use enriched fuel.

The solid fuel form for all LWRs under consideration is enriched UO_2, and the reference fuel cycle is open. Physically, LWR fuel assemblies are much larger and heavier than CANDU fuel bundles and may require some form of size-reduction processing (e.g., sawing, grinding) to meet current NWMO disposal requirements, otherwise a larger used fuel container would be needed. PWR fuel assemblies tend to be larger and heavier than BWR fuel assemblies. LWR used fuel has higher decay heat generation rates due to greater burnup and larger radionuclide inventories. BWRs and PWRs are similar in that they may have different isotopic fingerprints (i.e., ratios of isotopes) than CANDU wastes, but no new radionuclides are expected to be encountered [38]. Additional details of the BWRX-300, the SMR-160, and the NPM SMRs are provided in sections 3.5.1.1, 3.5.1.2, and 3.5.1.3.

3.5.1.1 BWRX-300

The BWRX-300 is a smaller evolution of an earlier GE Hitachi Nuclear BWR design. One key design feature is its use of steel bricks [49], which means large steel brick components would account for part of the waste generated during decommissioning. Decontamination or volume reduction of these components may be a challenge due to their size and shape. Bricks located near the core would also be activated by neutron radiation. However, the volume of decontamination waste may be smaller because the BWRX-300 uses natural circulation [50], which eliminates several large pumps, reducing the volume of contaminated equipment to be removed. Another key feature of this design is the reactor containment structure, which is mostly embedded underground. This introduces greater potential for contamination of the surrounding soil and groundwater, and for corrosion of the reactor building's external structure from exposure to groundwater. Other features specific to BWRs relate to their fundamental design and direct cycle. Having all the power generation components (including turbines) on the primary side results in a wider range of piping and equipment that would be exposed to contamination deposits and a larger amount of waste generated during operation per unit of power generated. Subsequently, more solid waste would be generated during decommissioning. The total activity



of waste generated during decommissioning tends to be two to three times greater in BWRs than in PWRs or PHWRs [39]. The volume of radioactive gases and the need for off-gas processing is also about two orders of magnitude greater for BWRs [51]. The BWRX-300 does not rely on soluble boron for reactivity control [50], thereby minimizing the amount of boron in use, which may result in less corrosion and less media for processing of active liquid wastes. As a BWR, the BWRX-300 will operate with an average enrichment of 3.81% U-235 fuel [52]. Notably, it will operate to a higher core discharge burnup [53] than PWR SMRs.

3.5.1.2 SMR-160

The SMR-160 is a PWR designed to be partially underground, with approximately half of the total height of the containment system located below grade [38]. This introduces the potential for contamination of the surrounding soil and groundwater, and corrosion of the reactor building's structure due to groundwater. After removal of used fuel from the used fuel pool, all the used fuel for the life of the plant can be stored in interim dry storage on-site using a specially designed canister inside an underground storage cask array. The specially designed canister can also be used to transport the fuel off-site with the addition of an overpack [38]. The SMR-160 is designed to operate without boron, which may result in less corrosion and less media used for processing of active liquid wastes. Features specific to PWRs include greater amounts of Inconel/Incoloy materials, resulting in greater activities of Ni-59 and Ni-63 in PWR waste streams.

3.5.1.3 NPM

The NPM PWR is designed to be installed in a pool of water [38], resulting in potentially large volumes of contaminated water requiring cleanup after final shutdown. The core module can be swapped for refuelling, which is done in another part of the pool [45]. Similar to the SMR-160, the NPM may result in greater activities of Ni-59 and Ni-63 than typically encountered in CANDU waste streams, which may be by one to two orders of magnitude [40], [54].

3.5.2 High-Temperature Gas-Cooled Reactor Waste Streams

HTGRs are typically graphite-moderated and helium-cooled. Neither of these materials are typically encountered in CANDU reactor waste. The graphite tends to become activated, with the key activation products being C-14 from the graphite material, and H-3 and chlorine-36 (Cl-36) from impurities [44]. Large volumes of decommissioning graphite waste can be expected for any HTGR. Depending on the proximity of the graphite blocks to the fuel, the graphite may become LLW or ILW. Although HTGRs do not produce new radionuclides relative to CANDU waste, their activities are distributed in different waste forms. For example, as indicated in [48], a CANDU reactor has a higher overall production of C-14 [10 125 GBq•GW(e)$^{-1}$•a^{-1}] than an HTGR [3 371 GBq•GW(e)$^{-1}$•a^{-1}]; however, the majority of HTGR C-14 is embedded in the graphite whereas the majority of the CANDU C-14 is in the heavy water coolant and moderator and the associated purification

system ion exchange resins. Furthermore, because graphite is combustible in air, the combination of graphite dusts and air can present an explosion hazard [55].

Additional considerations with packaging, storage, and disposal of graphite waste include the release of Wigner energy [56], which is the accumulation of energy from the knock-on effect of lattice defects introduced into the graphite by neutron irradiation. Graphite is known to be susceptible to these impacts, so the amounts of Wigner energy and the kinetics of its release must be well understood when considering packaging, storage, and disposal. Failure to manage the Wigner energy can result in runaway exothermic reactions, such as experienced by the Windscale pile fire in the United Kingdom in 1957 [56].

While the helium coolant in HTGRs is inert and not activated, it can pick up small amounts of radioactive gases that escape from the fuel along with radioactive contaminants from the walls of the cooling channels, thereby becoming contaminated. It is anticipated that the helium coolant can be processed and the contamination would be captured in absorbent media (e.g., filters, charcoal, ion exchange resin, etc.), cleaning up the helium for free release [57].

Another important consideration in HTGRs is the use of coated particle fuel (TRISO fuel). Two types of TRISO fuel are pebble bed and prismatic block, both of which usually contain tiny uranium oxycarbide (UCO) particles coated with silicon carbide and then an additional layer of carbon [45]. The silicon carbide-coated fuel is considered a high-assay low-enriched uranium (HALEU), requiring enrichment between 5% and 20%. The TRISO-coated fuel particles are embedded within a graphite fuel sphere or compact (block). Opening the fuel cycle for TRISO-bearing fuels is not expected in the near future and the conventional disposal method would be to place intact TRISO-bearing pebbles or prismatic blocks in the repository, without removing the TRISO particles from these pebbles or blocks. Additional features specific to the MMR, the Xe-100, the U-Battery, and the StarCore SMRs are provided in sections 3.5.2.1, 3.5.2.2, 3.5.2.3, and 3.5.2.4.

3.5.2.1 MMR

The MMR is an HTGR with TRISO fuel in prismatic graphite blocks in a sealed transportable core [43]. It is a dual loop system, with helium as the primary coolant and molten salt as the secondary heat transfer fluid [43]. Therefore, all the waste features of helium-cooled HTGRs discussed in Section 3.5.2 need to be considered, as well as MSR features that are discussed in Section 3.5.3. Specifically, the molten salt mixture would likely contain activation products (i.e., Na-22, Na-24, F-18, and C-14) and corrosion products from the surrounding process equipment (i.e., cobalt-60 [Co-60], Ni-59, Ni-63, iron-55 [Fe-55], and iron-59 [Fe-59]), and would require an off-gas treatment as well as a processing system to process a portion of the salt for removal of unwanted constituents [46]. Both processes would generate filters and media waste. Upon decommissioning, the molten salt mixture would need to be processed or immobilized (e.g., by vitrification in a glass matrix) [42]. The processing operations would also result in new waste streams such as contaminated components and structures, sludges, and contaminated media. Although these waste forms already exist from CANDU reactor operations, the waste streams resulting from HTGR operations would contain different isotopic fingerprints. The MMR is expected to be fuelled as an entire prismatic fuel block [38]. The large physical size of the block core introduces challenges for decommissioning and physical handling.

3.5.2.2 Xe-100

The Xe-100 is an HTGR with a moderator made up of graphite blocks, helium coolant, and TRISO pebble bed fuel. It would produce a large volume of graphite waste (up to 100 m^3 [57]), considering the graphite moderator and reflector blocks. A common industry practice to reduce the impact of graphite waste is regular outage maintenance to replace impacted graphite blocks, which reduces their time in the reactor and diverts as many graphite blocks as possible to LLW. The helium coolant is intended to be cleaned up through online processing, thereby generating filters and media waste. The helium itself would be released from the closed loop

as non-radioactive waste upon decommissioning [57]. In addition to filters and other process media contaminated with graphite activation products and gaseous impurities, the waste generated during operation will include radioactive graphite dust, collected by a dust and aerosol removal system. This active graphite dust is intended to be packaged in a storage bin with no further processing. Fine graphite dust may be pyrophoric, and provisions would need to be made to handle it in an inert atmosphere. As mentioned previously, very small amounts of fission products are expected due to self-containment by the TRISO fuel pebbles [42]. The TRISO-containing fuel pebbles containing silicon carbide and graphite-coated UCO fuel are designed for online refuelling, and the spent fuel is sent to a special fuel storage canister destined for dry storage [42]. No wet storage or fuel bay is required.

3.5.2.3 U-Battery

The U-battery is an HTGR with a fuel block design, which requires refuelling every five years [43]. Although designed for UO2 TRISO fuel, it includes the possibility of using thorium with a beryllium oxide reflector. It is designed to be cooled by helium but includes the possibility of being cooled by carbon dioxide, which would generate even greater amounts of C-14. If beryllium oxide is used as the reflector, the handling, processing, treatment, and disposal of the reflector would need to consider the highly toxic nature of beryllium.

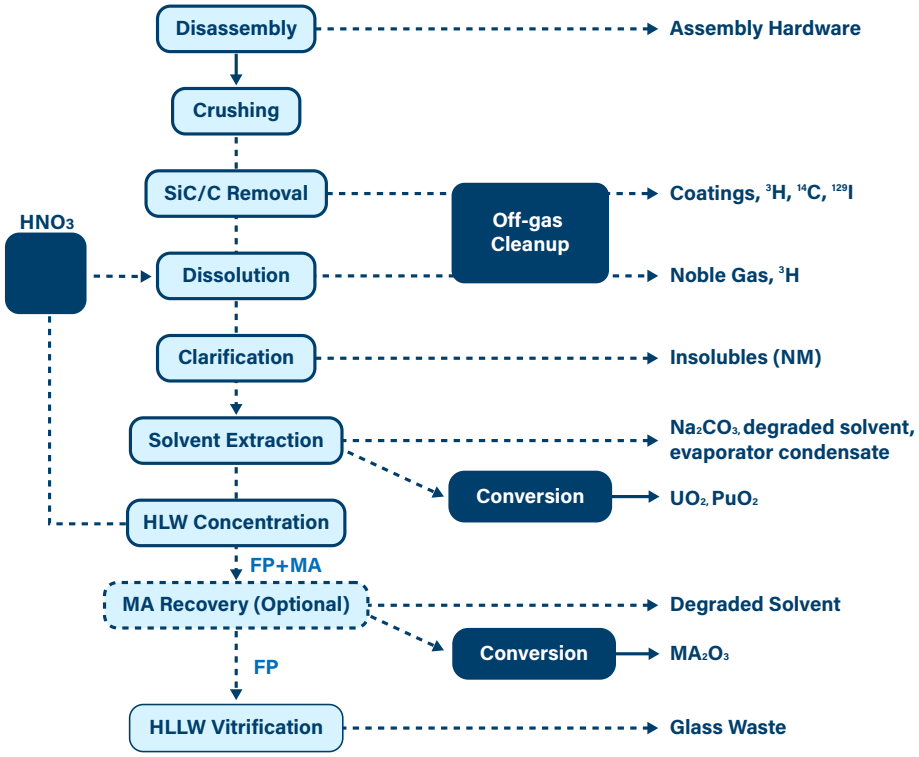
3.5.2.4 StarCore

The StarCore is an HTGR using TRISO fuel in a prismatic matrix [43]. Secondary loop cooling is by nitrogen, and it is assumed that the secondary loop is far enough from the core to avoid activation so additional generation of C-14 is minimized. The reactor steel-walled concrete silo structure would be installed partially below grade. This introduces the potential for contamination of the surrounding soil and for corrosion of the reactor building structure from exposure to groundwater. Refuelling is planned every five years, when the entire graphite fuel prismatic block is removed from the reactor pressure vessel and replaced. The entire block would be stored for 12 months in an on-site underground fuel storage silo before being transported to a permanent repository [38].

3.5.2.5 High-Temperature Gas-Cooled Reactor Fuel and other High-Level Waste

Fuel recycling/reuse for these HTGRs is expected to be complex with the current state of technology. The ceramic (silicon carbide) encapsulation of the TRISO fuel particles is expected to make access to the fuel kernel itself (i.e., the UO_2, UCO, uranium nitride, or ThO_2 material) difficult. In an IAEA report, TRISO fuel was deemed “less attractive for reprocessing” [41, p. 77]. However, the report also noted that both aqueous and non-aqueous processes are being developed (at the conceptual stage) to remove the coatings and treat the fuel kernels. This potential was also explored in the Nuclear Fuel Cycle Science and Engineering textbook [58]. The IAEA report also noted that TRISO fuel has a “large volume for final disposal” [41, p. 77], which is due to the comparatively low density of fissile material in the fuel element. Due to physical changes in the graphite caused by irradiation, the number of fuel cycles for the prismatic block is expected to be limited to three or four cycles [41], although the Xe-100 concept uses an average of six fuel cycles per pebble. Ultimately, the fuel elements will need to be disposed of; however, the relatively high cost of reactor grade graphite may eventually justify reprocessing of the graphite [41]. Oxidizing the graphite has also been proposed as a means to reduce waste, allowing the storage of TRISO particles without the need for their bulky compacts [59], but criticality implications would need to be considered. The fuel is considered suitable for both long-term storage and disposal in a repository [60]. The TRISO fuel particle coatings provide a very strong barrier to contain radionuclides during storage and geological disposal of used fuel. The corrosion rates of TRISO coatings are very low in both wet and dry conditions [41]; the measurements indicate that TRISO particles should maintain their integrity for a million years or more in a geological repository. However, more research is necessary to confirm the durability of the coatings, as other mechanisms for coating failure have been identified (e.g., pressure vessel failure, silicon carbide thermal deposition failure, and kernel migration failure) [61]. It was noted that if the reactor uses thorium, as proposed for the MMR and the U-Battery, actinide production is estimated to be lower than for a PWR [41]. It was also noted that

Figure 9: Process flow and primary waste stream for aqueous HTGR processes. **NM:** noble metals, **MA:** minor actinides, **HLLW:** high-level liquid waste, **HLW:** high-level waste, **FP:** fission product, **UDS:** undissolved solid. There are more types of waste produced from the HTGR aqueous process compared to the LWR. Reproduced from [41] with permission, © IAEA.



there is no waste in the form of activated metals from the fuel as there are no irradiated claddings. A proposed recycling route, reported in [41] and [58], is presented in Figure 9. The primary waste challenges involve the production of a large amount of activated graphite, and a significant amount of C-14 from the activation of carbon-13 (C-13) and nitrogen-14 (N-14). It was also noted in [41] that removal of the carbon and silicon carbide coatings will produce a large amount of carbon waste, or an additional fractured coating waste stream.

Figure 9 shows that the primary waste streams from recycling include coatings (contaminated by H-3, C-14, and iodine-129 [I-129]), noble gases and gaseous H-3 and HTO (tritiated water), noble metals, sodium carbonate, UO_2, and plutonium oxide (to be reused as reactor fuel), degraded solvents, minor actinides in the form of MA_2O_3, and glass waste. For more details, see Table 13 (Case 4) in [41].

3.5.3 Molten Salt Reactor Waste Streams

MSRs are a type of breeder reactor that can operate in the fast neutron spectrum, which does not require a moderator. However, some MSRs operate in the thermal spectrum, which does require a moderator. MSRs generally operate at or near atmospheric pressure. The fuel, fertile material, and coolant may be mixed together in one homogeneous salt mixture fluid, or a molten salt fuel may be separate from a molten salt coolant mixture. At an elevated temperature, this mixture of various salts melts and becomes a clear, nonviscous fluid [62]. Molten salts used are fluoride or chloride based; some contain lithium, beryllium, sodium, zirconium, or potassium, with the most common being lithium fluoride and beryllium fluoride mixtures, known as FLiBe or 2LiF-BeF_2 [45]. The MSR design type is very different from the baseline CANDU reactor, and therefore has several novel waste streams.

MSR operations would generate waste in the form of dry active LLW, such as filters, personal protective equipment, and other consumables, and ILW, such as media used to process off-gases and the gas products themselves. In addition to activation products, fission products, such as gaseous xenon and cesium isotopes, would be released from the molten salt mixture, necessitating an off-gas system [44]. ILW would likely arise from the media wastes generated by processing the off-gases and the processed gas waste. H-3 gas is also likely to be generated from the neutron irradiation of lithium and processed by the off-gas treatment. The trapping of fission and activation products would likely require ceramic filters and vitreous carbon [46], which would generate ceramic and vitreous carbon ILW waste streams. If the gases cannot be purified for release into the environment, they may require absorption onto a solid media for storage as an LLW stream. Secondary coolant molten salt would also become activated and require similar off-gas treatment. Furthermore, the secondary loop may be diverted to a processing system to process a portion of the salt for removal of impurities [46]. Due to the corrosive nature of molten salt, elevated levels of corrosion products from the component materials are expected. This would generate additional filter and media waste (ILW or LLW) from the processing operations. In addition to these wastes, the operational activities may be labour intensive, resulting in high volumes of LLW consisting of personal protective equipment and other consumables [46].

MSR decommissioning would generate waste from dismantled reactor components and the salt fuel mixture. Due to the homogeneous distribution of fuel, activation products, and fission products in salt fuel mixtures, the entire mixture and surrounding materials would become activated. The reactor vessel itself would be activated and contaminated by fission products, corrosion products, and precipitates from the salt mixture. This is expected to generate large volumes of reactor component waste as ILW, up to four times the volume for similarly sized water-cooled reactors [57]. Given the corrosive nature of the salt fuel mixture, most components are expected to be fabricated from nickel-based alloys, which would increase the amounts of Ni-59 and Ni-63. In addition to being radioactive, the salt mixture is chemically reactive with a tendency to generate explosive or corrosive by-products upon contact with air or moisture (e.g., hydrolysis to produce hydrofluoric acid) [44]. It may be necessary to create new waste packages to contain the corrosive salt mixture or new treatment processes to render the salt non-corrosive. Additional major challenges are the formation of volatile uranium hexafluoride (UF_6) complexes by radiolytic decomposition and avoiding criticality. The latter was a challenge posed by the solidified fuel salt stored at Oak Ridge National Laboratories following the shutdown of its MSR experiment in 1969 [44]. For this reason, the industry recommended approach is to process the actinides from the salt mixture into a more stable ceramic waste form [44].

The processing required for treatment of the salt fuel mixture is likely to be pyrochemical, and still needs to be developed in detail [41]. The processing would result in three streams: a) an HLW stream of residual actinides and used fuel in a stable ceramic form; b) a stream of the remaining activated fluoride salts, also containing fission products and corrosion products from the surrounding structures; and c) the remaining salt mixture, likely to be ILW. The last stream will need to be immobilized, for example, by vitrification in a glass matrix [42]. Unlike secondary CANDU waste that is mostly contaminated with activation and corrosion products, MSR processing operations would also result in new waste streams, such as contaminated components and structures, sludges, and contaminated media, all of which would likely be contaminated with high concentrations of fission products and residual actinides. An alternative to processing the salt fuel mixture would be to heavily neutron-poison the mixture and pump it into shielded, subcritical geometry decay canisters. As the canisters cool, the salt fuel mixture would solidify [42], creating one large HLW stream instead of the processing option, which would generate a smaller HLW stream and an ILW stream. With either option, used fuel management will require new packaging, on-site storage, possible off-site interim storage, and transport capabilities.

The activation products generally encountered from FLiBe are F-18, Na-24, Na-22, chromium-51 (Cr-51), Fe-59, Fe-55, cobalt-58 (Co-58), Co-60, H-3, and C-14 [44]. The F-18 isotope is a lower consideration due to its

short half-life (109 minutes) [63]. Beryllium-10 (Be-10), Na-24, and Na-22 radionuclides are not encountered in CANDU waste in large quantities. Novel materials and radionuclides, such as protactinium-233 (Pa-233), which is very reactive with oxygen and water, are possible intermediate products in a thorium MSR [41]. Pa-233 also decays into U-233, which is a proliferation risk. Its presence and processing would need to be considered in any plans to use thorium fuel in an MSR design. Additional features specific to the SSR-W300 and the IMSR400 are provided in sections 3.5.3.1 and 3.5.3.2.

3.5.3.1 SSR-W300

The fuel salt for this reactor is approximately 60% sodium chloride (NaCl), 20% plutonium chloride ($PuCl_2$), 20% uranium (III) chloride (UCl_3), and the coolant salt is a zirconium fluoride, potassium fluoride, and sodium fluoride (ZrF_4-KF-NaF) mixture [43]. The reactor is fuelled with low purity, reactor grade plutonium reprocessed from stocks of spent uranium oxide fuel [38]. Unique features of the SSR-W300 include the fuel being held in vented tubes instead of mixed in with the coolant salt. Noble fission products escape from the vented tubes into the coolant while other fission product gases condense on the upper fuel tube walls and fall back into the fuel mixture [43]. This feature minimizes the challenges associated with processing the fuel-coolant mixture post decommissioning. It also allows for online refuelling, where fuel assemblies are moved out of the core and replaced with fresh fuel assemblies on-site without the need for specialized refuelling facilities [64]. The use of sodium chloride and other chloride salts would trigger generation of Cl-36, Na-22, and Na-24 radionuclides. Because Na-22 and Na-24 are not generally encountered in notable quantities in CANDU waste, they must be considered. Moltex Energy's Waste to Stable Salt technology would generate fresh fuel from used fuel using a pyrochemical process [42]. The products of this processing are new waste streams, which are described in Section 3.5.3.3. Considerations would need to be given about whether the Waste to Stable Salt processing is co-located with the SSR-W300 or established at a different location. Packaging and transport also need to be considered for either option. The majority of the reactor material is zirconium-based to avoid corrosion problems associated with steel [64]. This will result in large amounts of zirconium decommissioning ILW, with increased quantities of long-lived zirconium-93 (Zr-93) (1.53×10^6 y half-life).

3.5.3.2 IMSR400

The IMSR400 uses graphite as a moderator and a fluoride salt mixture as the fuel. Uranium tetrafluoride (UF_4) fuel is dissolved in fluoride salts without lithium or beryllium, minimizing the production of H-3 and Be-10. The core is an integral unit, fully replaced every seven years [38]. The secondary loop coolant is a mixture of zirconium fluoride and potassium fluoride (ZrF_4-KF) [43]. Both the primary coolant-fuel and the secondary coolant would need to be processed, creating additional waste streams as described in Section 3.5.3. Vessel construction would use a nickel-based alloy, Hastelloy N, to avoid corrosion problems associated with steel. However, because Hastelloy N suffers from helium embrittlement, the manufacturer envisions a 7-year lifetime for the reactor vessel. Consequently, as much as 1 m^3/GW_{th}-y of nickel alloy LLW and/or ILW waste would be generated from decommissioning each vessel [44]. The graphite moderator would likely result in additional large volumes of ILW. High C-14 activities are expected from activation, in addition to contamination from the salt. For example, the graphite from the Oak Ridge MSR experiment acquired surface contamination by molybdenum, tellurium, ruthenium, and niobium fission products, whereas tritium, cesium, and strontium radioisotopes (generated via decay of xenon and krypton fission products) diffused into the porous structure of the graphite [44]. The reactor silo structure would be installed below grade, introducing the potential for contamination of the surrounding soil and corrosion of the reactor building structure from exposure to groundwater.

3.5.3.3 Molten Salt Reactor Fuel and other High-Level Waste

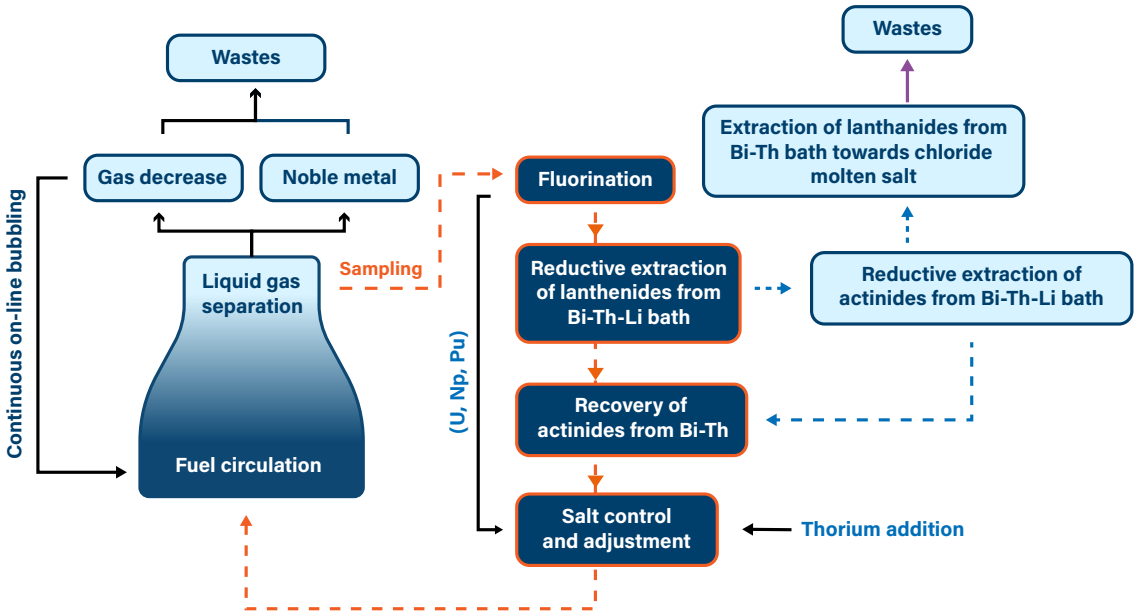
This section describes the HLW streams generated by the SSR-W300 and the IMSR400. In both of these reactors, the fuel is in the form of a molten salt [58]. In the IMSR400, the fuel flows through the reactor core, whereas in the SSR-W300, the molten salt fuel remains fixed within fuel carrier tubes. Continued

research is required on the pyroprocessing of fluoride salts containing nuclear materials (including low concentrations of actinides) [41]. Pyroprocessing is preferred over aqueous processing because it allows the salt to remain in a molten state throughout the process [65] and does not require dissolving the spent fuel in an acid solution. The primary components of the HLW stream are noble gases (which may be recycled after a few hours of decay or stored for ~100 years), HTO, iodine, C-14 (only for thermal spectrum MSRs), noble metals, technicium-99 (Tc-99), cesium isotopes, strontium isotopes, lanthanides, and zirconium. HLW may also be formed from decontamination of equipment, vessels, or piping [41]. The fast MSR reprocessing process is under development but is illustrated in Figure 10 [41]. The MSR process for fuel reprocessing uses an online, continuous gas extraction system and an online/offline lanthanide extraction process [41]. Approximately 10 to 40 L/day of fuel are expected to be processed [58]. Noble gases are "bubbled" from the mixture. Noble metals are also removed so they do not plate-out on essential surfaces, such as heat exchangers, where their decay heat could cause damage. Uranium, neptunium, and plutonium are recycled [41]. The main difficulties of waste reprocessing are anticipated to be the confinement of tritium (produced especially from lithium, if part of the carrier salt mixture) and complications due to the presence of zirconium. While protactinium was identified as problematic for the thermal MSR reprocessing, it is not a concern with the fast MSR reprocessing [41].

3.5.4 Lead-Cooled Fast Reactor Waste Streams

LFRs are another type of fast breeder reactor that uses a lead or lead-bismuth eutectic liquid metal to cool the reactor [41]. Waste generated during operation includes lead-contaminated solid media and filter waste, lead-contaminated liquid and gaseous wastes, and bulk lead from the cooling system components. Waste generated during decommissioning includes additional reactor component wastes. A unique consideration of LFRs is the environmental risk associated with lead toxicity [41]. Any processing of lead contamination must be controlled to avoid exposure to workers and

Figure 10: Fast MSR process flow and primary waste streams. Black arrows indicate a gas stream, orange arrows indicate a fluoride stream, blue dotted arrows indicate metallic bath handling, and the purple arrow shows molten chloride use. Reproduced from [41] with permission, © IAEA.



the environment. Fuel is typically UO2 or uranium nitride ceramic fuel with less than 20% enrichment that is inserted into stainless steel cladding and then backfilled with helium [42]. Fuel management would likely require on-site storage, off-site storage, disposal, and transport capabilities. If lead is used as the fuel-clad bond, it may need to be removed before disposal due to toxicity concerns [42]. Treatment processes must consider C-14 and I-129 releases, as well as the potential for hydrogen gas generation [42]. Additional concerns relate to the corrosive nature of molten lead, which will require extensive cleanup systems for impurities from the lead coolant [45] and could result in large amounts of filters and solid media wastes. Activation products from lead-based materials include bismuth-210 (Bi-210), Po-210, lead-210 (Pb-210), and lead-205 (Pb-205) [44], most of which are not encountered in CANDU waste. Po-210 is expected at high activities and is also very toxic. However, with a half-life of only 138 days, the Po-210 in the lead mixture would decay sufficiently in about three years to designate the lead waste as ILW [46]. Nonetheless, the interim storage for the first three years would need to consider elevated decay heat and dose rates [42]. The specific features of the SEALER reactor, the only LFR that has been considered for implementation in Canada, are provided in Section 3.5.4.1.

3.5.4.1 SEALER

The SEALER LFR uses a melted natural lead coolant and can use either UO_2 or enriched uranium nitride fuel [46]. Novel aluminum-steel alloys are proposed for corrosion resistance of components in contact with liquid lead [43]. This would generate radioactive corrosion products, such as Co-60, Fe-55, Fe-59, Ni-59, Ni-63, Cr-51, and manganese-54 (Mn-54). The reactor is designed for a 30-year equipment operating life with no refuelling [46]. The primary vessel and steam generator dimensions are compact to allow for their transport to the site by air [38]. The lead coolant is expected to form stable compounds with iodine and cesium, diminishing their release [38]. Noble gases produced by fission would need to be captured by an off-gas system. In the case of uranium nitride fuel, generation of C-14 must be considered in the off-gas management [42].

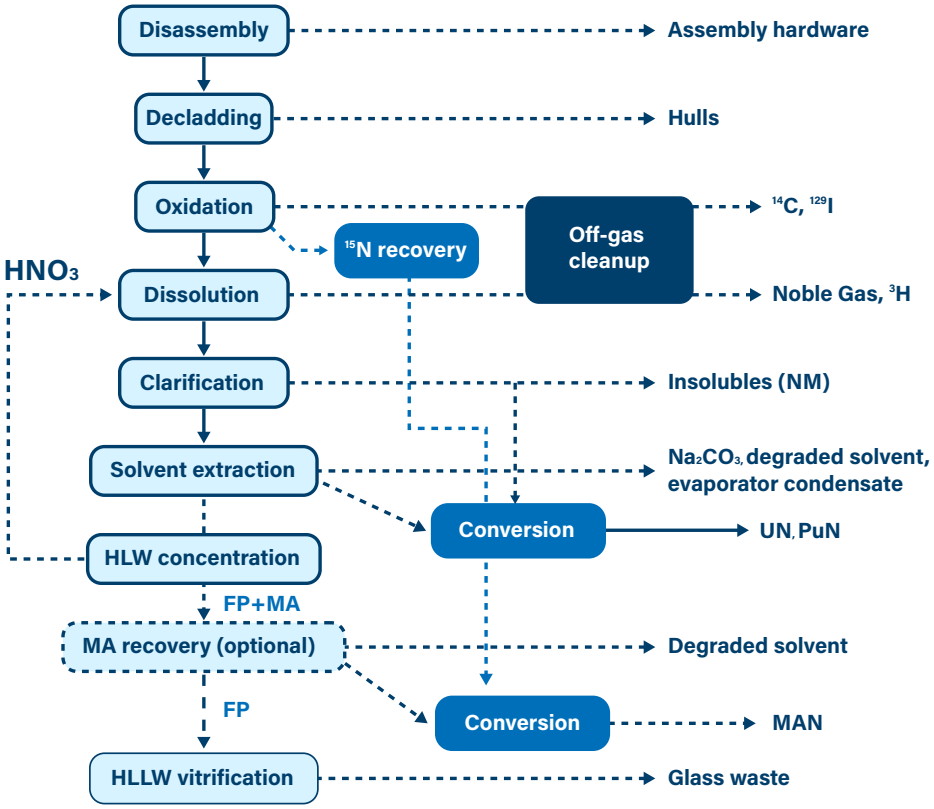
3.5.4.2 Lead-Cooled Fast Reactor Fuel and other High-Level Waste

The LFR uses uranium nitride fuel [66], and the fuel cycle is illustrated in Figure 11. In nitride-based LFRs, like their oxide fuel counterparts, the waste stream includes a large amount of radioactivity and primary side waste, the possibility of hydrogen gas generation, and impacts on vitrification. If the cement process is employed to condition assembly hardware and hulls, then hydrogen gas generation must be considered as a waste stream. As higher concentrations of noble metal are encapsulated in glass, as well as higher alpha activity in the glass, then difficulties with both the vitrification and integrity of the final waste form must be carefully considered. Alternative approaches to vitrification may be considered, such as cold crucible technology (i.e., metal melting) [41]. Specific to nitride-based LFRs, high concentrations of nitrogen-15 (N-15) in the fuel would be required to prevent the generation of C-14 from N-14. This would necessitate recycling the N-15 due to its cost. However, wastes from the oxidation step (prior to dissolution) to recover N-15 are not well characterized [41]. It should be noted that nitride compounds are very susceptible to oxidation and hydrolysis, and are pyrophoric when in powder form. The primary LFR HLW streams are the assembly hardware (consisting of the wrapper tube, nozzle, neutron shield, and handling head), fuel hulls (cladding and spacers), C-14, I-129, noble gases (krypton, xenon), H-3, insolubles (noble metals), sodium carbonate, degraded solvent (for major and minor actinides), evaporator condensate, Tc-99 [58], cesium/strontium, glass waste, and any high-level liquid wastes produced from decontamination of equipment, piping, or vessels [41].

3.5.5 Sodium-Cooled Fast Reactor Waste Streams

Similar to LFRs, SFRs are fast reactors that typically operate without a moderator. Sodium is the coolant, which creates additional implications with its highly reactive nature. Hot sodium reacts violently with water, and it catches fire when it comes in contact with air, emitting dense clouds of white sodium peroxide smoke and hydrogen gas, with the potential for a hydrogen explosion [62]. Sodium also reacts with zirconium,

Figure 11: Primary waste streams and process flow for nitride-based LFRs. NM: noble metals, MA: minor actinides. Reproduced from [41] with permission, © IAEA.



thereby precluding the use of zirconium alloys in reactor components. Generally, both the primary and secondary loop coolants are liquid sodium. A cover gas is required to prevent contact of the sodium with atmospheric air. Handling and any processing of the waste must be carefully planned and executed to prevent contact of the sodium with air or water. The treated sodium would likely need to be solidified into a stable matrix waste form as LLW. Because a primary coolant purification system is used to capture corrosion and fission products, their presence in the sodium waste would be minimized, but sodium-contaminated purification waste streams, including filters and purification media, would be generated instead. Other possible processes include the controlled reaction with sodium hydroxide and the transformation to sodium carbonate [41].

Sodium reactivity concerns were documented from the experimental sodium-cooled fast reactors that operated during the late twentieth century [44]. Decommissioning was complicated by the large volumes of metallic sodium coolant containing Na-22 and cesium isotopes leached from ruptured fuel elements. The sodium was processed through a pilot scale water-based process under an inert atmosphere, but it resulted in several hundred cubic metres of LLW [44].

Sodium-cooled reactors use a steel-clad uranium metal and elemental sodium solid fuel [44]. The fuel would need processing to convert it into a more stable chemical form for storage and eventual disposal, which is discussed in Section 3.5.5.2. Similar to the molten salt and lead-cooled reactors,

the processing would generate additional ILW waste streams in the form of filters, media wastes, and contaminated process equipment. The resulting processed fuel waste would need to be immobilized, for example, by vitrification. The off-gas contaminated with C-14, I-129, and semi-volatile radionuclides and noble gases would also need to be treated, generating additional LLW or ILW filter and media wastes.

In addition to the back-end steps of the nuclear fuel cycle, the fuel fabrication would need to be carried out inside hot cells and under an inert gas cover [41]. This would create hot cell equipment wastes and a need to process the inert gas.

Handling and transporting the fuel, both before and after operation in the reactor, must be done carefully and in a controlled manner to avoid issues with pyrophoricity of the metallic sodium and uranium hydrides. In addition to pyrophoricity concerns, used fuel storage and treatment facilities would need to accommodate elevated decay heat and dose rates as well as potentially lower structural integrity due to radiation damage [42].

Common reflector and shielding materials include steel and boron carbide, but beryllium, magnesium, lead, and other materials can also be used [44]. Common activation product radionuclides are Na-22 and Na-24 from the sodium coolant, H-3, C-14, and Be-10 from the boron carbide, Be-10 from the beryllium, and iron, manganese, cobalt, and nickel isotopes from the steel.

The features specific to the ARC-100 reactor, the only SFR that has been considered for implementation in Canada, are provided in Section 3.5.5.1.

3.5.5.1 ARC-100

The ARC-100 is an SFR with a pool type configuration [67]. The fuel is a uranium-zirconium mixture, sodium-bonded binary metallic fuel with an average enrichment of 13.1%.

Liquid radioactive waste is generated by washing the primary sodium coolant from components. The reactor design is set up with an evaporation-based system to treat the active liquid wastes. The resulting solid concentrate is packaged and the vapours from the evaporation process are filtered prior to discharge. Argon cover gas is also processed by filtration to treat the cover gas and remove the airborne particulates released from the sodium pool [67].

The sodium in the secondary loop is not expected to become activated due to the shielding design [62]. Therefore, the secondary loop sodium processing is expected to be more straightforward and focused on removing the impurities, resulting in filter and media wastes. However, the sodium itself would be considered a conventional hazardous waste stream upon decommissioning.

The reactor structure would be installed below grade [43], introducing the potential for contamination of the surrounding soil and corrosion of the reactor building structure from exposure to groundwater.

The ARC-100 uses steel as the reflector and shield assemblies [38], so high activities of C-14 and Be-10 are not expected, although some C-14 would be generated from the activation of steel.

The ability to store the used fuel within the reactor vessel eliminates the need for a separate used fuel storage pool [67].

3.5.5.2 Sodium-Cooled Fast Reactor Fuel and other High-Level Waste

The ARC-100 is a uranium-zirconium alloy fuelled SFR [67]. SFRs can operate with a closed fuel cycle and have unique prospects for actinide management [41]. The ARC-100 manufacturer has stated that it will operate with an open fuel cycle until the closed fuel cycle has regulatory approval, public acceptance, and is economically viable [68].

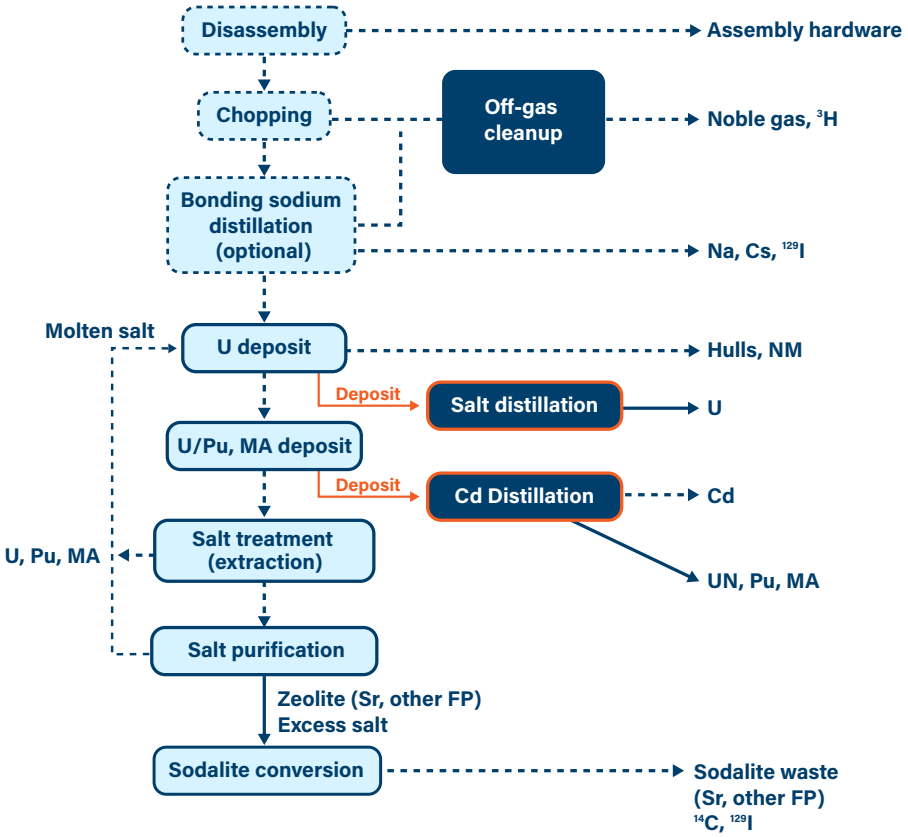
Reprocessing of fuels, which is the chemical separation of uranium and plutonium from these spent fuels, can be accomplished via one of two routes for metallic-fuelled SFRs: a) aqueous, or b) pyroprocessing/injection casting [58]. Figure 12 shows a typical reprocessing flow, with the waste streams identified at appropriate steps.

The four essential SFR challenges related to waste are: a) uncertainties in the behaviour of H-3, which must be removed from the off-gas system, and I-129 and C-14, which would be present in sodalite waste; b) high-temperature alloying of noble metals; c) management of cadmium waste (from the cadmium electrode used in pyroprocessing); and d) the high volume of salt waste. The primary waste streams from the reprocessing of fuel for the ARC-100 would be the assembly hardware (wrapper tube, nozzle, neutron shield, and handling head), noble gases (krypton, xenon), H-3, Na, Cs, I-129, hulls, noble metals, cadmium, and sodalite waste (containing strontium, other fission products, C-14, and I-129) [41].

3.5.6 Heat Pipes/Solid Core Reactor Waste Streams

The heat pipes/solid core reactor is a subset of the HTGR design, and at the time of writing, the eVinci is the only known reactor of this design. The eVinci is a microreactor that uses TRISO fuel and a graphite core block that acts as the moderator and the core structure. The graphite block contains hexagonal unit cells oriented horizontally along the length of the core, and contains channels for fuel, burnable absorbers, alkali metal heat pipes, and shutdown rods [69]. This would produce decommissioning waste in the forms of activated graphite and steel piping material. The cooling is a fully passive heat pipe-cooled design

Figure 12: Primary waste streams and process flow for metallic-fueled liquid metal reactor. NM: noble metal, MA: minor actinide, FP: fission product. Reproduced from [41] with permission, © IAEA.



[47]. Heat pipes use liquid sodium, so very little waste is expected to be generated during operation due to the fully passive design and minimal operational staff presence. However, waste from the liquid sodium coolant is expected to produce Na-22 and Na-24 radionuclides, and will likely require processing upon decommissioning. The reactor is designed to be manufactured and fuelled in a factory, then packaged and transported to site via air, water, or land [38]. It is designed to operate for eight years [47], after which the unit can be transported to the factory for refuelling or long-term storage [38]. Helium is used as cover gas, which minimizes the oxidation of core components [69]. The helium would carry the fission products and noble gases through an off-gas system, producing filter and media waste. The helium itself would be cleaned up and released upon decommissioning.

3.5.6.1 Heat Pipes/Solid Core Reactor Fuel and other High-Level Waste

The eVinci is a 15 MW_{th} /5 MW_e reactor and is designed to run for eight or more full-power years before refuelling. The eVinci uses TRISO fuel with HALEU (19.75% U-235) in its kernels. According to documentation filed with the United States Nuclear Regulatory Commission, the eVinci's TRISO fuel kernels will be UCO-based [70]. The compacts (fuel elements) are graphitic cylinders and will bear the TRISO particles [70]. UCO-based TRISO fuel was discussed both as a fuel and in waste form in Section 3.5.2.5, which describes recycling and waste streams for HTGRs.

3.5.7 Waste Streams from Front-End Steps of the Nuclear Fuel Cycle

The fuel used in existing CANDU reactors does not require enrichment. In contrast, most SMR designs require the fuel to be enriched. Uranium enrichment is the process of increasing the concentration of isotope U-235 in natural uranium, which consists of 99.3% U-238 and 0.7% U-235 [58], [71]. To meet the enriched fuel requirements, a uranium enrichment process may need to be implemented in Canada. Because enrichment uses the gas state, the mined and purified yellowcake (U_3O_8) needs to be converted to uranium hexafluoride (UF_6) gas [71]. Other than a volume increase due to more fuel usage, there are no changes to the waste that results from the enrichment process. The Port Hope Conversion Facility in Ontario already converts UO_3, produced from U_3O_8 in the Blind River Refinery, into UF_6 for export to other countries. Therefore, only the enrichment process would need to be implemented in Canada or subcontracted to another country. After enrichment, the UF_6 is converted back to UO_2. If the enrichment process is implemented in Canada, new waste streams would be generated for the nuclear fuel cycle front-end, which would include depleted uranium, uranium contaminated process equipment, uranium contaminated media used to treat gaseous and liquid effluents, and fluorine wastes.

For SFRs in particular, the fuel fabrication would need to take place inside hot cells and under an inert cover gas [32] to avoid sodium contact with atmospheric air. This would generate hot cell equipment wastes and inert gas process wastes. Some forms of fuel, such as the silicon carbide-coated TRISO fuel are considered a HALEU fuel, which is enriched to between 5% and 20%. The fuel kernel itself can be either UO_2 or UCO. Because enrichment above 10% requires additional licensing and security measures [71], typical HALEU fuel greater than 10% enrichment is produced on the fuel manufacturing site, where it can be converted from UF_6 and fabricated into fuel, thereby avoiding the need for special transport packages [71].

3.5.8 Summary of High-Level Waste Features

Table 2 presents an analysis and summary of the HLW streams discussed for each SMR design type in the context of differences from CANDU reactors. These differences were considered for the assessment of current standards and regulations.

3.6 Regulations and Governance Applicable to Radioactive Waste

The management of radioactive waste in Canada is guided by the Government of Canada's Policy for Radioactive Waste Management and Decommissioning [72] and is regulated by the CNSC.

Table 2: Summary of HLW features of SMR designs. **HLW** = high-level waste, **CANDU** = Canada deuterium uranium, **PHWR** = Pressurized heavy water reactor, **LWR** = Light water reactor, **PWR** = Pressurized water reactor, **BWR** = Boiling water reactor, **HTGR** = High-temperature gas-cooled reactor, **MSR** = Molten salt reactor, **LFR** = Lead-cooled fast reactor, **SFR** = Sodium-cooled fast reactor.

Reactor Design	Key HLW Differences from CANDU Reactors
LWR (includes PWRs and BWRs) ▪ BWRX-300 ▪ SMR-160 ▪ NPM	▪ Increased decay heat of the fuel, owing to higher burnup [58]. ▪ Fuel assemblies are physically larger than CANDU fuel bundles and may require downsizing or larger waste containers.
HTGR ▪ MMR ▪ Xe-100 ▪ U-Battery ▪ StarCore	▪ Larger volume of HLW as fuel particles are embedded in graphite pebbles or blocks [58]. ▪ While pebble bed reactors have smaller fuel elements, prismatic block reactors may have larger elements of different geometry, requiring a different-sized container. ▪ Increased decay heat of the fuel, owing to higher burnups. ▪ Graphite is combustible in air; graphite dust mixtures with air may ignite. ▪ If reprocessed, primary HLW streams would include coatings (contaminated by H-3, C-14, and I-129), noble gases and gaseous H-3 and HTO, noble metals, sodium carbonate, UO_2, and plutonium oxide (to be reused as reactor fuel), degraded solvents, minor actinides in the form of MA_2O_3, and glass waste.
MSR ▪ SSR-W300 ▪ IMSR400	▪ Graphite is combustible in air; graphite dust mixtures with air may ignite. ▪ HLW may result from decontamination of equipment, vessels, or piping [41]. ▪ Fuel in salt state will need vitrification prior to disposal, or will need to be reprocessed [58]. ▪ Geometry of vitrified used fuel is flexible. ▪ Primary HLW streams are noble gases (which may be recycled after a few hours of decay or stored for ~100 years), HTO, iodine, C-14 (only an issue for the thermal spectrum MSRs), noble metals, Tc-99, cesium isotopes, strontium isotopes, lanthanides, and zirconium.
LFR ▪ SEALER	▪ Increased decay heat of the fuel [41] compared to CANDU bundles. ▪ Addition of N-15 to the fuel may make fuel manufacturing costly; wastes from the oxidation step to recover N-15 are not well known [41]. ▪ Primary HLW streams are the assembly hardware (consisting of the wrapper tube, nozzle, neutron shield, and handling head), fuel hulls (cladding and spacers), C-14, I-129, noble gases (krypton, xenon), H-3, insolubles (noble metals), sodium carbonate, degraded solvent (for major and minor actinides), evaporator condensate, Tc-99 [58], cesium/strontium, glass waste, and any high-level liquid wastes produced from decontamination of equipment, piping, or vessels [41]. ▪ Fuel elements/assemblies may be different than CANDU reactors and current used fuel containers may need to be modified.

Reactor Design	Key HLW Differences from CANDU Reactors
SFR ▪ ARC-100	▪ Increased decay heat of the fuel compared to CANDU bundles. ▪ Main uncertainties are a) the behaviour of H-3, which must be removed from the off-gas system, and I-129 and C-14, which would be present in sodalite waste; b) high-temperature alloying of noble metals; c) management of cadmium waste (from the cadmium electrode used in some pyroprocessing); and d) the high volume of salt waste. ▪ Fuel elements/assemblies may be different than CANDU reactors and current used fuel containers may need to be modified. ▪ Sodium in the fuel assembly, unless conditioned, would become part of HLW.
Heat Pipes/Solid Core Reactor ▪ eVinci	▪ Graphite is combustible in air; graphite dust mixtures with air may ignite. ▪ Fuel has increased decay heat and is larger in volume compared to CANDU fuel [70].

The following regulatory documents are a key part of the CNSC's regulatory framework for waste management and decommissioning [10]:

- REGDOC-1.2.1, *Guidance on Deep Geological Repository Site Characterization*
- REGDOC-2.11, *Framework for Radioactive Waste Management and Decommissioning in Canada*
- REGDOC-2.11.1, *Waste Management, Volume I: Management of Radioactive Waste*
- REGDOC-2.11.1, *Waste Management, Volume II: Management of Uranium Mine Waste Rock and Mill Tailings* (not applicable to this project)
- REGDOC-2.11.1, *Waste Management, Volume III: Safety Case for Disposal of Radioactive Waste*
- REGDOC-2.11.2, *Decommissioning*
- REGDOC-3.3.1, *Financial Guarantees for Decommissioning of Nuclear Facilities and Termination of Licensed Activities* (not applicable to this project)

The following roles and responsibilities outlined in REGDOC-2.11 help ensure the safe and secure management of radioactive waste, including that from any potential SMRs to be implemented in Canada [10]:

- Natural Resources Canada is responsible for developing a federal nuclear energy policy across the nuclear supply chain, including establishing the framework for radioactive waste management and decommissioning.
- The CNSC regulates and enables the management of radioactive waste, through licensing the management of radioactive waste, including its generation, handling, processing, transport, storage, and disposal.
- Waste owners, also referred to as licensees, are required to manage the radioactive waste in a safe and secure manner, and to arrange for its long-term management.
- The NWMO is responsible for designing and implementing Canada's plan for safe, long-term management of used nuclear fuel, as well as for developing and implementing a plan for the effective safe management of ILW and non-fuel HLW.

The CNSC regulations listed above and additional supplementary information, including the following, were reviewed for applicability to waste from SMRs:

- CNSC's Radionuclide Information Booklet [73]
- CNSC's Nuclear Substances and Radiation Devices Regulation [74]
- CNSC's Radioactive Waste website [13]

The review focused on any potential items missing from the regulations that could prohibit or limit the deployment of any SMR technology, and identified the following three issues:

1. Section 8.2 in REGDOC-2.11.1, Volume I [10], includes the features to be taken into consideration when handling radioactive waste, and it does not explicitly mention the conventional safety for people and environment (as low as reasonably achievable, ALARA).

   "For the selected waste handling methods, the licensee shall take into account:

   - the characteristics of the waste
   - the types of containment systems and packages
   - the minimization of radiological risks in accordance with the ALARA principle"

   Based on the SMR waste features presented in Table 2, chemically reactive coolants (such as that of MSRs and SFRs) and combustible fuel elements or moderators (such as that of SFRs) require special handling procedures to ensure the safety of both workers and the environment.

2. CNSC's Radionuclide Information Booklet [73] does not include the radionuclide Po-210, which is a key radionuclide to consider as a result of lead activation in LFRs.

3. At the time of writing, NWMO's Adaptive Phased Management strategy considers Canada's existing used nuclear fuel, although it is adapting to emerging technologies, including emerging sources of used nuclear fuel from SMRs, fuel reprocessing possibilities, and other advances [75], [76]. One key requirement in managing Canada's used fuel is for the fuel to "be transportable to the repository site, and be able to fit into a licensed transportation package" [76]. This introduces the requirement for the development and/or licensing of new transport packaging for the different sizes and shapes of fuel assemblies that would be produced by the different SMR designs.

3.7 Standards Applicable to Radioactive Waste

The CSA N292 Series of standards complement the CNSC's regulatory framework on radioactive waste management, and include the following:

- CSA N292.0, *Radioactive Waste Management: Common Requirements of the CSA N292 Series of Standards* [12]
- CSA N292.4, *Storage of Radioactive Waste and Irradiated Fuel* [77]
- CSA N292.5, *Guideline for Exemption or Clearance of Materials that Contain, or Potentially Contain, Nuclear Substances* [78]
- CSA N292.7, *Deep Geological Disposal of Radioactive Waste and Irradiated Fuel* [79]
- CSA N292.8, *Characterization of Radioactive Waste and Irradiated Fuel* [80]

These standards were reviewed to assess their applicability to new waste streams, waste forms, and radionuclides that could be generated by the SMR designs reviewed in Section 3.3. Specifically, the standards were reviewed to determine their inclusion of the three categories of waste features presented in Section 3.4:

1. CANDU waste features;
2. Slight difference from CANDU waste features (qualitative or quantitative); and
3. Completely new SMR waste features.

The critical review assessed whether various clauses included in the CSA N292 Series also apply to SMR waste features. Gaps were identified only for some of the completely new SMR waste features, as shown in Table 3.

Table 3: Gaps in CSA N292 Series for completely new SMR waste features. **SMR** = small modular reactor, **HALEU** = High-assay low-enriched uranium, **TRISO** = tristructural isotropic, **HLW** = high-level waste.

New Waste Feature	Gaps Identified?	Details of Gaps
Higher burnup and hotter fuel	No	
Enriched fuel	Yes	▪ Non-proliferation ▪ Criticality ▪ Additional security and safeguards for HALEU fuel
Light water coolant/ moderator	No	
Boron waste	No	
Graphite waste	Yes	▪ Large blocks of graphite/large number of graphite fuel pebbles may need new/different packaging ▪ Graphite dust may be pyrophoric
Gas coolant	No	
Uranium nitride fuel	Yes	▪ Different physical size/shape of fuel assemblies and the impact on packaging for transport
Metallic fuel	Yes	▪ Different physical size/shape of fuel assemblies and the impact on packaging for transport
TRISO fuel	Yes	▪ Different size/shape of fuel pebbles/compacts and the impact on packaging and transport
Molten uranium chloride/fluoride salt coolant and fuel	Yes	▪ Chemically reactive molten salt mixture with a tendency to generate explosive or corrosive by-products upon contact with air or moisture ▪ Production of HLW from decontamination of equipment, vessels and piping, and possibly reprocessing
Liquid lead or liquid sodium coolant	Yes	▪ Toxicity and environmental risk ▪ Chemically reactive in contact with air or moisture
Cover gas waste	Yes	▪ Processing may generate HLW
New radionuclides (Na-22, Na-24, F-18, Po-210)	No	
Combustible fuel elements or moderator	Yes	▪ Containment system design ▪ Scope of hazards assessment ▪ Needs considerations for prevention of fire/explosion

Additional SMR waste features that have slight differences from CANDU features (qualitative or quantitative), but do not necessitate updates to the CSA N292 Series are presented in Table 4.

Table 4: SMR waste features with slight differences from CANDU waste features. **SMR** = small modular reactor, **CANDU** = Canada deuterium uranium.

Waste Feature	Description of Differences
Greater Ni-59 and Ni-63 activities	Ni-59 and Ni-63 are already present in waste from CANDU reactors, and do not impose any special considerations
Lower H-3 and C-14 activities	H-3 and C-14 are already present in waste from CANDU reactors, and do not impose any special considerations
Different size/shape fuel assemblies	This has no impact on the CSA N292 Series
More gaseous waste	Gaseous waste is already present from CANDU reactors; larger amounts have no impact on the CSA N292 Series
More decommissioning waste per MWe	Waste generated from decommissioning is already covered in the CSA N292 Series; greater amounts have no impact
Contaminated soil and groundwater from underground embedded core	Soil and groundwater waste is already present in waste from CANDU reactors, and does not impose any special considerations
May not require wet storage of fuel	Wet storage of fuel is included in the CSA N292 Series; its absence has no impact

Recommendations for updates to the CSA N292 Series are provided in Table 5, and are specific to CSA N292.0 and CSA N292.4. No gaps were found in CSA N292.5, CSA N292.7, and CSA N292.8. Findings on general improvements, unrelated to waste from SMRs, were also identified and are included in Table 5.

Table 5: Recommended updates to CSA N292 Series based on waste features from SMRs. SMR = small modular reactor, HALEU = High-assay low-enriched uranium, TRISO = tristructural isotropic.

Feature of Waste from SMRs	Consideration of Waste Feature	Covered (Standard and Sections)	Recommendations
Hotter used fuel	Heat removal	CSA N292.0: ▪ 4.4.2 (General considerations for design and operations of systems) ▪ 4.5.2 (Considerations for the radioactive waste classification system) ▪ 11.3.2.1 (Design heat load capacity) ▪ 12.5.5 (Design considerations)	None
Enriched fuel	Non-proliferation	CSA N292.0: ▪ 4.12 (Safeguards)	None

Feature of Waste from SMRs	Consideration of Waste Feature	Covered (Standard and Sections)	Recommendations
Enriched fuel	Criticality concerns with wet storage of enriched used fuel	CSA N292.0: ▪ 12.7.1.1 (Performance and safety assessment approaches) ▪ 12.5.5 (Design considerations) ▪ N292.4: ▪ 7.1.2 (Main Safety functions) ▪ 7.3.2.1 (Configuration and materials for criticality safety) ▪ 7.2.1 (Application of requirements for existing wet storage systems)	None
Enriched fuel	HALEU requires enrichment between 5% and 20%. Enrichment > 10% requires additional licensing and security measures	Not currently covered	CSA N292.0, Section 8 (Transport): ▪ Add "considerations for safeguarding and transport of enriched fuel".
Graphite waste	Large graphite block components	Not currently covered	CSA N292.0: ▪ Add ability to store as waste "large graphite blocks".
Graphite waste	Pyrophoric graphite dust handling	Not currently covered	CSA N292.0: ▪ Add "pyrophoricity" as a hazard associated with handling waste. ▪ Add process/procedure for handling pyrophoric waste.
Graphite waste	Wigner energy release potential in packaged graphite waste	Not currently covered	CSA N292.0: ▪ Add a consideration for waste containment to consider stored energy, such as Wigner energy in graphite.
TRISO fuel	Different size/shape of fuel pebbles/compacts	Not currently covered	CSA N292.4: ▪ Add ability of primary storage for fuel to consist of fuel baskets, modules, or other similar components.
Chloride/ fluoride salt coolant/fuel	Chemically reactive molten salt mixture with a tendency to generate explosive or corrosive by-products upon contact with air or moisture	CSA N292.0: ▪ 6.5.7 (Explosivity, pyrophoricity, and chemical reactivity) ▪ 6.5.2.2 (Corrosive materials	CSA N292.0: ▪ Add requirement to keep salt wastes isolated from water to prevent an increase in corrosivity. ▪ Add controls for chemical reactivity. For example, the following note could be added in Section 4.4.2 (General considerations for design and operations of systems): *"Note: Waste with the potential for pyrophoricity, explosivity, or other significant reactivity shall be handled in an inert atmosphere. This could be using a glove box, hot cell, flowing inert cover gas, or other physical method to prevent contact between the waste and the reactant."* ▪ Add considerations for safe management of waste to include hazardous emissions and potential for chemical explosion. CSA N292.4, Section 3.1 (Definitions): ▪ Add "Containment… or ingress of oxidizing species, such as water or oxygen, into the storage system."

Feature of Waste from SMRs	Consideration of Waste Feature	Covered (Standard and Sections)	Recommendations
Chloride/ fluoride salt coolant/fuel	Reprocessing used fuel from reactor to generate fresh fuel (e.g., Moltex Energy's Waste to Stable Salt technology)	Not currently covered	CSA N292.0, Figure 2, General flow diagram, Radioactive waste management: ▪ Figure includes steps for processing but does not include steps to reuse as fuel; add option for fuel reuse beside "On-site storage: Maintenance and monitoring" box.
Liquid lead coolant	Processing of lead must avoid exposure to workers and the environment (environmental risk due to lead toxicity)	CSA N292.0: ▪ 4.4.1. (Safety requirements for radioactive waste management) ▪ 4.9.4 (Non-radiological hazards)	None
Liquid sodium coolant	Handling and processing of waste from sodium-cooled reactors must prevent contact of sodium with air or water	CSA N292.0: ▪ 6.5.10 (Environmental conditions)	None
Combustible fuel elements or moderator	Containment system design	Not currently covered	CSA N292.0: ▪ Add a design consideration for thermal, fissionability/criticality, and combustibility. CSA N292.4: ▪ Add a safety function consideration for protection against fire or explosion.
Combustible fuel elements or moderator	Scope of hazards assessment	Not currently covered	CSA N292.0 ▪ Add a hazard control for release of excessive heat and fire.
Combustible fuel elements or moderator	Safety objective	Not currently covered	CSA N292.4: ▪ Add a safety consideration for fire/explosion.
Chloride/ fluoride salt coolant/fuel	Chemically reactive molten salt mixture with a tendency to generate explosive or corrosive by-products upon contact with air or moisture	CSA N292.0: ▪ 6.5.7 (Explosivity, pyrophoricity, and chemical reactivity) ▪ 6.5.2.2 (Corrosive materials	CSA N292.0: ▪ Add ability to store as waste "large steel blocks".
General	Large SMR core components (e.g., StarCore's graphite fuel prismatic block)	Not currently covered	CSA N292.0: ▪ Add ability to store as waste "entire SMR core".
General	Entire unit transported to factory for refuelling or storage (e.g., eVinci)	Not currently covered	CSA N292.0: ▪ Add ability to store as waste "entire SMR core".

Feature of Waste from SMRs	Consideration of Waste Feature	Covered (Standard and Sections)	Recommendations
General	Radioactive waste characterization	Not currently covered	CSA N292.0: ▪ Add to list of characterization categories, "thermal, fissionability/criticality, combustibility"
General	Definitions	Not currently covered	CSA N292.0 ▪ Consider adding definitions for "defective fuel" and "damaged fuel".
General	Waste characterization requirements	Not currently covered	CSA N292.0: ▪ Add "a unique identifier" to the list of waste characterization documentation.
General	Clarification regarding conditioning/processing	Not currently covered	CSA N292.0: ▪ Add procedure for waste management to include processing/immobilization. ▪ Add considerations for waste generators to ensure that the waste is conditioned, if required. ▪ Add requirement for records to describe any conditioning or processing performed on the waste or waste package.
General	Effect of small core size	Not currently covered	None

3.7.1 Other Standards

In addition to the CSA N292 Series, the following Canadian standards further complement radioactive waste management-related activities:

- CSA N288 Series on environmental management of nuclear facilities
- CSA N294, *Decommissioning of Facilities Containing Nuclear Substances* [81]
- CSA N393, *Fire Protection for Facilities that Process, Handle, or Store Nuclear Substances* [82]

These standards were reviewed at a much higher level than the detailed critical review conducted on the CSA N292 Series, and the findings indicated that:

- The CSA N288 Series could consider including provisions for potential soil and groundwater contamination from some SMR designs, where the reactor core is buried underground.
- CSA N294 has no gaps, and appears to be applicable to waste from SMRs as written.
- CSA N393 has no gaps, and appears to be applicable to waste from SMRs as written.

4 Conclusion and Recommendations

This report summarized the radioactive waste forms expected to be generated throughout the lifecycles of SMR designs that are planned or being evaluated for deployment in Canada. It also reviewed the CSA N292 Series and other applicable standards and regulations, identified gaps, and provided recommendations to ensure that waste from SMRs is reflected in the relevant Canadian standards and regulations.

In comparison to CANDU reactors, which served as the baseline for existing regulations and standards in Canada, the findings indicate that LWRs produce



radioactive waste forms largely consistent with CANDU reactors. The findings also indicate that HTGRs and heat pipes/solid core reactors produce a few novel waste forms and materials, but MSRs, LFRs, and SFRs produce multiple novel waste forms, unique radionuclides, requirements for processing, and other radioactive waste considerations.

Waste streams from each of the 12 designs considered in this report were described across six distinct categories. In all cases, the fuel is physically different in size, shape, and mass than current CANDU fuel bundles. The radioisotopes produced by LWRs would be the same as CANDU reactors, but in different quantities due to different discharge burnups. MSRs, LFRs, and SFRs produce high-level liquid waste that needs to be processed and immobilized.

In general, the current regulatory framework in Canada covers the management of waste from SMRs. However, one recommendation for this regulatory framework is to explicitly address the conventional safety of workers and the protection of the environment.

Finally, the current CSA N292 Series adequately reflects the waste streams that are anticipated from SMRs. Minor changes are recommended for two standards, namely CSA N292.0 and CSA N292.4, pertaining to:

- Enriched fuel, which requires non-proliferation measures, criticality measures, and additional security for HALEU fuel;
- Graphitic waste, which may require additional consideration due to flammability and explosion hazards, and may require new packaging; and
- Additional toxicity and environmental concerns with liquid lead or liquid sodium-cooled reactors.

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