EPRI

Impact of Advanced Reactors on the Back-End Management of the Nuclear Fuel Cycle

Advanced Reactor Fuel Cycle Landscape Study

3002023916

Impact of Advanced Reactors on the Back-End Management of the Nuclear Fuel Cycle

Advanced Reactor Fuel Cycle Landscape Study

3002023916

Technical Update, December 2022

EPRI Project Manager

D. Moneghan

DISCLAIMER OF WARRANTIES AND LIMITATION OF LIABILITIES

THIS DOCUMENT WAS PREPARED BY THE ORGANIZATION(S) NAMED BELOW AS AN ACCOUNT OF WORK SPONSORED OR COSPONSORED BY THE ELECTRIC POWER RESEARCH INSTITUTE, INC. (EPRI). NEITHER EPRI, ANY MEMBER OF EPRI, ANY COSPONSOR, THE ORGANIZATION(S) BELOW, NOR ANY PERSON ACTING ON BEHALF OF ANY OF THEM:

(A) MAKES ANY WARRANTY OR REPRESENTATION WHATSOEVER, EXPRESS OR IMPLIED, (I) WITH RESPECT TO THE USE OF ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT, INCLUDING MERCHANTABILITY AND FITNESS FOR A PARTICULAR PURPOSE, OR (II) THAT SUCH USE DOES NOT INFRINGE ON OR INTERFERE WITH PRIVATELY OWNED RIGHTS, INCLUDING ANY PARTY'S INTELLECTUAL PROPERTY, OR (III) THAT THIS DOCUMENT IS SUITABLE TO ANY PARTICULAR USER'S CIRCUMSTANCE; OR

(B) ASSUMES RESPONSIBILITY FOR ANY DAMAGES OR OTHER LIABILITY WHATSOEVER (INCLUDING ANY CONSEQUENTIAL DAMAGES, EVEN IF EPRI OR ANY EPRI REPRESENTATIVE HAS BEEN ADVISED OF THE POSSIBILITY OF SUCH DAMAGES) RESULTING FROM YOUR SELECTION OR USE OF THIS DOCUMENT OR ANY INFORMATION, APPARATUS, METHOD, PROCESS, OR SIMILAR ITEM DISCLOSED IN THIS DOCUMENT.

REFERENCE HEREIN TO ANY SPECIFIC COMMERCIAL PRODUCT, PROCESS, OR SERVICE BY ITS TRADE NAME, TRADEMARK, MANUFACTURER, OR OTHERWISE, DOES NOT NECESSARILY CONSTITUTE OR IMPLY ITS ENDORSEMENT, RECOMMENDATION, OR FAVORING BY EPRI.

THE FOLLOWING ORGANIZATIONS, UNDER CONTRACT TO EPRI, PREPARED THIS REPORT:

COMMONWEALTH SCIENTIFIC AND INDUSTRIAL RESEARCH ORGANISATION (CSIRO)

UNIVERSITY OF NEW SOUTH WALES (UNSW)

This is an EPRI Technical Update report. A Technical Update report is intended as an informal report of continuing research, a meeting, or a topical study. It is not a final EPRI technical report.

NOTE

For further information about EPRI, call the EPRI Customer Assistance Center at 800.313.3774 or e-mail askepri@epri.com.

Together…Shaping the Future of Energy®

ACKNOWLEDGMENTS

ABSTRACT

Globally, many advanced reactors (AR) designs—including light water small modular reactors (lwSMRs) and advanced non-light water reactors (non-LWRs)—are under development, preparing for further demonstration, under construction, or are operating. While Generation II and III LWRs will (for the foreseeable future) continue operation with established fuels and fuel forms and provide non-emitting firm power generation, the commercial deployment of non-LWR ARs will introduce new fuels and new high-level radioactive waste (HLW) streams requirement management on the back-end of the nuclear fuel cycle. The new waste characteristics associated with AR operations will introduce new needs, challenges, and opportunities to the owner-operator concerning irradiated fuel and HLW management.

This study explores the anticipated impacts of AR operation on the back-end management of the nuclear fuel cycle in terms of storage, transportation, disposal, and recycling activities. The study focusses on specific AR design variants for which adequate information was publicly available. These included a sodium-cooled fast reactor (SFR), pebble-bed and prismatic-block high-temperature gas-cooled reactor (HTGR) variants, and a solid-fueled version of the molten salt reactor (MSR)—the fluoride-salt-cooled high-temperature reactors (FHR). The contents presented within should be considered representative and are meant to drive further discussion. This study considers the following metrics as indicators of back-end management impacts:

- Radionuclide composition of waste streams in all groundwater pathways (i.e., clay, granite, salt) and human intrusion scenarios, heat generation, and waste form
- Expected volumes of non-LWR waste (expressed as cubic meter per gigawatt-year of electrical energy produced, m^3/GW_{yr})
- Aspects of treatments, disposal, and processing options for spent TRISO fuel, graphite fuel block, and FLiBe salt
- Aspects of recycling scenario for each non-LWR technology
- The comparative analysis between SFR, pebble-bed HTGR, prismatic-block HTGR, and solid fuel MSR and their waste management strategy

CONTENTS

LIST OF FIGURES

LIST OF TABLES

1
INTRODUCTION

1.1 Background

Globally, many advanced reactor (AR) designs, including small modular reactors (SMRs) based on light water reactors (LWR) technology and advanced non-LWR designs, are under development, preparing for demonstration, under construction, or are operating. While light water SMRs will continue operation with established fuels and fuel forms, there are less new challenges regarding waste management. Therefore, LWRs are not the focus of this evaluation and, instead, provide a useful, familiar benchmark against which the impacts of non-LWR technologies can be compared and understood. The expanding use of new fuels and generation of new high-level radioactive waste (HLW) streams arising from non-LWR operation will present owner-operators with new needs, challenges, and opportunities with respect to management of irradiated fuel and HLW.

This work is intended to inform key AR stakeholders, including potential owner-operators, regulators, and technology developers, on pressing and longer-term issues associated with management of irradiated fuel and associated HLW streams to reduce unintended consequences and premature elimination of options by offering a prioritized and time-phased landscape of this domain. This study focuses on non-LWR technologies.

1.2 Scope

The anticipated impacts of non-LWR operation on the back-end management of the nuclear fuel cycle (storage, transportation, disposal, and recycling activities) are explored and evaluated in terms of:

- Physical and chemical form of non-LWR waste streams
- Radionuclide composition and heat generation
- Expected volumes of non-LWR waste (expressed per NPP or per installed capacity)
- Aspects of recycling of actinides
- Limits on or requirements for compatibility with (1) borehole disposal waste packaging and/or (2) larger mined repository storage/transportation/disposal packages

The study focused on three leading AR technology concepts under active development and/or demonstration: sodium-cooled fast reactors (SFRs), high-temperature gas-cooled reactors (HTGRs), and the solid-fuel variant of molten salt reactors (MSRs)—the fluoride high-temperature reactor (FHR).

1.3 AR Technology Selection

AR technology selection for this review is provided in Table 1-1, together with a short explanation.

Table 1-1
AR technology concept variants selected for this exploratory study

	AR technology	Fuel Type	Purpose for Selection
1.	Sodium Fast Reactor	Metallic fuel	The near-term need with the largest open questions relates to the US Advanced Reactor Demonstration Project awardees.
2.	High-Temperature Gas Reactor	Pebble TRISO[*]	
		Prismatic TRISO	
3	Molten Salt Reactor	Pebble TRISO	Demonstrations planned for later in the decade or early in the 2030s.

[*] TRistructural ISOtropic coated particle fuel, commonly embedded in spherical or hexagonal graphite compacts.

2
METHODOLOGY

2.1 Methodology for Activation-Decay Calculations

All activation, heat, chemical composition and isotope composition data were produced using the FISPACT-II [1] software package; the software produces the time evolution of a specified material when exposed to a given neutron flux. FISPACT-II calculates the evolution of the inventory of nuclides in a target material that is irradiated by a time-dependent neutron flux. It does this by solving the coupled rate equations for the transmutation and decay of all nuclides present. Further quantities, such as the heat and activity, are then determined based on the number of nuclides of each isotopic species present and their intrinsic nuclear properties.

The following assumptions/steps were made in the use of FISPACT-II:

- The material composition simulated is assumed homogeneous. Only the fuel material is simulated, e.g., UO_2 fuel pellet, TRISO compact/pebble, etc.
- The energy spectrum of the neutron flux used is assumed to be homogenous throughout the material and constant as a function of time when in the reactor. The neutron flux is 0 when cooling.
- The material is simulated as being within the reactor until Time to burnup (EFPD divided by the capacity factor).
- The fuel power density of the material (W/cm^3) determines the number of neutrons present in the reactor. This fuel power density is assumed to be constant as a function of time when in the reactor.
- The fuel power density is calculated from the fuel burnup, time within the reactor and material composition. Burnup is converted from Gigawatt-days/metric ton-Uranium (GWd/t(U)) to GWd/t(fuel) material using the composition of the material. Fuel power density is then obtained by dividing by the time to burnup and multiplying by the mass density.
- 1 kg of material is simulated. Quantities are then normalized by energy production in Gigawatt-electric-years (GW_e-yr), assuming a traditional Rankine power cycle with 34% conversion efficiency.

2.2 Methodology for Waste Stream Characterization

For each of the three AR concept variants, waste streams are characterized using the following metrics:

- Volume of spent fuel, replaceable components (e.g., coolant fluid), and decommissioning components (e.g., reactor pressure vessel, steam generator). Volumes are expressed in m^3/ GW_e-yr of electricity production.
- Discharged fuel is further characterized regarding its decay heat production, total activity, elemental and radiological composition.

Decay heat production is expressed in three different ways to allow different interpretations depending on the issue being addressed:

- W_{th}/kg of fuel: illustrates intrinsic characteristics of the spent fuel expressed per mass of the fuel, which may be required in comparison with a decay heat limit used to maintain fuel cladding integrity and ensure retrievability
- W/m^3 of fuel: relevant for final disposal assessments regarding density of disposal (i.e., m^3 of fuel per volume of disposal gallery or cavern) in a geologic repository
- W/GWyr of electricity produced: allows comparison between AR technologies and the reference PWR of heat generation based on annual energy production

The same metrics are used for the total activity of the spent fuel, i.e., TBq per kg of fuel, TBq per m^3 of fuel, and TBq per GWyr of energy produced. Decay heat is one of the nuclear fuel management considerations and follows trends similar to those associated with dose.

The chemical composition of the spent fuel is calculated five years after cooling and is expressed in absolute values and relative to the chemical composition of a traditional Gigawatt-scale PWR plant discharged fuel. For a total of 99 chemical elements the mass has been produced, expressed in grams per kg of fuel, grams per m^3 of fuel, and grams per GWyr electricity produced.

Although the calculations produced nearly all possible radioisotopes in spent fuel, only a small set of radioisotopes are presented for discussion. For an assessment of the impact of the explored AR concept variants on disposal, the selection of these radioisotopes was based on consideration of two categories of commonly evaluated exposure scenarios for geologic disposal performance:

- Gradual release via the groundwater pathway (unperturbed scenario) in three host rocks
- Exposure due to inadvertent human intrusion

The gradual release scenario via the groundwater pathway considers disposal in a geological repository with multiple barriers. Release of radionuclides occurs very gradual over time periods of tens of thousands of years, whereby radionuclides are released from the repository into the surrounding geosphere. For this exploration, dominant radioisotopes were identified from performance assessments of geologic disposal systems in granite, clay, and salt host environments.

Granite performance assessment results are based on the Spanish reference granite repository concept. For this case, Enresa, the non-profit Spanish nuclear fuel management organization, primarily used GoldSim™ for modeling. This case assumes canister failure will occur gradually over the range of 1300 to 10,000 years. High-level waste is assumed to be immobilized in glass, which has an assumed constant corrosion rate over 72,000 years. Discharged fuel has a corrosion rate calculated with an a-radiolysis dependent leaching model, which corresponds to a lifetime for uranium oxide of 10 million years, and mixed oxide (mixed uranium and plutonium) of 1 million years. Peak dose is defined by I-129 [2], with Cs-135, minor actinides and fission products contributing to dose after around 1 million years, the latter groups being sorbed by the bentonite clay prior to that point. Clay performance assessment results were based on the reference repository concept from Belgium for disposal of HLW by the Belgian Nuclear Research Center (SCK'CEN). Their evaluations derive from modeling done in the SAFIR 2 report which performed safety evaluations. I-129 presence drives peak dose, with other contributions due to Sn-126, Se-79, and Tc-99. Minor contributions due to actinides occur after

several million years, due to the strong sorption in clay during the preceding years [2]. Salt performance assessment results are based on the German repository concept. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), or the Society for Plant and Reactor Safety, evaluated this concept for an altered evolution scenario. This considered groundwater intrusion via an anhydrite vein into the disposal facility. Modeling is based on work done for discharged fuel disposal in a salt formation, within a framework developed by the Spent Fuel Performance Assessment (SPA) project of the European Commission [2].

Initially migration in the geosphere is very slow due to the very low permeability of the host rock (rock salt, clay, or granite) and the near absence of any hydraulic gradient. As a result, radionuclide migration is predominantly by molecular diffusion, an inherently slow process [2]. Most of the radionuclides are also retarded by adsorption onto the various rock minerals; the combined effect of diffusion and adsorption causes transport through the host rock to take hundreds of thousands of years during which many radionuclides decay to negligible levels [2]. When the radionuclides subsequently end up in the groundwaters overlying or underlying the host rock, they are further dispersed to levels that are no longer harmful to humans or the environment. Based on previous post-closure safety assessments for the geologic disposal of spent fuel and high-level waste from advanced reactors, Table 2-1 shows the set of radionuclides of concern which were selected for consideration in this study [2]: C-14 ($t_{1/2}$ = 5.70E+3 years), Cl-36 ($t_{1/2}$ = 3.01E+5 years), Cs-135 ($t_{1/2}$ = 2.30E+6 years), I-129 ($t_{1/2}$ = 1.61E+7 years), Se-79 ($t_{1/2}$ = 1.10E+6 years), Ra-226 ($t_{1/2}$ = 1.60E+3 years), Sn-126 ($t_{1/2}$ = 2.3E+5 years), Tc-99 ($t_{1/2}$ = 2.14E+5 years), and Zr-93 ($t_{1/2}$ = 1.53E+6 years).

Table 2-1
Radionuclides of concern selected from groundwater exposure scenarios

Isotope	Half-Life (years)
Carbon-14	5.70E+3
Chlorine-26	3.01E+5
Cesium-135	2.30E+6
Iodine-129	1.61E+7
Selenium-79	1.10E+6
Radium-226	1.60E+3
Tin-126	2.30E+5
Technetium-99	2.14E+5
Zirconium-93	1.53E+6

These radionuclides of concern were selected from post-closure assessments for enriched uranium oxide (UOX) fuel with an average burnup of 50 Gigawatt-days/metric ton heavy metal (GWd/tHM or GWd/MTHM), with the spent fuel disposed in a geologic repository after 50 years cooling. The different types of host rock considered are: granite, clay, and salt [2]. The selected radionuclides are mainly long-lived fission and activation (C-14 and C-36 from activation of impurities in the fuel or cladding) products, of which several are mobile in the subsurface (e.g., Cl-36, I-129, Se-79) due to very low or zero sorption on various geomedia [3]. Actinides like Th-230 and Ra-226 give rise to a second peak in the dose rate; however, only after a very long

time (>one million years), and the peak dose is typically less than that of the fission and activation products. For this reason, only Ra-226 was selected as a radionuclide of concern, considered representative for the impact of actinides [2].

The second scenario concerns inadvertent human intrusion; the scenario assumes that during exploratory drilling, core material is brought to the surface that contains radioactive material from the disposed waste. The core is then analyzed in the laboratory by a geotechnical worker who is exposed (i) during grinding or cutting of the samples (inhalation dose), (ii) from direct external irradiation, and (iii) from ingestion associated with contamination of hands. This intrusion scenario assumes the core contains material from one disposal canister containing 4 UOX spent fuel assemblies. As the canister ages, the dose to the human worker from this intrusion scenario reduces, due to the impact that radioactive decay has on inventory of radioisotopes. The maximum dose to the hypothetical worker is associated with intrusion at the moment that institutional controls have been lifted from the repository site. Conservative assumptions set this date at 100 years post-closure. In this event, total peak individual dose can be calculated as 21 sieverts (Sv). If this intrusion occurs 10,000 years post-closure, total dose falls to 1.6 Sv. At 100,000 years post-closure, this dose is 0.9 Sv. Radionuclides of concern that were selected for consideration from this scenario are shown in Table 2-2.

Table 2-2
Radionuclides of concern selected from human intrusion exposure scenarios

Isotope	Half-Life (years)
Americium-241	4.33E+2
Plutonium-238	8.77E+1
Plutonium-239	2.41E+4
Plutonium-240	5.65E+3
Thorium-229	7.43E+3
Thorium-230	7.54E+4

A final group of radionuclides was selected for an assessment of the impact on storage and transport. The selection of radionuclides of concern is based on a study by Cumberland et al. [4], who listed the top 5 dose contributing radionuclides for dry storage cases containing 5-year cooled spent nuclear fuel assemblies. Dose rates are used to demonstrate compliance with regulatory requirements on radiation protection during interim storage and transport of spent nuclear fuel. These radionuclides are presented in Table 2-3.

Table 2-3
Radionuclides of concern selected for storage and transportation [5]

Isotope	Half-life (years)
Cobalt-60	5.27E0
Cesium-134	3.01E+1
Europium-154	8.59E0
Barium-137m	4.86E-6
Curium-244	1.81E1

2.3 References

1. Sublet, J.-C., et al., FISPACT-II: an advanced simulation system for activation, transmutation and material modelling. Nuclear Data Sheets, 2017. **139**: p. 77-137.

2. Lensa, W.v., R. Nabbi, and M. Rossbach, RED-IMPACT. Impact of partitioning, transmutation and waste reduction technologies on the final nuclear waste disposal. Synthesis report. 2008.

3. Freeze, G., et al., *Generic Deep Geologic Disposal Safety Case.* SAND2013-0974P, REV, 2013. **1**.

4. Cumberland, R., G. Radulescu, and K. Banerjee, *The Relationship Between Dose Rate and Decay Heat for Spent Nuclear Fuel Casks*. 2020, Oak Ridge National Lab.(ORNL), Oak Ridge, TN (United States).

5. Andrews, H.B., et al., *Review of molten salt reactor off-gas management considerations.* Nuclear Engineering and Design, 2021. **385**: p. 111529.

3
LIGHT WATER SMALL MODULAR REACTORS

Small Modular Reactors (SMRs) and light-water Small Modular Reactor (lwSMR), often used interchangeably, are commonly defined as power reactors up to 300 MWe, though several designs exceed that rating. Their components and systems can be fabricated in a factory and transported as modules to their designated sites for installation as demand arises. SMRs will employ passive safety features, have fewer parts and components, operate with smaller nuclear cores (and thus smaller source terms), and leverage their modular design to be constructed faster and at less capital cost to the customer [1, 2].

The key driving forces of SMR development are fulfilling the need for flexible power generation for a wider range of users and applications; for instance, replacing ageing fossil power plants, providing the opportunity of partial or dedicated use in non-electrical applications such as providing heat for industrial processes, hydrogen production or sea-water desalination. Process heat or cogeneration results in significantly improved thermal efficiencies leading to a better and quicker return on investment. Given their level of technology readiness, SMRs are also expected to contribute to decarbonization in the near-term [1, 2].

In the U.S., current lwSMR designs include NuScale and SMR-160 [2]. NuScale has received the approval of its design certification by the US Nuclear Regulatory Commission (NRC), SMR-160 is at the preliminary design stage, while the UK SMR is entering the licensing process [3]. The SMR fuels are rather well known regarding their form, enrichment, and burnups (Table 3-1). Some characteristics that occur in several designs of lwSMR fuels are:

- UO_2 fuel assemblies are shorter than conventional LWR
- Discharged fuel storage will likely involve the wet storage followed by dry storage
- Storage and transport casks may need to revise loading geometry due to size differences
- Spent fuel waste per GWe is similar to conventional LWR; slightly larger % of the structural waste such as the pressure vessel has a larger fraction of total surface to volume ratio
- Spent fuel composition expected to be similar to conventional LWR spent fuel with similar disposal pathways

Table 3-1
Fuel characteristics of selected SMR design variants [1, 2]

SMR Design	Fuel Form/Type	Enrichment	Burnup (GWd/ton)	Designers
NuScale	UO_2 pellets; FA based on standard PWR	< 4.95%	>30	NuScale Power Inc.
SMR-160	UO_2 pellets; existing technology	4.95%	45	Holtec International
BWRX-300	UO_2	3.4% (avg)/ 4.95% (max)	49.5	GE-Hitachi Nuclear
UK SMR	UO_2	4.95% (max)	55-60	Rolls Royce

Water-cooled SMR developers, in general, adopt radioactive waste management plans similar to that of operating advanced water-cooled reactors. Advances are also being made in the dry storage technologies. Holtec International, the developer of the SMR-160, has developed the Multi-Purpose Canister MPC-37 that enables the on-site storage of all spent fuel for the life of the plant within an array of the HI-STORM UMAX modules, an underground vertical storage cask design. Another advantage of SMRs is the reduction in dose to personnel in decontamination and decommissioning, a reduced need for inspections of the reactor pressure vessel, and a reduction in the activation of steel components [1].

3.1 References

1. IAEA, Advances in Small Modular Reactor Technology Developments. 2020, IAEA: Vienna.

2. IAEA, Advances in Small Modular Reactor Technology Developments. 2018, IAEA: Vienna.

3. Blain, L. *US nuclear regulator greenlights its first small modular reactor*. 2022; Available from: https://newatlas.com/energy/nrc-certifies-nuscale-nuclear/.

4

SODIUM-COOLED FAST REACTORS (SFRS)

4.1 Introduction

Fast Reactors (FR) or fast-spectrum reactors utilize higher-energy neutrons which allow for direct fission of abundant isotopes like U-238. They are capable of breeding fissile isotopes and consuming many isotopes present in discharged LWR fuel, including uranium/plutonium and the transuranic (TRU) elements. Several coolant types have been evaluated for FRs, leading to technologies such as the sodium-cooled fast reactor (SFR) selected for this evaluation, the lead or lead-bismuth cooled fast reactor (LFR, LBFR), and the gas fast reactor (GFR).

Fast-spectrum reactors can utilize a range of U/P fuel incorporated into a wide variety of forms, including:

- Oxides
- Carbides
- Nitrides
- Metals

The SFR is a fast reactor cooled by liquid sodium metal. Sodium as a coolant offers excellent heat transfer properties with a thermal conductivity of 142 W/m·K and operating temperatures without pressurization of the reactor pressure vessel (RPV) commonly around 500°C. Sodium has a low melting point (98°C) and high boiling point (883°C) relative to this operating temperature and is less corrosive than some metal coolants.

4.2 SFR Technology Variant

A 1475 MW_{th}/600 MW_e SFR is used for this exploratory study. This is based on a design that includes a sodium primary coolant loop, a sodium intermediate coolant loop, and a steam (Rankine) power conversion cycle. The fuel is metal uranium-zirconium [1]. There are two types of fuel assemblies in the selected concept variant SFR core: driver fuel assemblies and feed assemblies. Both assemblies use U-10%Zr metallic fuel with ferritic-martensitic stainless steel clad. The driver fuels consist of 16.5% enriched fuel while feed fuel assemblies are made of depleted uranium (0.3% enrichment) [1]. The core sits near the bottom of a reactor vessel, which is enclosed within a guard vessel. The core also has two rows of reflector assemblies made of steel rods and one row of B_4C shield assemblies. The feed assemblies with their fuel pins are typically arranged on a hexagonal pitch.
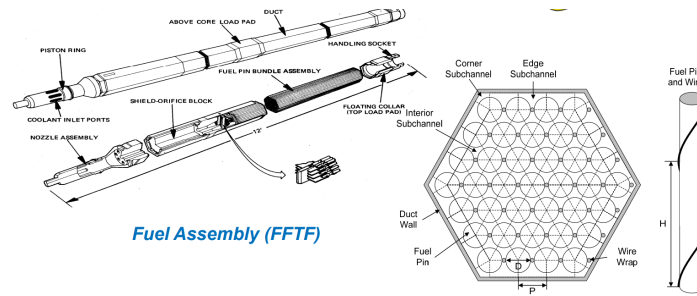
Figure 4-1
A typical SFR fuel assembly and their spatial arrangement [2]. Image in Public Domain, courtesy of Argonne National Lab

The SFR core employs control and safety assemblies, which consist of movable control rods that act as the reactor's primary reactivity control and safety assemblies that provide redundant shutdown capabilities [1]. During the reactor operations, these safety assemblies will be completely withdrawn above the core and are only required when the primary control rods are unable to shut the reactor down due to an extremely unlikely failure.

4.3 Waste Identification and Characterization

To identify the key waste streams for HLW and potential intermediate level waste (ILW) of concern, several assumptions have been made for the calculation of SFR discharged fuel volume and for the activation calculations. The calculations focus on the primary pool configuration which consists of the reactor core, primary pump, primary Intermediate Heat Exchanger (IHX) and primary sodium. Thus, the steam turbine generator will not be considered as part of the active volume calculation.

- Due to the scarcity of IHX data for the SFR, its specifications are assumed to be the same as an Advanced Burner Reactor due to its similarity in pool-type reactor design [3];
- The SFR is a pool-type SFR. Thus, the volume for liquid sodium is assumed to be the same as the RPV's solid volume [1].
- SFR control and safety lifetime period is assumed to be similar to fast reactor control rods such as BN-600 and Superphénix [4].
- The replacement period for the steel reflector and shield assemblies are assumed to have the same replacement period as the feed fuel assemblies; 495 EFPD × 11 cycles [1].
- Burnup is assumed at an average value of 150 GWd/tHM.
- The replacement period for all heat exchanger vessels is assumed to be 20 years [5];
- The wall thickness for the HTX vessel is the assumed identical to the RPV's wall thickness.

Calculated waste volumes for the different waste streams are graphically presented by means of a Sankey diagram. One of the characteristics of this diagram relates to the thickness of the lines: the greater the volume of waste involved, the thicker the line. The most notable stream for this SFR variant is the sodium coolant, with an annual production of 126 m^3 per GWe·yr. One of the SFR design approaches is to minimize the amount of sodium in the core through reducing the gaps between the ducts and keeping the amount of coolant between fuel pins as minimal as possible by using of a tight pitch hexagonal lattice [1]. This study assumed the sodium would completely fill up the reactor pressure vessel for a total of 40 years of core lifetime.

Figure 4-2 shows the volume of waste arising from the SFR design variant selected. This SFR produces a smaller volume of discharged fuel waste (3.57 m^3/GWe-yr) compared to the representative large PWR (5.68 m^3/GWe-yr). This smaller volume of fuel is primarily due to the higher burn-up that arises from the reactor design. As the reactor is breeding more fissile materials from the driver fuel assemblies, more actinides are produced and burn in-situ relative to the fuel. While this leads to a higher fuel utilization, it also affects the discharged fuel compositions, as is shown later. All the non-fuel waste streams are considered low-level and intermediate-level waste consistent with the IAEA classification of these wastes [2].
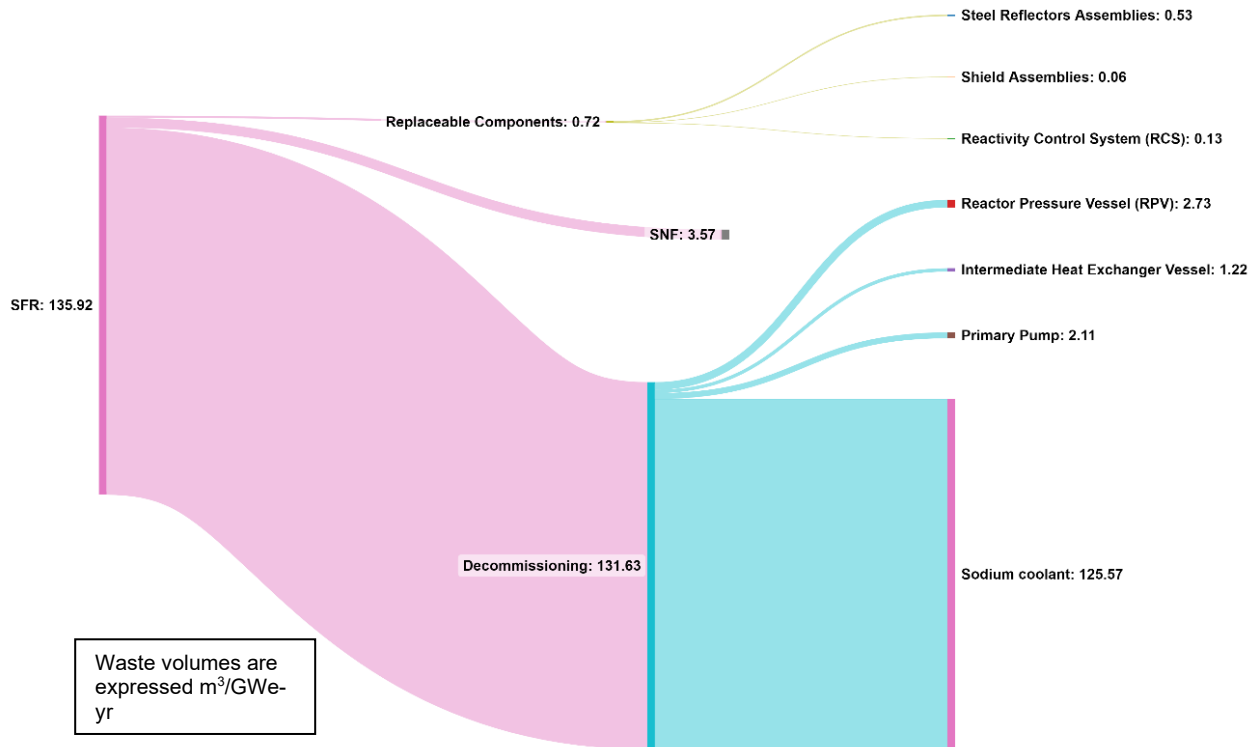
Figure 4-2
Sankey Diagram showing annual volumes of generated waste (m³/GWe-yr) and contributions from various reactor components for SFR

Limitations of the FISPACT-II code on neutron flux beyond the core limit the ability to precisely model the activity of waste streams beyond the discharged fuel. Due to the variance in plant layout that inherently contribute to the flux present in non-fuel systems, it was chosen to represent those waste streams with only a volume calculation. An assumed categorization of non-fuel wastes as intermediate level waste is likely bounding, and costs can be estimated based on the volumes presented with more specific regional disposal information.

The calculated total activity of the spent fuel and the sodium coolant is presented in Figure 4-3 as the upper graph, normalized by the total energy production (Bq/GWe-yr). For the first 100 years, the total activity for SFR and PWR discharged fuel are of the same order of magnitude. From 100 year onwards, a faster drop in activity is observed for SFR fuel compared to PWR fuel.

The decay heat is expressed normalized by total energy production in the lifetime of the reactor (W_{th}/GW$_e$yr). The lower image in Figure 4-3 shows that until approximately 10 years after release from the reactor, the SFR fuel has a similar decay heat profile to that of the PWR but lower in magnitude. From

approximately 10 years onwards, the decay heat profile for SFR starts to deviate from that for the PWR, characterized by a more rapid decay with a maximum difference of about one order of magnitude in decay heat. This is potentially due to the lower concentration of Am and Cm, as detailed below.
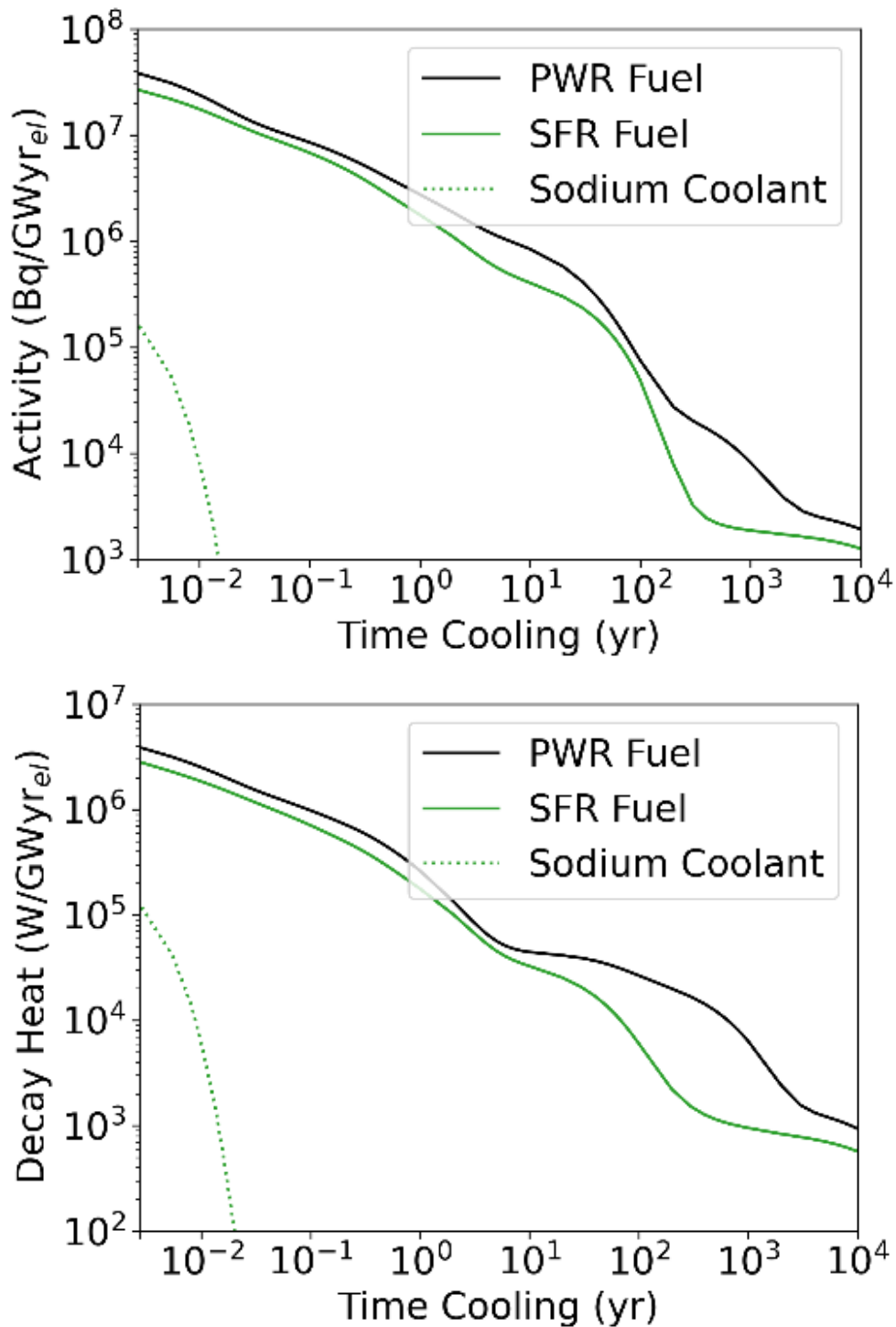
Figure 4-3
Calculated total activity and decay heat for SFR fuel and sodium coolant normalized by annual electricity production

Coolant activity and depletion have also been considered as part of this study to project if there is significant activity for disposal considerations. The need to replenish the coolant due to loss of the chemicals through transmutation was also considered, but limited to nuclear effects and no other factors, such as chemical reactions, impurities or corrosion. The coolant's activity and radioactive heat, which can be seen as the dotted line in Figure 4-3, were found to be more than two orders of magnitude lower than the fuel at discharge and became of comparable magnitude only after approximately 100 years. After approximately 300 years, there's a significant drop in coolant activity. Sodium-23 (Na-23) is the main isotope that is depleted within the coolant; however, after 15 years within the reactor, the isotope has a marginal fraction of depletion, 4.8×10^{-5}. Importantly, most of the transmuting isotopes activate and decay into isotopes of the same chemical species, thus there is limited loss of chemical species in sodium fuel during reactor operations. This depletion curve is shown in Figure 4-5.
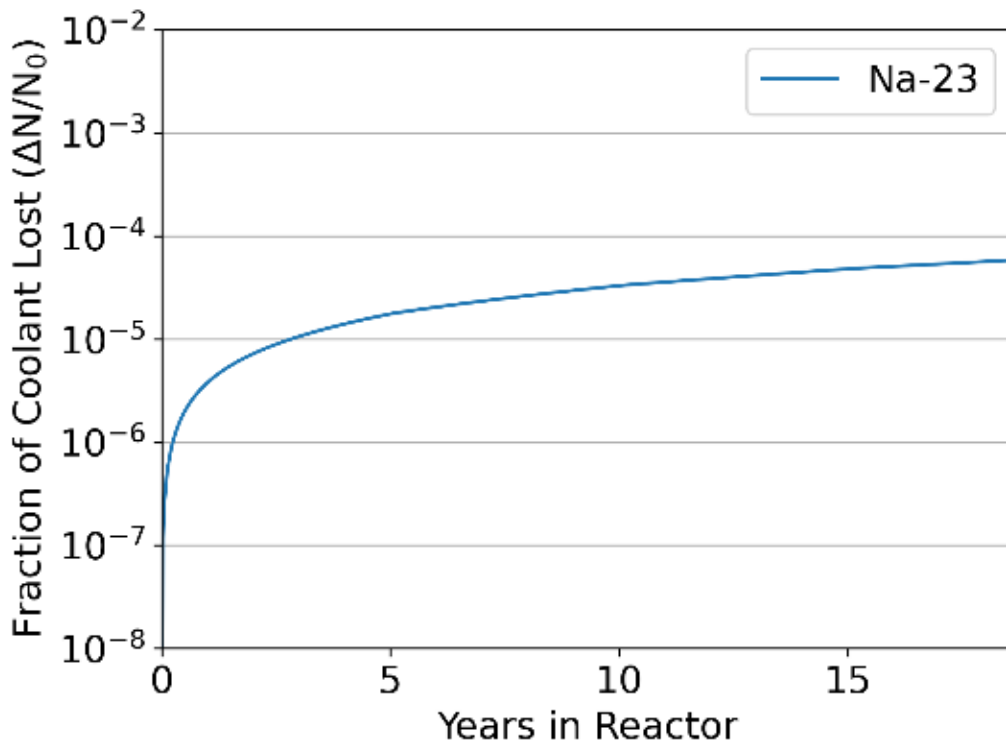
Figure 4-4
Sodium coolant depletion by activation and transmutation

The elemental composition of the SFR discharged fuel five years after removal from the reactor is compared to that of the PWR discharged fuel; consistent with the previous metrics, the elemental composition is expressed in g/GWe-yr in Figure 4-5. For the selected concept variant, SFR produces a smaller amount of Am, which is consistent with the higher burn-up and faster neutron spectrum of the SFR reactor, allowing more fission of TRUs. Depending on the selection made from among fast reactor design variants, this may vary. Fission products appear to have a nearly identical yield for SFR and PWR, with mercury ($Z = 80$) being the exception. Much greater differences are observed for light elements, where SFR spent fuel is practically devoid of Li, Be, B, C, N, O, and F.

Actinides
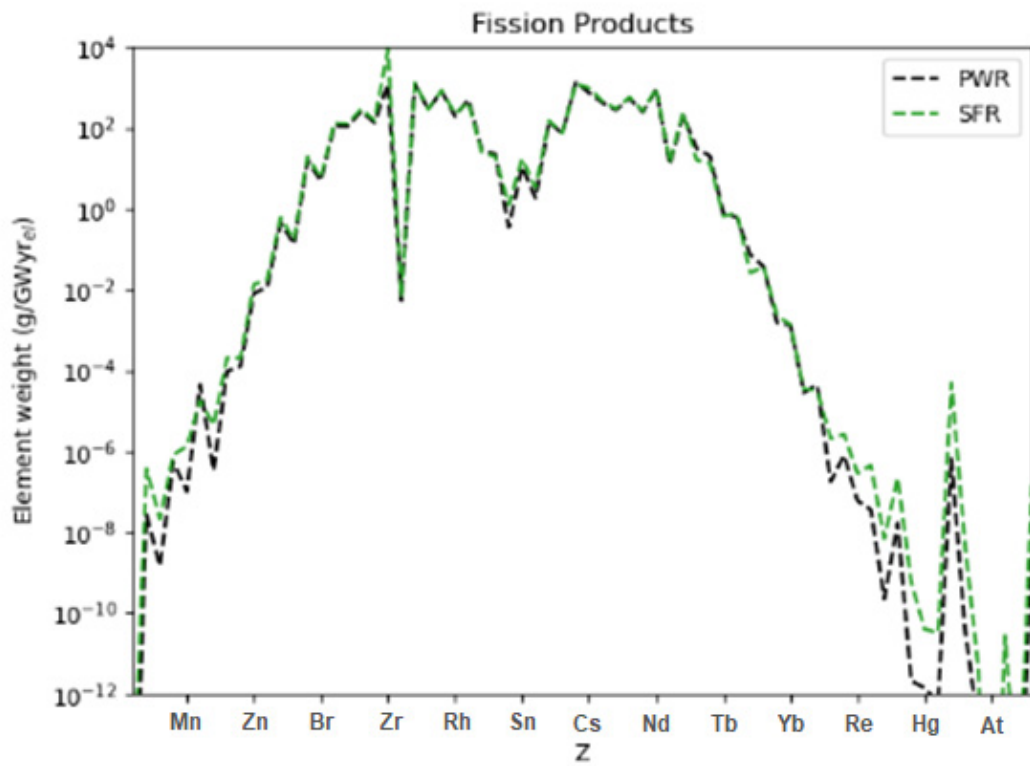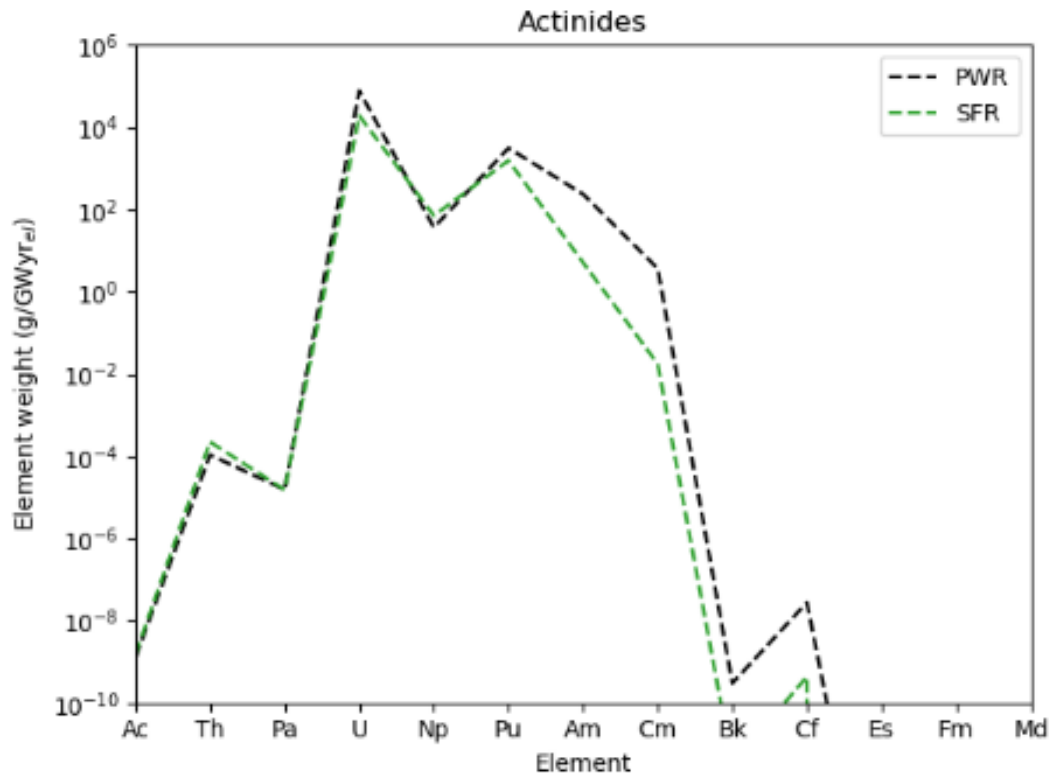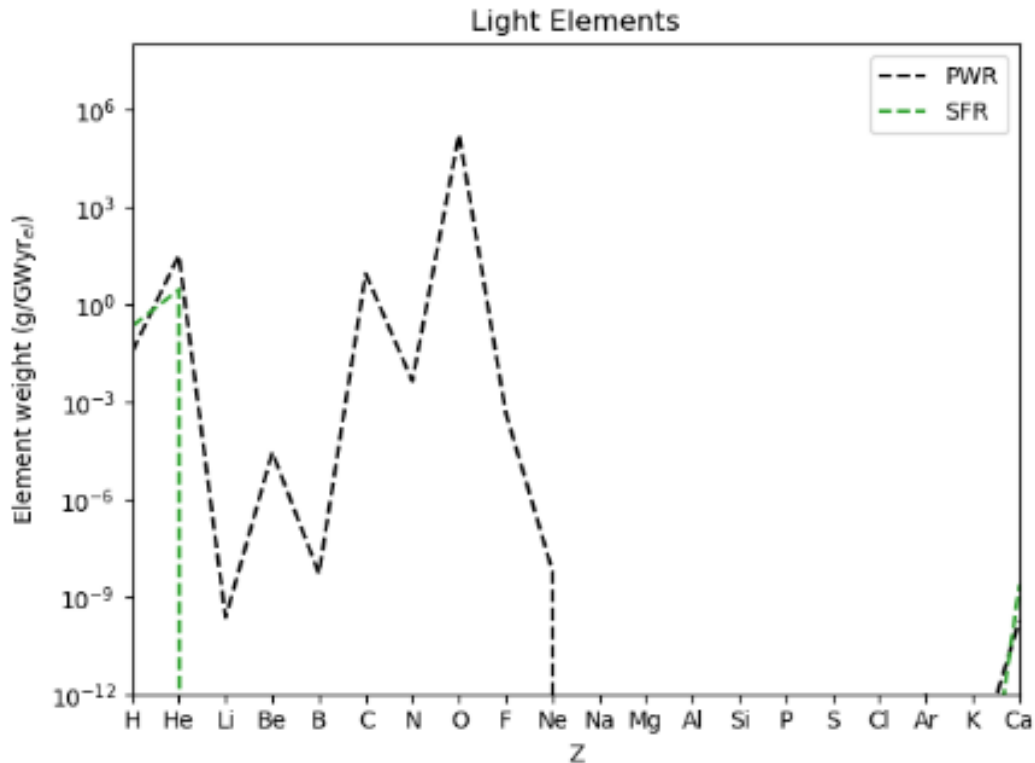
Fission Products

Figure 4-5
Chemical composition at 300 years for SFR and PWR: actinides (top), fission products (middle), and light elements (bottom)

The radionuclide composition for discharged SFR fuel 300 years after release from the reactor is compared to that for discharged PWR fuel. The compositional difference is shown in Figure 4-6. Radionuclides of concern are those that are main contributors to the total dose to humans in the case of geological disposal, when human exposure is either via the groundwater pathway or via inadvertent human intrusion. In all three host rocks, the variant SFR fuel contains much greater activities of radioisotopes of concerns than in the PWR fuel, with the exception of C-14, which is not anticipated to be present in meaningful quantities. Particularly stark differences are observed for Cl-36 (which is produced in trace amounts in PWR) and Cs-135 (14.1× in SFR). For the human intrusion scenario, different radionuclides of concern are at play. Except for one isotope (Am-241, which is in part consumed due to the deep burn present in this SFR variant), SFR discharged fuel produces higher activities for all other radionuclides of concern compared to the PWR.
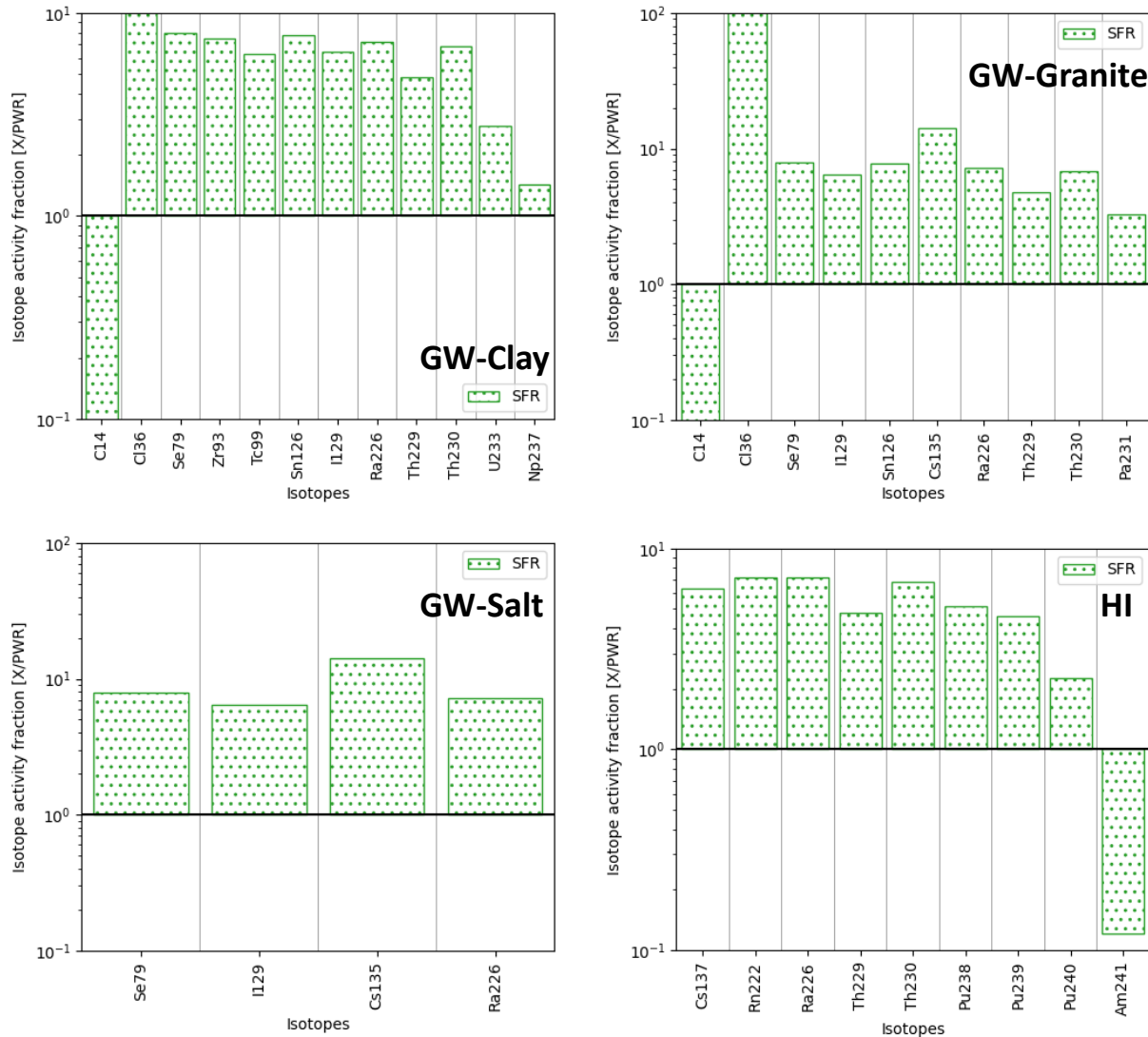
Figure 4-6
Comparison in isotope composition of discharged fuel between SFR and PWR at 300 years for groundwater scenario (GW-Clay, GW-Granite, GW-Salt) and human intrusion scenario (HI). Isotope activities in fraction of PWR reactor (>1 means isotope in SFR has higher activity than PWR and vice versa for <1).

4.4 Waste Form Options

One concept being considered for the back-end of metallic fuels is recovery of the uranium and plutonium via an electrochemical (also known as pyroprocessing) method. For metallic fuel, the chloride-based waste stream generated during pyroprocessing is a LiCl-KCl stream, which results from the treatment of metallic fuel [7]. The traditional approach to treat these waste streams is direct immobilization for disposal in a waste form suitable for chloride-bearing waste, such as glass-bonded sodalite or, a tellurite glass. Alternatively, separation of fission products concentrated into a reduced volume of electrolyte allows for recycling of the remaining salt [7].

For the residual metallic fuel waste that did not dissolve into the processing salt during electrorefining, metal waste forms are developed by using the hardware and cladding hulls from the fuel assembly that are added to the electrorefiner together with the chopped fuel segments [7]. These include steel (or Zircaloy cladding hulls in some fuel forms) that remain on the fuel segments which make up the majority of the hardware, along with any contaminated plenums, end pieces, or ducts [7].

4.5 References

1. Hejzlar, P., et al., *Terrapower, LLC traveling wave reactor development program overview.* Nuclear Engineering and Technology, 2013. **45**(6): p. 731-744.

2. Hill, D.R. Sodium Cooled Fast Reactors (SFR) in Gen IV International Forum. 2016. Argonne National Laboratory, USA.

3. Cahalan, J., et al., *Advanced burner reactor 1000mwth reference concept*. 2007, Argonne National Lab.(ANL), Argonne, IL (United States).

4. IAEA, Absorber materials, control rods and designs of shutdown systems for advanced liquid metal fast reactors. 1995, IAEA.

5. Oh, C.H., E.S. Kim, and M. Patterson, *Design option of heat exchanger for the next generation nuclear plant.* Journal of Engineering for Gas Turbines and Power, 2010. **132**(3).

6. (IAEA), I.A.E.A., Waste from Innovative Types of Reactors and Fuel Cycles. A Preliminary Study, in IAEA Nuclear Energy Series. 2019, IAEA: International Atomic Energy Agency (IAEA).

7. Frank, S., et al., *Waste Stream Treatment and Waste Form Fabrication for Pyroprocessing of Used Nuclear Fuel*. 2014, Idaho National Lab.(INL), Idaho Falls, ID (United States).

5

HIGH-TEMPERATURE GAS-COOLED REACTORS (HTGRS)

5.1 Introduction

5.1.1 High-Temperature Gas-Cooled Reactors (HTGRS)

High-temperature gas-cooled reactors (HTGR) are an advanced reactor design that operate at high-temperatures (above 700°C) and use helium as a coolant. The fuel uses coated compounds of uranium (tristructural-isotropic (TRISO) coated particle fuel with graphite as the bulk binder phrase). Due to the high-temperature environment, these reactors may be able to support a wide range of industrial processes requiring large amounts of heat or steam, which may impact siting decisions.

5.1.2 Pebble-Bed HTGR

The 200MW$_{th}$/76 MW$_e$ pebble-bed HTGR design is scaled down from the large LWRs, aiming to improve economics through the reduction of construction times, design modularization and simplification. The pebble-bed HTGR is based on the operations of Pebble Bed Reactors from around the world with a core life of 60 years [1]. Its design specifications are similar to the Advanced Gas Reactors (AGRs)-5/6/7 [2], which consists of helium as the reactor coolant, graphite reflectors as moderators to slow down the neutrons to thermal energies and a steam generator for electricity generation and process heat applications.

The pebble-bed HTGR fuel is comprised of hundreds of thousands of graphite pebble fuel elements each containing over ten thousand of uranium oxycarbide (UCO) TRISO particles (though multiple uranium materials can be used in TRISO). A depiction of one kind of TRISO fuel is shown in Figure 5-1. The pebbles contain UCO kernels enriched at 15.5 wt% and have smaller diameter compared to the UO$_2$ fuel kernels used in Germany and China [1, 3]. For the fuel handling system, as the pebble fuel passes through the reactor, its burnup will be measured to determine the amount of useful fuel available. The fuel will keep recycling through the reactor until it reaches the target burnup before being deposited into a spent fuel cask [1].
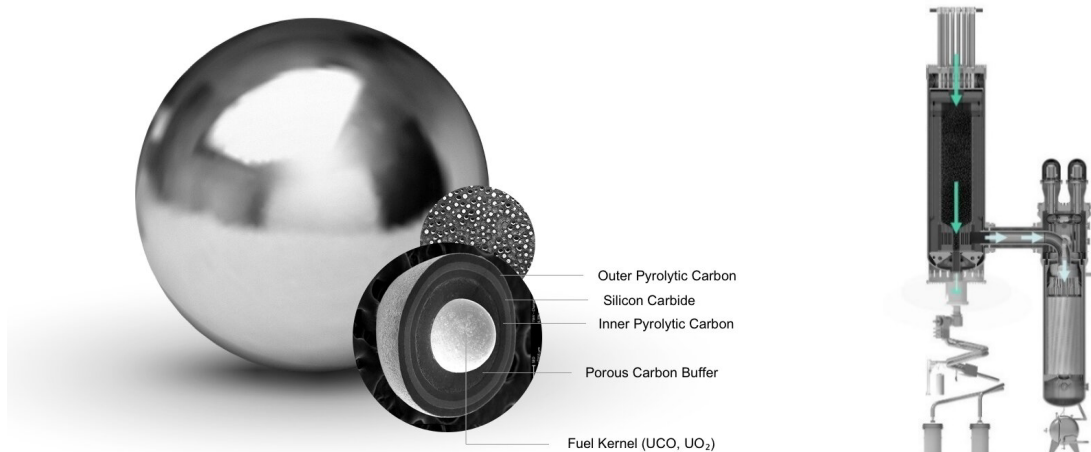
Figure 5-1
Illustration of graphite pebble fuel for the X-Energy HTGR (left) [4] and the components of their Pebble-bed HTGR (right) [5]. Images Copyright © X-Energy LLC. Used with Permission.

5.1.3 Prismatic-Block HTGR

The 600MW$_{th}$/275 MW$_e$ prismatic-block High-Temperature Gas Cooled Reactor (HTGR) consists of three functionally-oriented pressure vessel units; the reactor core, the heat exchanger and the direct-cycle gas turbine to generate power while circulating the reactor coolant [1, 6]. The prismatic-block HTGR uses TRISO particles that contain UO$_2$ fuel kernels rather than UCO. An estimated ten thousand particles are compacted and assembled into a graphite-clad fuel rod [1]. The fuel rods consist of vertically-piled fuel compacts. These are inserted into coolant channels in a hexagonal graphite fuel block, forming annular-shaped coolant channels [1, 7]. The core consists of fuel columns like these arranged in annular rings [7]. The core also comprises inner reflector columns and outer reflector columns. Both columns are carbon-based materials and replaceable along with the fuel block and control rods. Surrounding these replaceable reflector regions are the fixed graphite reflectors.

5.2 Waste Identification and Characterization

Several assumptions have been made for the calculation of HTGR discharged fuel volumes and for the activation calculations.

- TRISO fuel to have the same specifications for all HTGR reactors, except for inner kernel composition.
- The downcomer width for the pebble-bed HTGR is assumed to be 5 cm [8].
- The wall thickness for HTX vessel and Gas-Turbine Vessel are assumed to be the same as the RPV's wall thickness.
- The replacement period for all heat exchanger vessels and steam turbine/generator vessels is assumed to be 20 years [10, 11].
- The RPV and internal structures are designed for a 60-year life [1]. For this evaluation, these will include the core, graphite reflectors and the Reactivity Control System (RCS).

- The effective core height for pebble-bed HTGR is assumed to be from the top of the core to the top of the pebble outlet cone. The piping legs (i.e., hot gas ducts) have a diameter of 900 mm and are based on a similar HTGR design: HTR-10 [9].
- To calculate the volume of 1 spent fuel block (excluding the fuel rods), the total volume for all fuel rods per one fuel block (m^3) is subtracted from the total volume of one fuel block (m^3)
- The fuel rod is assumed to be a solid 39-mm diameter cylinder block (derived from the coolant channel diameter).
- Burn-up is assumed to be 165 GWd/tHM for the Pebble Bed variant, and 120 GWd/tHM for the Prismatic variant.

5.2.1 Pebble-Bed HTGR

Compared to the typical burn-up of a PWR (49 GWd/t), the pebble-bed HTGR variant has a higher burn-up of 165 GWd/t with a lower TRISO fuel power density of 6.3 W/cm^3 [5]. Based on the pebble fuel dimensions, fuel power density, and reactor operating parameters, the calculated discharged TRISO fuel produces a HLW stream with an energy-equivalent volume of 85 m^3/GWe-yr, or about fifteen times greater than the PWR fuel volume. Figure 5-2 illustrates these waste stream volumes. Unlike the PWR where most of the fuel assembly volume is UO_2, the pebble-bed HTGR fuel includes the entire irradiated pebble, therefore its volume originates mainly from the bulk graphite binder material between TRISO particles, as well as the concentric layers that coat the kernel (12% by volume of a TRISO fuel particle is the UCO kernel).

With a RPV and internal structure design accounting for a 60-year life [1], the decommissioning waste volume is estimated at 67.7 m^3/GWe-yr. The main contributors are the graphite reflectors surrounding the pebble-bed core and the steam generator vessel. The latter also contains the helical coil tube bundle which acts as the heat exchanger. Despite the RPV for a pebble-bed HTGR being smaller in size compared to the PWR, the calculated energy-equivalent waste volume is higher. However, the majority of this waste is graphite, and therefore expected to have lower specific activity than the average waste arising from PWR reactors. All the non-fuel waste streams are considered low level and intermediate level waste consistent with the IAEA classification of these wastes [12]. The volume of waste that is graphite may have other challenges associated, as the presence of C-14 may drive treatment of this material as long-lived ILW. In some regions like the U.S., this waste stream may be classified as HLW as a result.
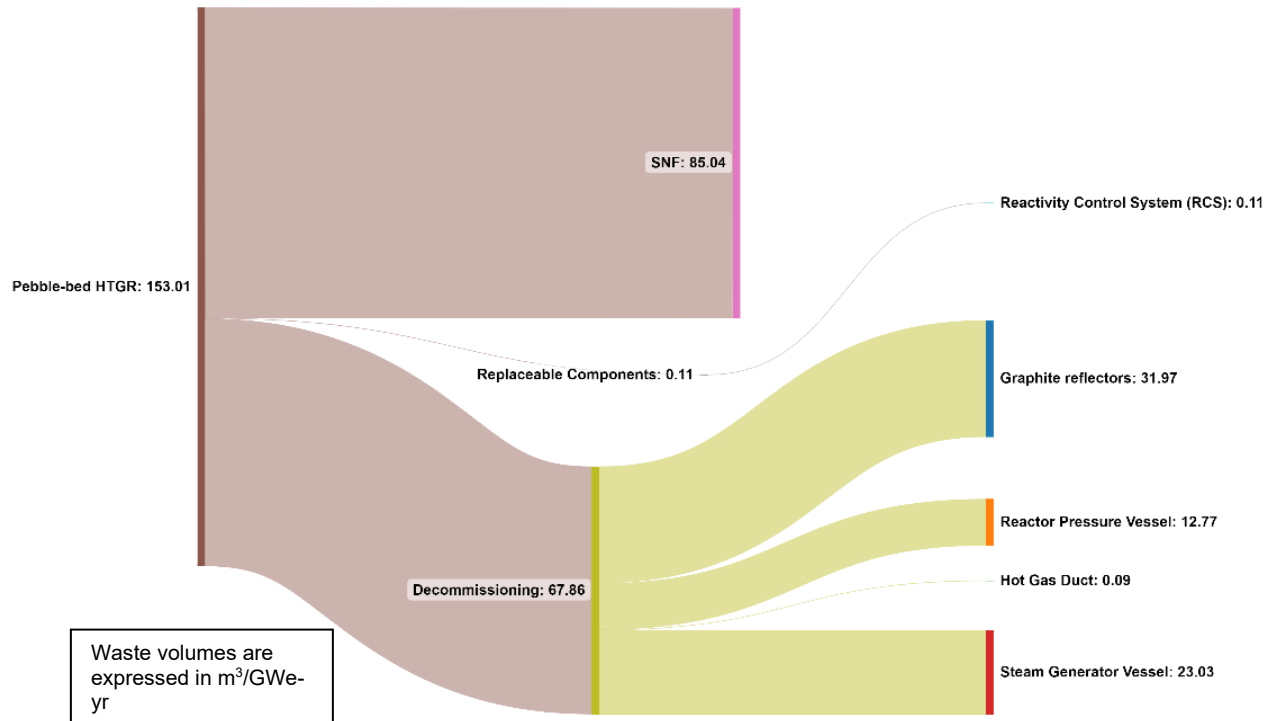
Figure 5-2
Sankey Diagram showing annual volumes of generated waste (m³/GWe-yr) and contributions from various reactor components for the pebble-bed HTGR

5.2.2 Prismatic-Block HTGR

The prismatic-block HTGR variant also employs TRISO-coated particles that contain UO2 fuel kernels enriched to 14% of U-235 [1]. With a higher burnup of 120 GWd/t [1], 269 tonnes of burned TRISO fuel are required to produce 1 GW of energy every year, which is equivalent to 150 m³/GWe-yr. However, the characteristics of prismatic-block HTGR fuel (i.e., a single hexagonal fuel block consisting of compacted TRISO-coated particles vertically piled in fuel rods [7]) contribute to a larger spent fuel volume of 344 m³/GWe-yr, shown in Figure 5-3. Several options of separating the graphite block and discharged TRISO fuel exist and will be discussed later in this report. All the non-fuel waste streams are considered low-level and intermediate level waste consistent with the IAEA classification of these wastes [12], with the same qualification of the graphite waste streams as in the Pebble Bed variant.
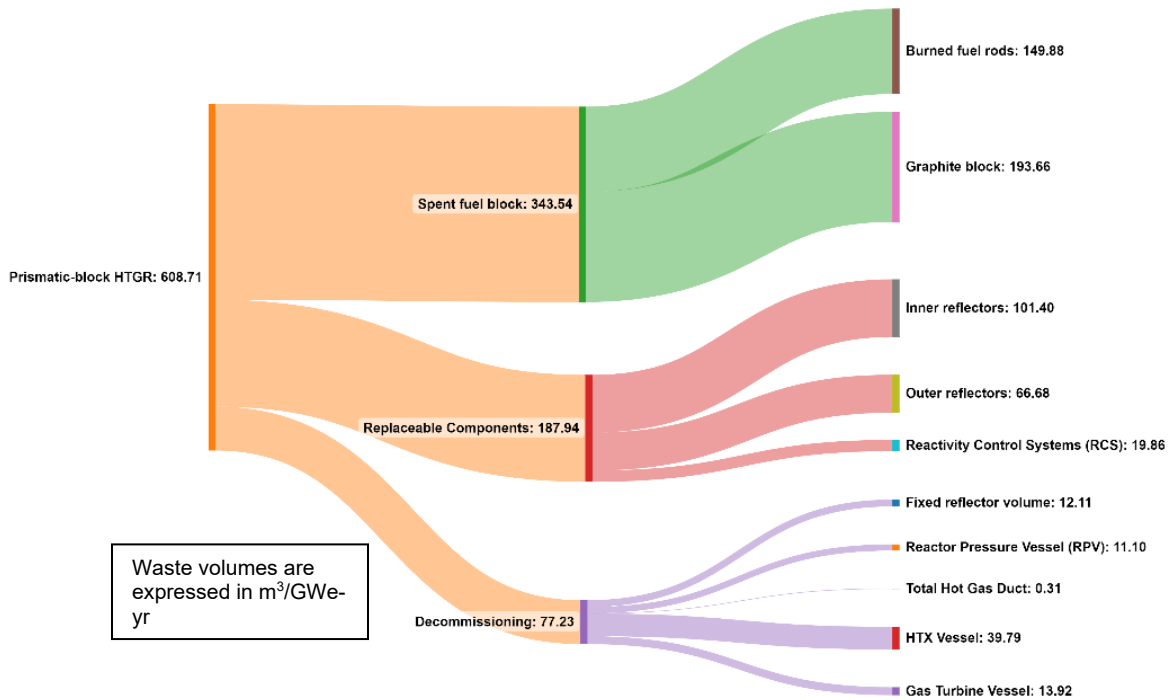
Figure 5-3
Sankey Diagram showing annual volumes of generated waste (m³/GWe-yr) and contributions from various reactor components for the prismatic-block HTGR

Figure 5-4 compares the calculated activity and decay heat of discharged fuel from the HTGR compared to PWR. The HTGR fuel shows a similar decay curve to the PWR fuel, with a slightly lower activity and heat for HTGR at all times after release from the reactor. The difference increases with time, and at 10,000 years it is about half on order of magnitude. At times close to discharge from the reactor, the decay heat and activity of HTGR fuel is around 10% lower.

The lower activity and heat of discharged HTGR fuel is consistent with the lower density of fissile material in the fuel, as the fuel element is composed primarily of carbon. This can impact the disposal considerations for these fuel forms. In the U.S., this irradiated graphite waste is excluded from disposal in the proposed geologic repository. In order to dispose of TRISO fuel kernels, current waste acceptance criteria compel some form of conditioning to remove the bulk graphite. This graphite must then be disposed of separately, possibly as HLW, as noted above. The remaining TRISO kernels would require further evaluation to understand optimal loading to avoid criticality concerns.
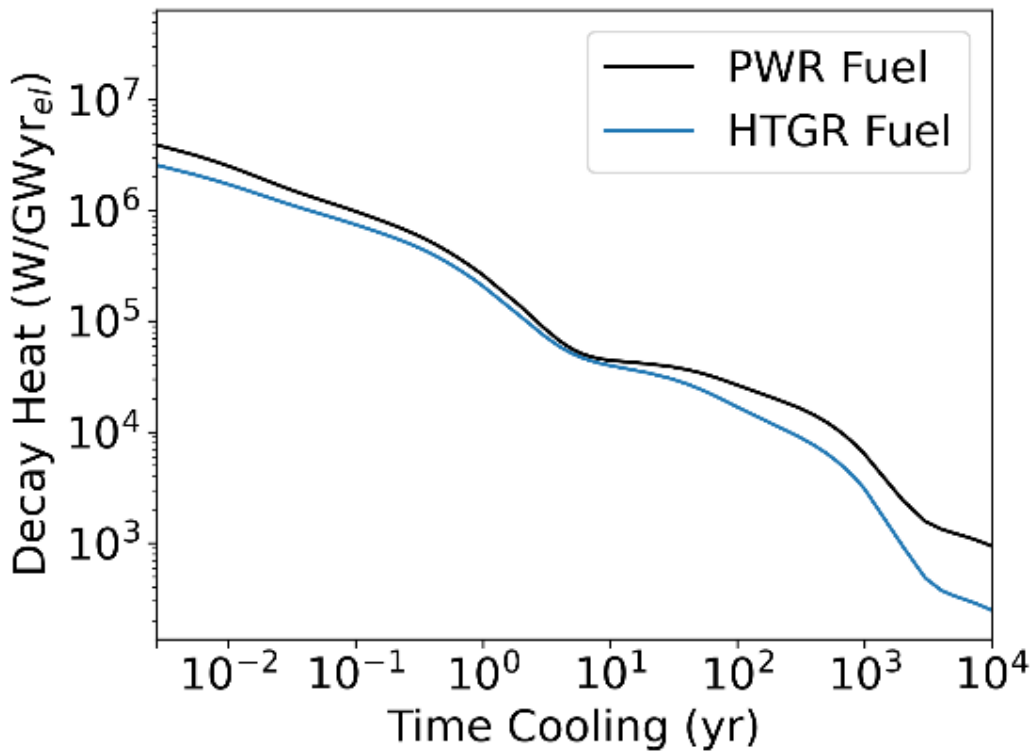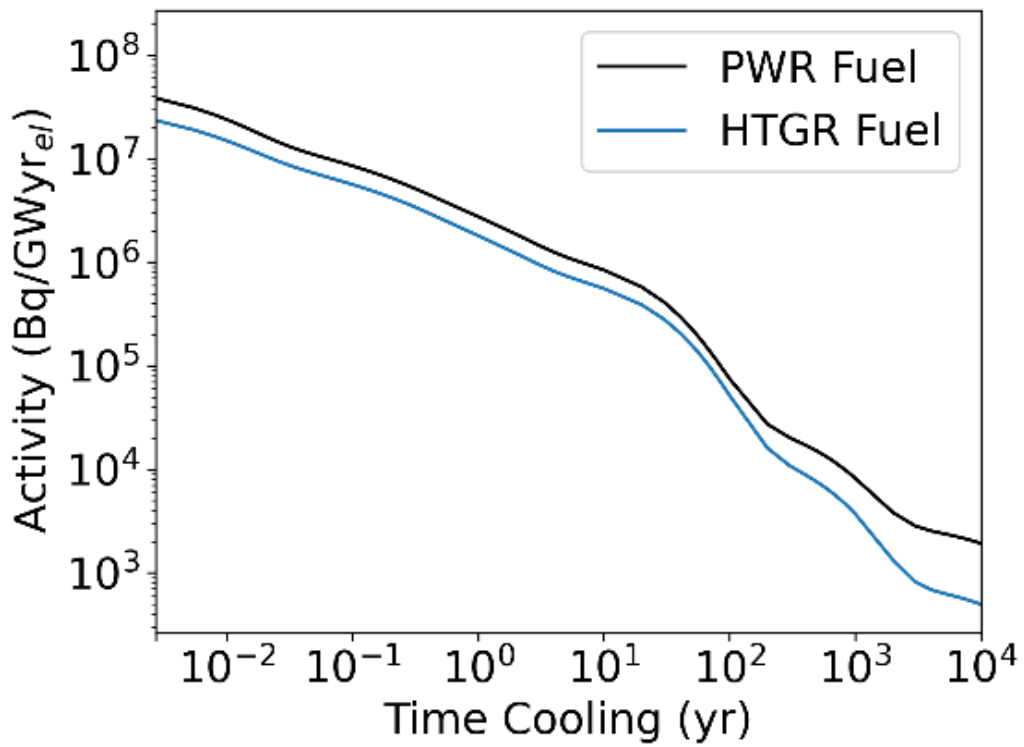
Figure 5-4
Calculated activity (top) and decay heat (bottom) for HTGR and PWR

TRISO fuel elements discharged from HTGRs, in both the pebble-bed and prismatic block designs, are expected to better retain radionuclides than PWR fuel, due to the structural integrity of the fuel kernel in the TRISO fuel particles [13]. A comparison between HTGR and PWR regarding the elemental composition of their fuel 5 years after discharging shows a nearly identical composition for elements that make up the actinides and fission products, as can be seen in Figure 5-5 in the top graph. Much greater differences are observed for light elements, where HTGR fuel has much less Li, Be, and C. The largest difference exists for elements Na, Mg, Al, Si, P, and S, where the PWR fuel has trace amounts of mass compared to more noticeable amounts for HTGR fuel, shown in Figure 5-5.
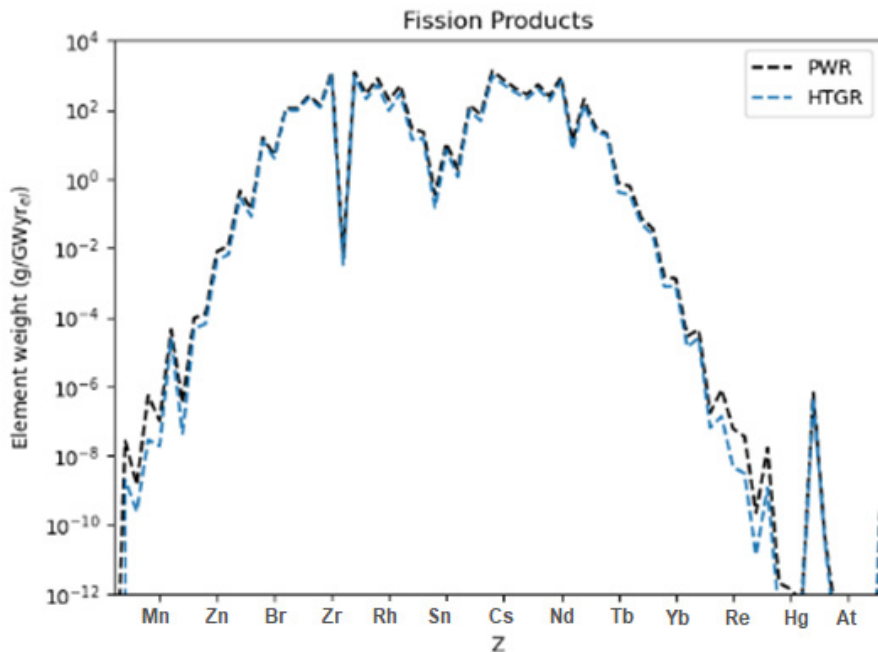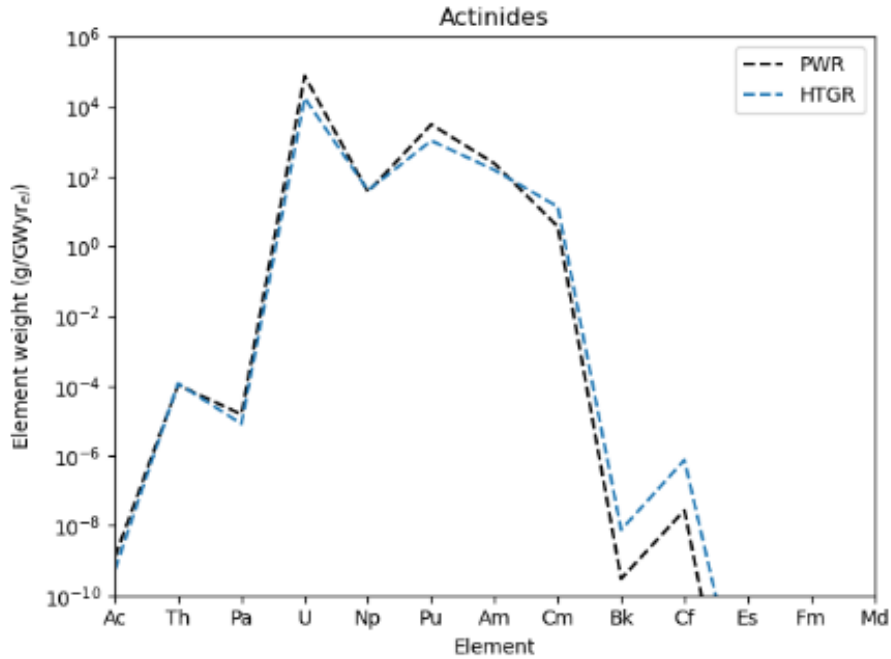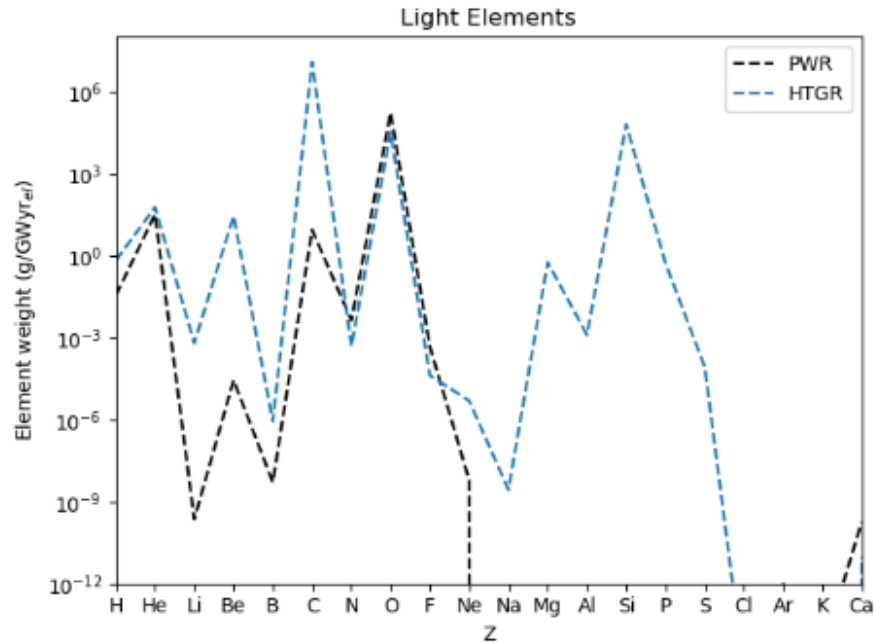
Figure 5-5
Chemical composition for HTGR and PWR: actinides (top), fission products (middle), and light elements (bottom)

Figure 5-6 shows the radionuclide composition for discharged HTGR fuel compared to that for PWR fuel, using the exposure scenarios listed in the methodology. For geological disposal in clay and granite, HTGR fuel produces a higher total activity than PWR for C-14 (5.668×), Cl-36 (negligible in PWR), Ra-226 (1.582×), and Th-230 (1.583×). HTGR fuel produces slightly lower activities for all other radionuclides of concern in Clay and Granite disposal host rocks. For disposal in Salt, only Ra-226 has a slightly increased activity for HTGR fuel compared to PWR fuel (1.582×); all other radionuclides have a 2-3 times lower activity. For the human intrusion scenario, the results are somewhat different, as there are several radionuclides of concern that are unique to this scenario, such as Rn-222, Pu-239, Pu-240, and Am-241. HTGR produces slightly higher activities for the radionuclides from the U decay chain: Pu-238 (1.527×), Th-230 (1.583×), Ra-226 (1.582×), and Rn-222 (1.582×). Compared to PWR fuel, HTGR fuel produces less Cs-137 (0.728×), Th-229 (0.684×), and considerably less Pu-239 (0.186×), Pu-240 (0.284×), and Am-241 (0.552×).
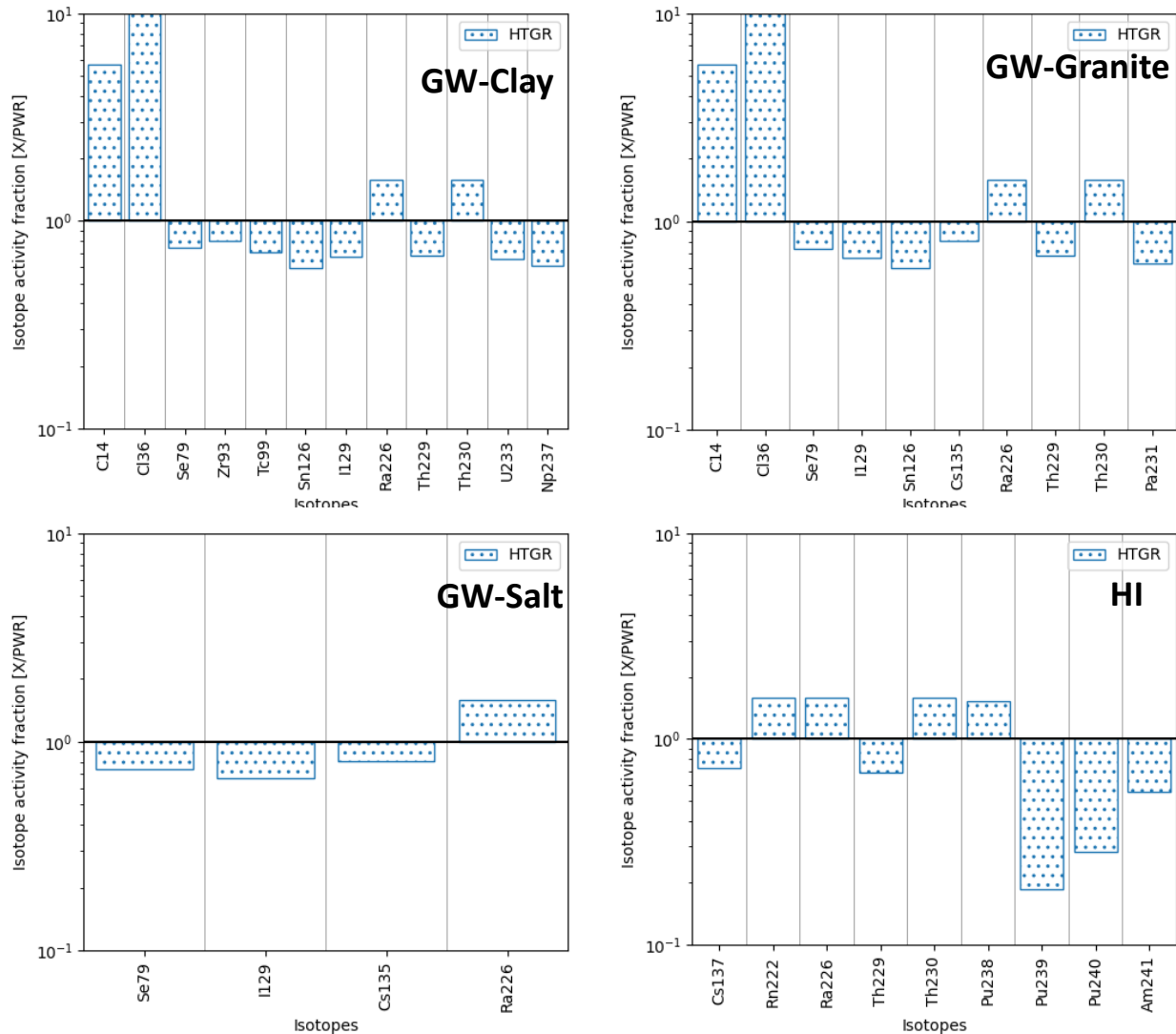
Figure 5-6
Comparison in isotope composition between HTGR and PWR fuel at 300 years for groundwater scenario (GW-Clay, GW-Granite, GW-Salt) and human intrusion scenario (HI). Isotope activities in fraction of PWR reactor (>1 means isotope in HTGR has higher activity than PWR and vice versa for <1).

5.3 Discharged TRISO Fuel Treatment, Disposal, and Reprocessing Option

Conceptually, there are three possible options for disposal of TRISO fuel in a repository [14]:

- Direct disposal of TRISO fuel in pebble and prismatic block form
- Reprocessing of discharged fuel to separate the fuel and fission products
- Disposal with the prior removal of the graphite binder phase

Despite commercial reprocessing of discharged fuel not currently being practiced in the U.S. [15], the study will include this option as part of its discussion and assess its potential at a high level.

A. Direct disposal of TRISO fuel in pebble and prismatic block forms

Currently, the first option, direct disposal of discharged HTGR fuel without the requirement of fuel conditioning or reprocessing, has been proposed in Germany due to their experience storing TRISO on-site [6, 13], but is excluded in the U.S. If allowed, it is the simplest option available and reminiscent of the current approach to LWR fuel in most countries [16]. The vast majority of radionuclides are contained within the kernels, which themselves are encased in the pebbles or surround by prismatic blocks. This potential disposal option may incur the lowest risk of release from handling and processing events. There are some storage technologies that have been developed for TRISO fuel in the past. The U.S. NRC has reviewed the CASTOR cask technology that has been used to store TRISO fuel pebbles from multiple reactors in Germany, which has resulted in their acceptance as a Department of Transportation certified Type B cask [17].

If allowed, direct disposal of the fuel kernels or the entire fuel compact (pebble or prismatic) in a geological repository requires careful consideration of heat load management. Despite discharged fuel for HTGRs producing less decay heat compared to PWR fuel, the fuel elements have a larger volume and thus occupy considerably more waste package volume. Nevertheless, longer waste packages or larger diameter could be an option for direct disposal of HTGR fuel to maximize disposal capacity within thermal design limits [13]. The fuel also could be placed in the same waste package designs proposed for PWR spent fuel, as illustrated in Figure 5-7 [13, 16].
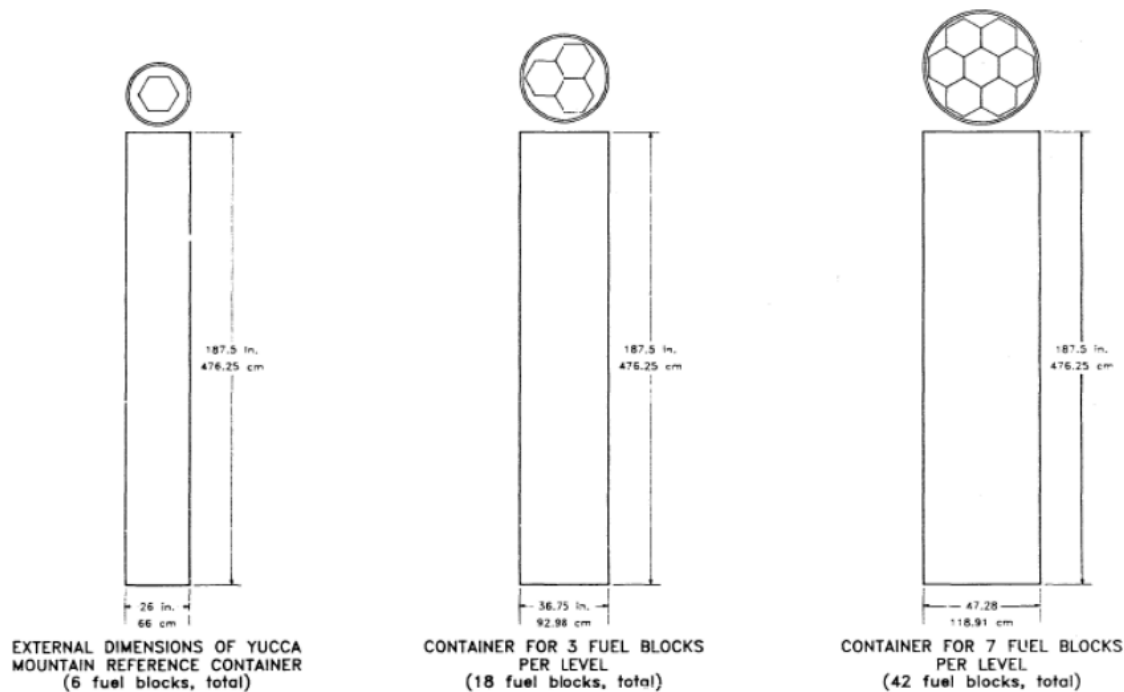
Figure 5-7
Conceptual waste package of similar dimensions and two larger waste package sizes for fuel from the Fort St. Vrain (FSV)'s gas-cooled reactor [13]. Image in Public Domain, courtesy of U.S. NRC.

B. Reprocessing of discharged fuel to separate the fuel and fission products

While this option is more complex, it offers higher utilization of the fuel [16]. Once the graphite matrix has been removed, which is no small feat, the TRISO kernels can be further reprocessed and separated through removing the outer carbon layer, followed by cracking the silicon carbide (SiC) layer, and finally removing the inner carbon layers. This exposes the fuel kernel for chemical separation of fission products and long-lived actinides [16, 20].

HTGR variants produce a greater amount of C-14 than a typical PWR (~10,000 kg/GWe-yr) due to neutron activation within the graphite elements. If the graphite remains in a solid form, its management falls under normal storage methods and regulations for radionuclide release and leaching [16]. However, if the graphite matrix goes through a process of combustion, the carbon dioxide produced must be exhausted or sequestered as there is no practical method to eliminate $(14C)CO_2$ at this time [16, 19]. In addition, compared to a PWR, the HTGR also produces 65 kg/GWe-yr more silicon due to the SiC coating in the fuel kernel particles. As a result, more waste volume will be generated due to these processes.

C. Disposal with the prior removal of the graphite binder phase

If the disposal of an entire TRISO compact, including the graphite binder phase, is unacceptable for a geological repository or if waste volume reduction is desired, the HGTR fuel compacts can be separated into two parts: TRISO kernels and graphite binder [13, 18]. This may be done via two types of separation methods - physical or mechanical separation of the fuel from the graphite binder or a chemical process such as burning, where the graphite is oxidized to carbon dioxide (CO_2) [13]. After separation, fuel processing and graphite processing must be considered separately.

During the mechanical extraction process for the prismatic HTGR fuel element, the fuel rods are pushed out of the graphite block without damaging the TRISO kernels [18]. Contamination from fission products which results from the physical separation may be a concern if there exist any failures of the TRISO kernels which have migrated into the exterior binder phase [13]. If the bulk graphite can be successfully separated from the fuel rods without any contamination, then this may allow the graphite to be disposed as LLW if there is sufficiently low activity of C-14, which would substantially reduce the fuel volume for disposal in the geological repository. This is conditional upon the fission product contamination being low enough to be compatible with near surface disposal. However, if the carbon stream contains significant quantities of long-lived radionuclides, it may fall under the ILW category [14], and require higher classification in some regions.

For the spherical pebble fuel, the separation method could be mechanical (crushing and screening), chemical (combustion), or a hybrid method (fluidized beds or salt dissolution). Note that each method has its own challenges: complex machinery, dust generation, and production of large volumes of hazardous waste streams [16]. The resulting graphite waste will likely contain significant amounts of C-14, with estimated activity of the order of 10^5 Bq/g [16].

Fission products released from TRISO kernels due to wear, fracture, or corrosion can be adsorbed onto dust particles, with the consequences being significant if the dust-gas mixture is vented to the atmosphere [20]. Nevertheless, volume reduction by separation could offer an

advantage for disposal; by reducing the HLW waste volume by about 97% [24] if the bulk graphite can be considered at lower classification than HLW. Various overpack, coating, or encapsulation technologies can be considered to produce an acceptable waste form [18, 22].

5.4 References

1. Subki, H., Advances in small modular reactor technology developments. 2020.

2. Mulder, E., Xe-100: Aspects of Design Important to its Safety Considerations, in NCSU Workshop. 2019, North Carolina State University: U.S.

3. Schenk, W. and H. Nabielek, Testing of irradiated spherical fuel elements at HTR MODUL relevant accident conditions. 1991.

4. Bowers, H. X-energy and the Xe-100. DC ANS April Meeting. 2017.

5. Mulder, E.J. *Overview of X-Energy's 200 MWth Xe-100 Reactor* 2021; Available from: https://www.google.com/url?sa=t&rct=j&q=&esrc=s&source=web&cd=&cad=rja&uact=8&ved=2ahUKEwiH8eWwo9n3AhXgR2wGHZQtDikQFnoECAYQAQ&url=https%3A%2F%2Fwww.nationalacademies.org%2Fdocuments%2Fembed%2Flink%2FLF2255DA3DD1C41C0A42D3BEF0989ACAECE3053A6A9B%2Ffile%2FDCB77DCC95AEF75D0CA6FE9C717CAA34436F7817D15A&usg=AOvVaw0eA1Yy817dwt2lO2u4cBXL.

6. TECDOC, I., *Advances in high temperature gas cooled reactor fuel technology.* Vienna (Austria): International Atomic energy Agency, 2012.

7. IAEA, Status report 101 - Gas Turbine High Temperature Reactor (GTHTR300C). 2011, IAEA.

8. LLC, K.P. *KP-FHR Specifications. .* 2020; Available from: https://kairospower.com/technology/#:~:text=The%20Kairos%20Power%20FHR%20(KP,to%20complement%20renewable%20energy%20sources.

9. IAEA, Chapter 4: The High Temperature Gas Cooled Reactor Test Module Core Physics Benchmarks. 2003, IAEA.

10. Oh, C.H., E.S. Kim, and M. Patterson, *Design option of heat exchanger for the next generation nuclear plant.* Journal of Engineering for Gas Turbines and Power, 2010. **132**(3).

11. Ltd, S. *Lifetime assessments. Assessing rotor lifetime*. 2021; Available from: https://www.sulzer.com/en/shared/services/lifetime-assessments#:~:text=Most%20industrial%20gas%20turbine%20manufacturers,from%20100%2C000%20to%20150%2C000%20EOH.

12. (IAEA), I.A.E.A., Waste from Innovative Types of Reactors and Fuel Cycles. A Preliminary Study, in IAEA Nuclear Energy Series. 2019, IAEA: International Atomic Energy Agency (IAEA).

13. Pan, N.H.X.H.Y.-M., Disposal Options And Potential Challenges To Waste Packages And Waste Forms In Disposal Of Spent (Irradiated) Advanced Reactor Fuel Types. 2019, Center for Nuclear Waste Regulatory Analyses: Texas, USA. p. 22.

14. Shropshire, D.E. and J.S. Herring, Fuel-cycle and nuclear material disposition issues associated with high-temperature gas reactors. 2004.

15. Energy, O.o.N. *5 Fast Facts about Spent Nuclear Fuel*. 2020; Available from: https://www.energy.gov/ne/articles/5-fast-facts-about-spent-nuclear-fuel.

16. McWilliams, A., High temperature gas-cooled reactor (HTGR) graphite pebble fuel: Review of technologies for reprocessing. 2015.

17. Maxted, M. Update on the German Graphite Fuel Project, Presentation to the South Carolina Governor's Nuclear Advisory Council, Jan. 29, 2018.

18. Li, J., Treatment and Disposal of the Radioactive Graphite Waste of High-Temperature Gas-Cooled Reactor Spent Fuel. 2016.

19. Del Cui, G., C. Forsberg, and W. Rickman, TRISO-coated fuel processing to support high-temperature gas-cooled reactors. ORNL, 2002. **27**: p. 4-00.

20. Zakrzewska-Kołtuniewicz, K.K.I.H.-K.L.F.G., Management of Radioactive Waste from HTGR Reactors including Spent TRISO Fuel—State of the Art. Energies, 2022. **15**.

21. Moormann, R., R.S. Kemp, and J. Li, Caution is needed in operating and managing the waste of new pebble-bed nuclear reactors. Joule, 2018. **2**(10): p. 1911-1914.

22. Kitcher, E.D., *A White Paper: Disposition Options for a High-Temperature Gas-Cooled Reactor*. 2020, Idaho National Lab.(INL), Idaho Falls, ID (United States).

6
MOLTEN SALT REACTORS (MSR)

6.1 Introduction

6.1.1 Molten Salt Reactors (MSRs)

A molten salt reactor (MSR) is a fast or thermal (with graphite as moderator) reactor cooled by molten salts (in liquid phase). In the original MSR design variant, the nuclear fuel is dissolved in the primary loop salt. One of its advantages is that the liquid fuel designs don't require the same degree of fuel fabrication steps compared to the more typical solid-fueled reactors, potentially reducing fuel cycle cost. The reactor can be designed to operate with either low or high-energy neutrons. Molten salts are highly corrosive in the presence of impurities (oxygen is a significant contributor) and oxidative fission products increasing their corrosiveness. Thus, most designs require the use of corrosion-resistant materials for the primary loop.

6.1.2 Fluoride-Cooled High-Temperature Reactor (FHR)

The 320 MW_{th}/140 MW_e pebble-bed fluoride salt-cooled high-temperature MSR is the design variant considered in this exploratory study. This FHR is comprised of hundreds of thousands of graphite pebble fuel elements that contain over ten thousand UCO TRISO-coated particles. The particles contain coated uranium oxycarbide (UCO) kernels with an enrichment of 19.75 wt%. The fuel kernel is coated by four layers of carbon- and ceramic-based materials. A major difference between MSR and the pebble-bed HTGR is the utilization of molten fluoride salt (2LiF:BeF$_2$, FLiBe) as the reactor coolant.

The core is contained within a cylindrical geometry with a graphite side-reflector and bottom and top graphite structures. The core internal structures enable reactivity control and shutdown elements [1, 2].
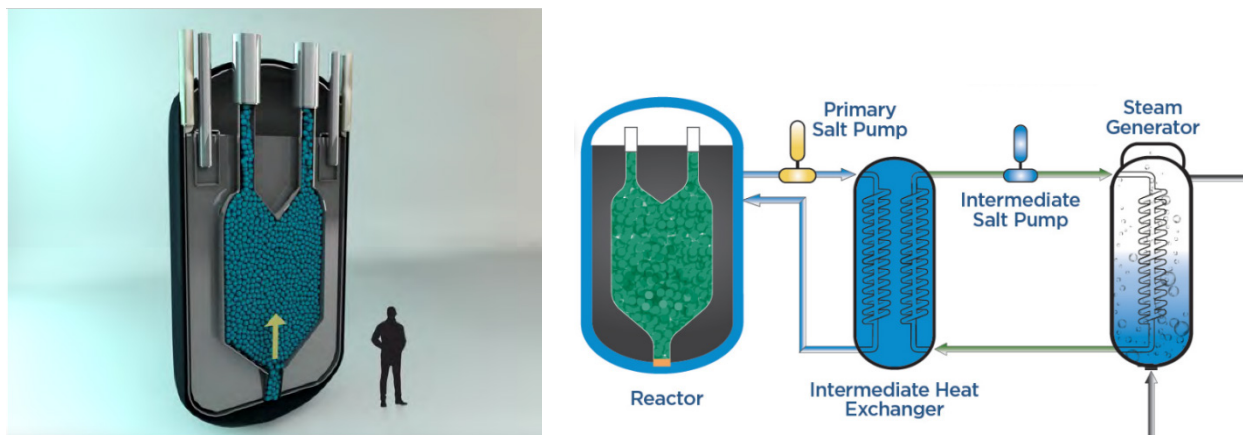
Figure 6-1
Illustration of the Kairos Power FHR (left) [12] and General Layout (right) [19]. Images Copyright© Kairos Power. Used with Permission.

Based on the current conceptual design, the MSR aims to keep the pebble fuel passing through the reactor while measuring its burnup to determine the available amount of useful fuel. The process of pebble recirculation will be repeated until the fuel is ready to be discharge for storage and disposal [1].

The primary mechanism for reactivity control and non-accident events is provided by inserting the control rods into the side reflector channels that surrounds the core. However, if there is an unexpected accident, the reactor can shut down by inserting the shutdown elements directly into the pebble-bed core which are driven by gravity once they are released by the reactor protection system [1]. Both the control and shutdown elements consist of a composite structure of neutron absorber material made of natural B_4C in an inert gas with SS316H cladding [1].

6.2 Waste Identification and Characterization

Several assumptions have been made for the calculation of FHR spent volume and for the activation calculations.

- The replacement period for the RPV will be assumed to be 20 years [1].
- MSR shares similarities in reactivity control for reactor maneuvering and non-accident events to the Pebble-Bed Modular Reactor (PBMR) [1, 3]. Thus, this report will assume MSR to have the same number of control rods and diameter as the PBMR.
- The Intermediate Heat Exchanger design and specifications will be assumed to be the same as the PB-FHR Heat Exchanger due to its similar flow schematic [4]. However, since the PB-FHR Heat Exchanger has a RPV with longer length (11.02 m), the dimensions for the MSR IHX will be rescaled relative to the PB-FHR Heat Exchanger using a 1:1.8 ratio.
- The specifications for the MSR IHX tubes are assumed to have the same diameter and wall thickness as for the PB-FHR [4].
- The FLiBe salt replacement is assumed to be 20 years [1].
- The wall thickness for HTX is assumed to be the same as the RPV's wall thickness.
- The replacement period for the heat exchanger vessel is assumed to be 20 years [5].
- The burn-up is assumed to be 180 GWd/tHM.

The largest waste stream for MSR is the FLiBe salt coolant with a produced energy-equivalent volume of 32.5 m^3/GW_e-yr. These waste streams are shown in Figure 6-2. This volume is sensitive to the replacement period, here assumed to be 20 years. Due to the high production cost of FLiBe [6], the salt coolant may be preferred to be used through the plant lifetime of 60–80 years, with contamination cleanup as the preferred option over salt replacement [1, 6]. All the non-fuel waste streams are considered low-level and intermediate-level waste consistent with the IAEA classification of these wastes [7]. As with previous systems involving graphite, the classification of these wastes may vary based on region. Further challenge is present due to the potential for intrusion of salt into the porous graphite.
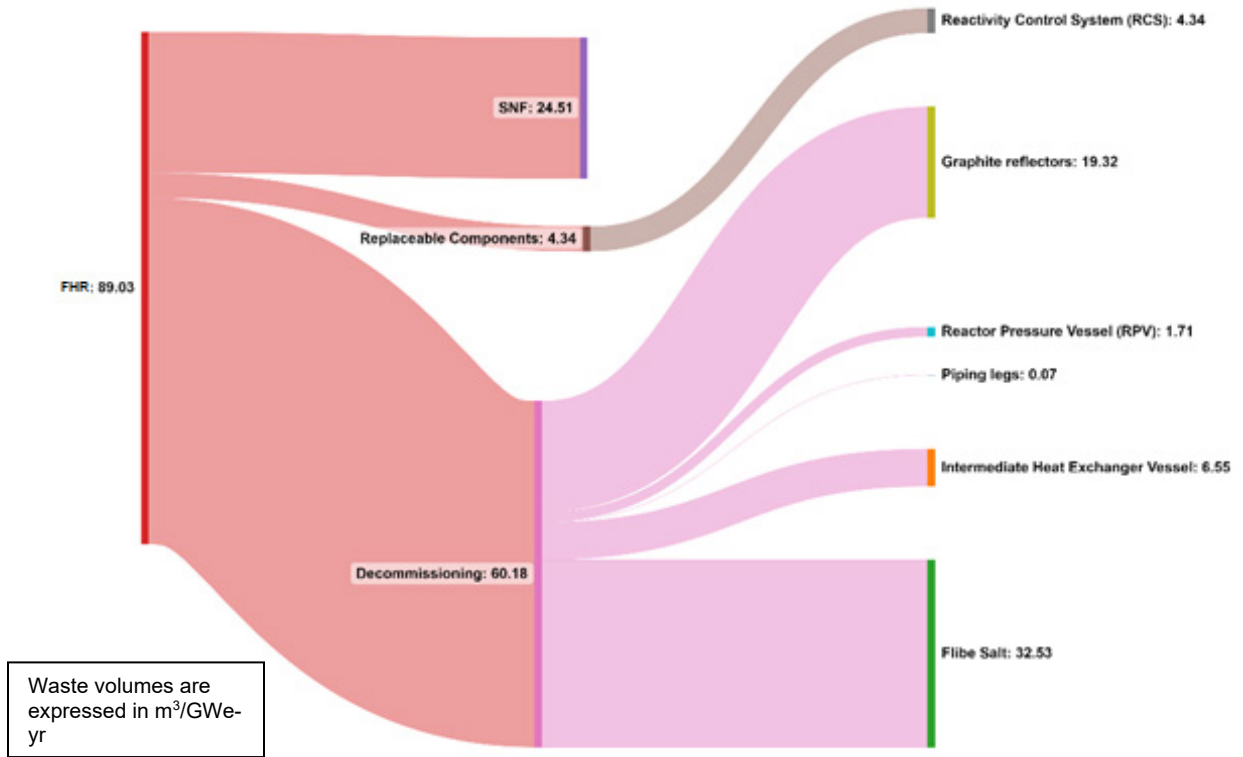
Figure 6-2
Sankey Diagram showing annual volumes of generated waste (m³/GWe-yr) and contributions from various reactor components of the FHR.

Figure 6-3 shows a comparison of calculated activity and decay heat between FHR and PWR. FHR used fuel exhibits higher activity and heat load than PWR until year one, where a crossover occurs. After the initial 100 years of similar decay pattern, the difference in activity and heat becomes progressively larger and at 10,000 years, the difference is about half on order of magnitude. This different may impact planning for initial fuel loading campaigns and may dictate different timing in a cooling system compared to the TRISO fuel discharged from an HTGR. This also may not cause an impact in the eventual disposition pathway, as TRISO with salt contained in the exterior pores may need to undergo a conditioning step to remove the salt from the fuel compact before further cooling, storage, recycling, or disposal.
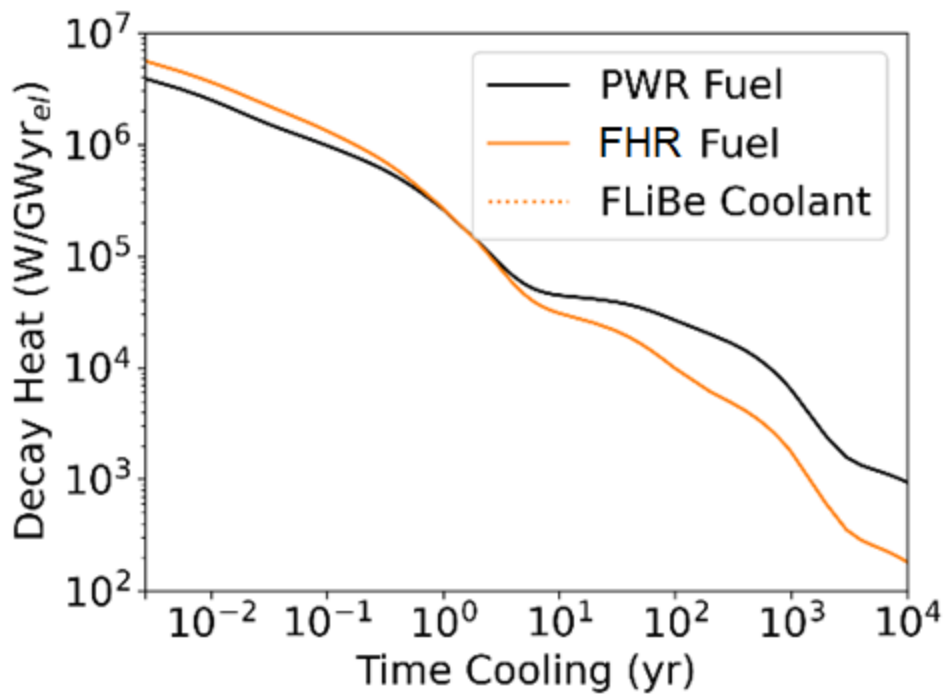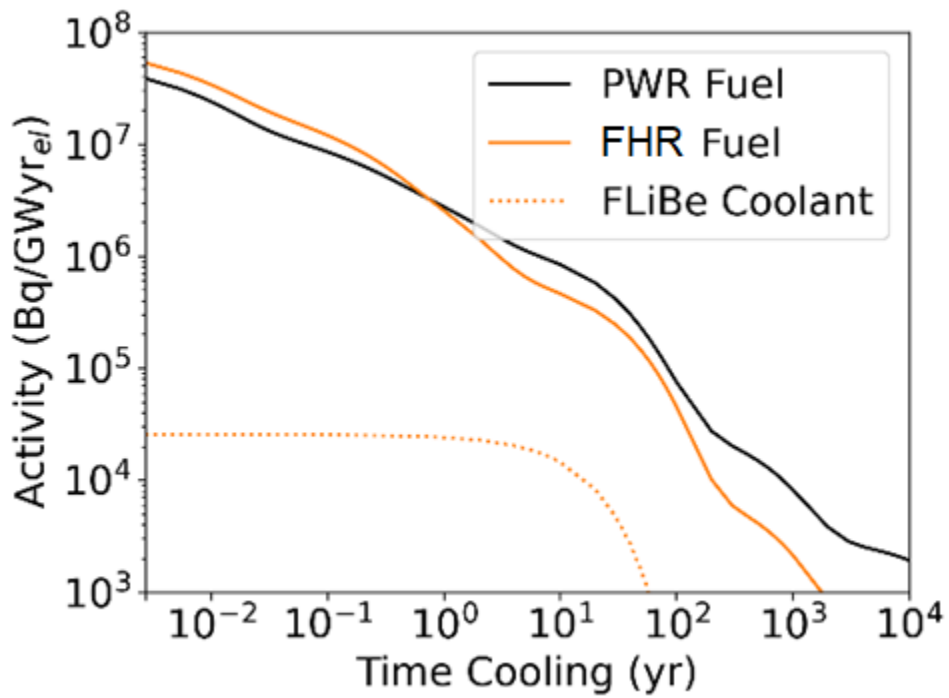
Figure 6-3
Calculated activity (top) and decay heat (bottom) for FHR fuel and FLiBe coolant

Coolant activity and depletion have been considered to project applicability to disposal. The study also considered the impact of initial salt isotopic composition on the need to replenish the coolant due to loss of the chemicals through transmutation (which only considers the changes due to nuclear effects and no other factors such as chemical reactions, impurities, or corrosion). Figure 6-3 shows the coolant has less activity and decay heat than the fuel itself, and it plateaus after removal from the reactor. After approximately 30 years, there is a significant drop in coolant activity. The decay heat of the FLiBe salt based on these assumptions appears negligible.

If natural isotopic concentrations of the constituent elements are used for FLiBe, after 200 days in the reactor, over 70% of Li-6 will be depleted in the reactor. Li-6 is a strong neutron absorber, and transmutes into Li-7, so the use of enriched Li-7 is often considered, with 50 ppm or less Li-6. This further avoids the challenges from the decay chain of Li-6. In the case that a Li-6 atom undergoes neutron absorption, an alpha particle and tritium are produced. Avoiding tritium production in an FHR system is likely ideal, which indicates that economic lithium enrichment technology may be an enabler for FHR deployment.

Aside from Li-6, depletion of FLiBe due to activation and transmutation is limited: after 200 days in the reactor Li-7, Be-9 and F-9 will deplete by 70 ppm, 30 ppm, and 30 ppm, respectively, as shown in Figure 6-4. This is proportionally more for Li-7, if the starting composition was enriched in Li-7. Importantly, all the daughter isotopes decay into hydrogen and helium, which are easily separated and extracted. A much greater loss of coolant volume is expected to arise from any potential chemical reactions (including corrosion induced by any oxygen impurities) or contamination in the primary salt loop.
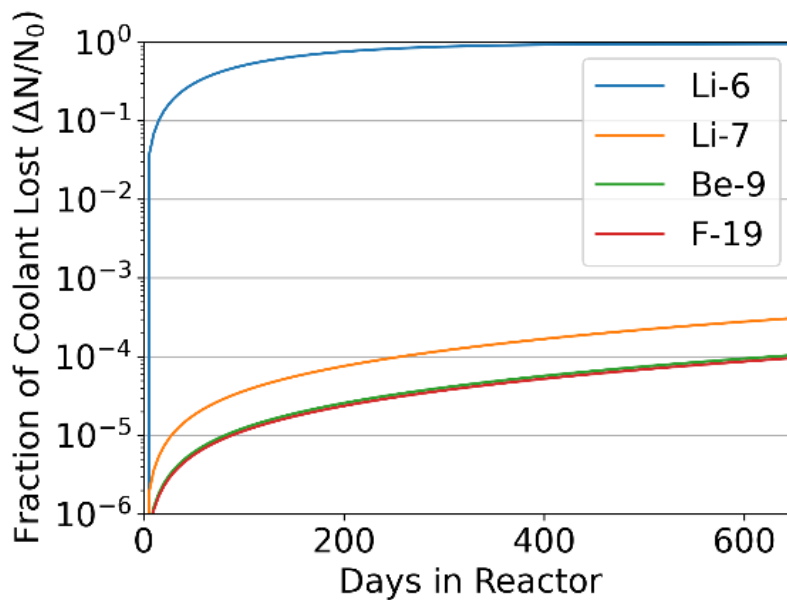
Figure 6-4
Volume fraction (%) of FLiBe isotopic depletion, assuming natural abundance starting composition

Elemental composition differences in discharged fuels from the FHR and PWR variants are prominent for the light elements, with FHR fuel containing less Li, Be, and N. The largest difference exists for elements Na, Mg, Al, Si, P, and S, where PWR has negligible mass compared to concentration in FHR fuel, as seen in Figure 6-5.

Actinides
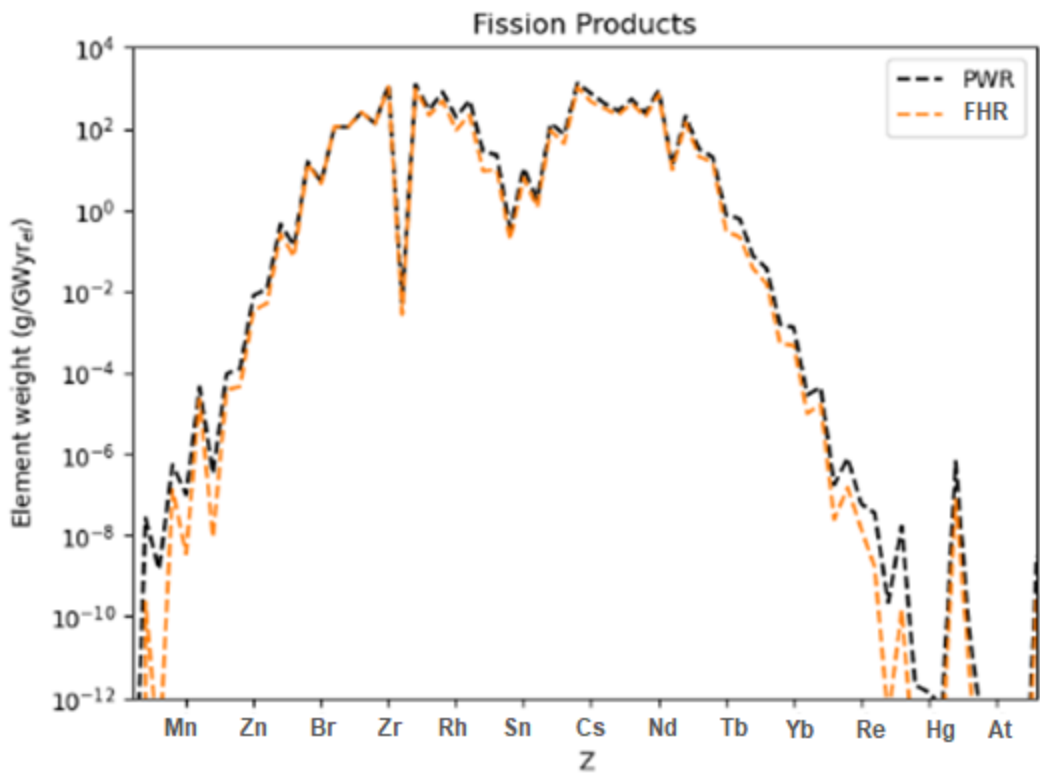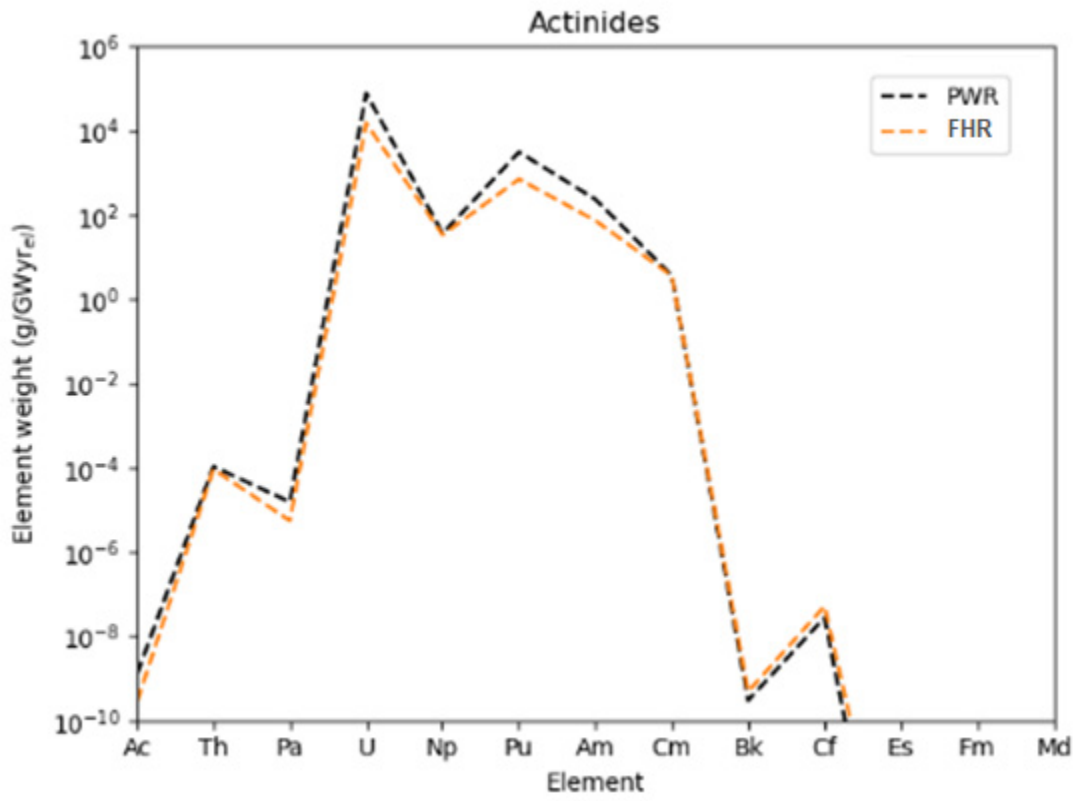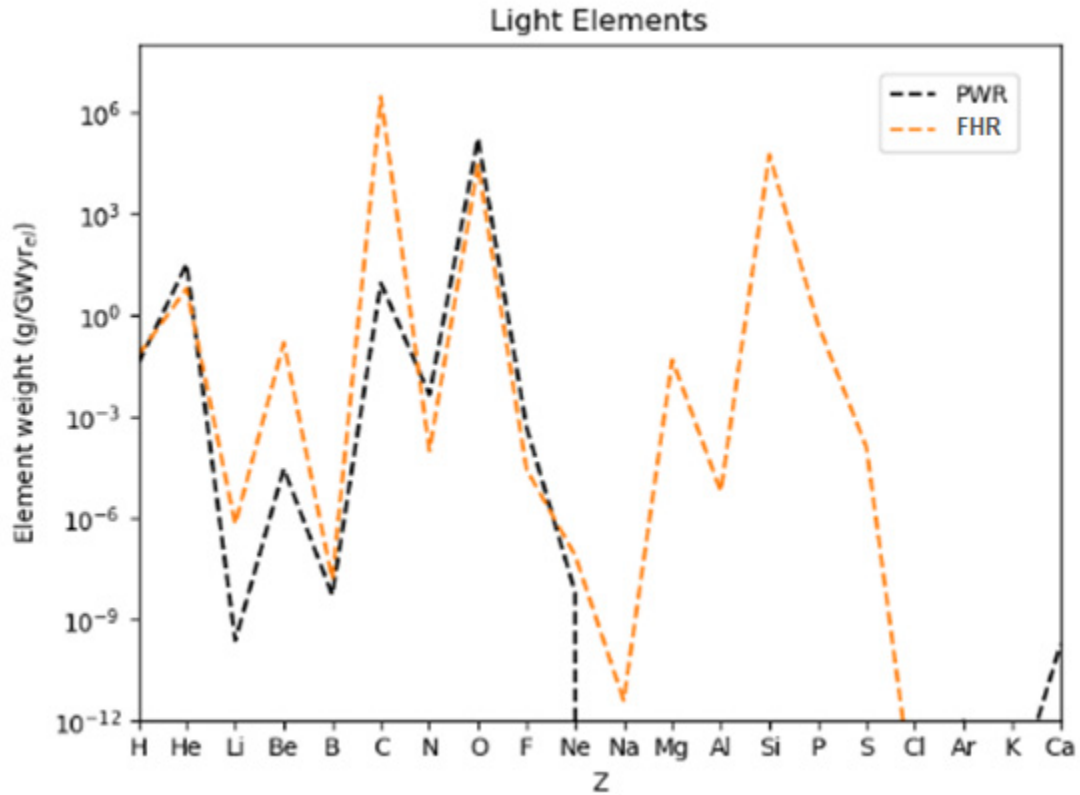
Fission Products

Figure 6-5
Chemical composition for FHR and PWR fuels: actinides (top), fission products (middle) and light elements (bottom)

Comparison of total radionuclide activity between discharged FHR and PWR fuels indicates the FHR fuel produces less activity (between 0.88× and 0.258×) for the majority of the radionuclides except for C-14 (1.925 ×) and Cl-36 (negligible in PWR), which may impact disposal in clay and granite. For disposal in salt, FHR fuel produces less activity for all radionuclides of concern (between 0.88× and 0.258×). The radionuclides of concern for the human intrusion scenario consistently have lower activities for FHR fuel compared to PWR fuel (between 0.771× and 0.118×), which can be seen in Figure 6-6.
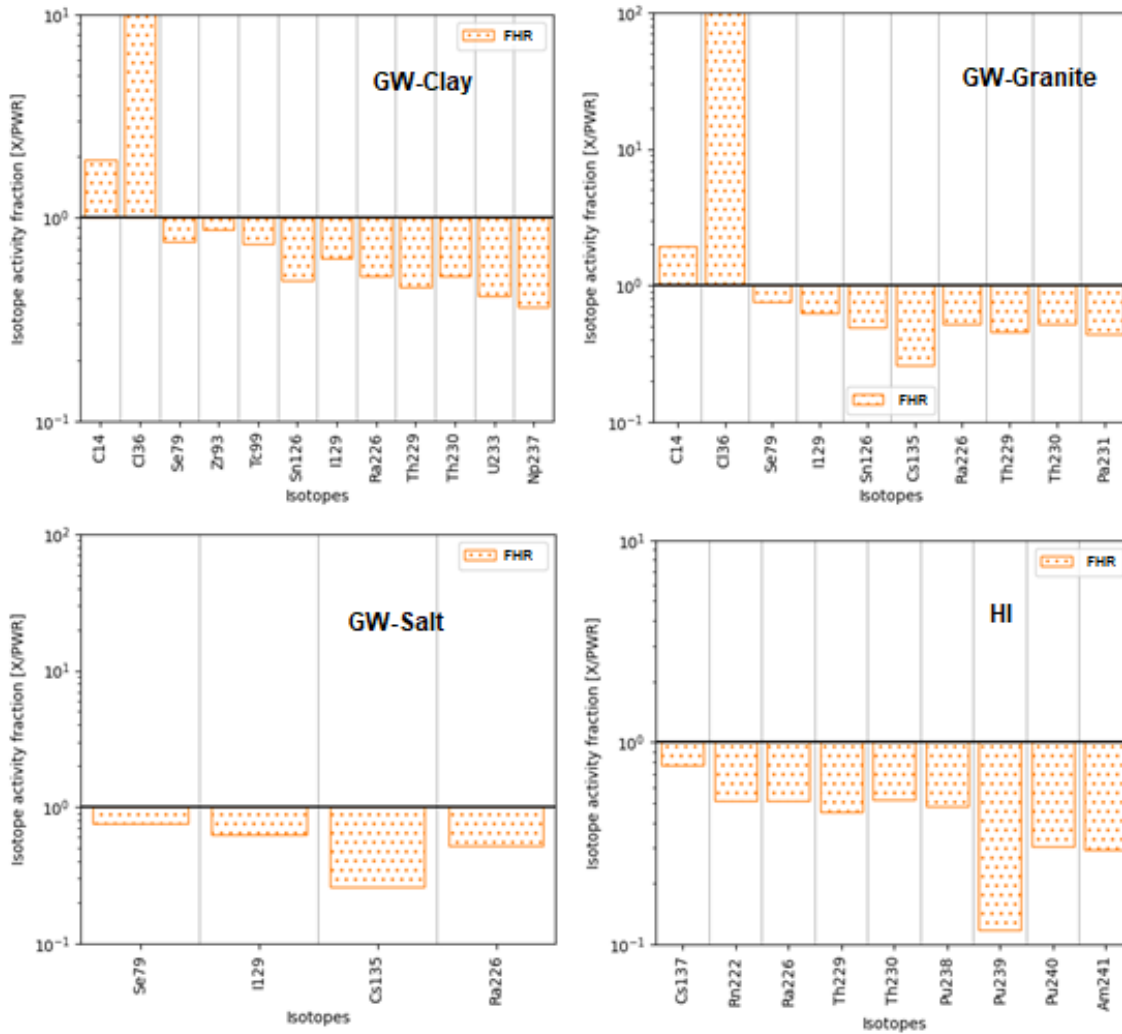
Figure 6-6
Comparison in isotope composition between FHR and PWR at 300 years for groundwater scenario (GW-Clay, GW-Granite, GW-Salt) and human intrusion scenario (HI). Isotope activities in fraction of PWR reactor (>1 means isotope in FHR has higher activity than PWR and vice versa for <1).

6.3 Treatment and Disposal of Spent TRISO Pebbles

Similar to the pebble-bed HTGR, the FHR graphite-moderated core design uses 220,000 graphite pebbles but with 16,000 UCO TRISO coated kernels per pebble as its fuel elements [4]. Due to the presence of FLiBe salt in the FHR, the method to treat these graphite waste streams will be different from the pebble-bed HTGR. A particular feature concerns the salt, which must be drained before the fuel temperature becomes lower than the salt melting point, at which time salt radiolysis can begin to occur [8]. The following options exist regarding the treatment of spent fuel and graphite for disposal:

- **No treatment** – The discharged fuel and graphite disposal canisters could be filled with an inert gas (helium or argon), as the small quantities of fluorine, tritium, and other salt components are expected to have an insignificant long-term safety impact. Thus, no

treatment is required. This option includes addition of getters in disposal canisters to chemically react with any gaseous salt vapors (fluorine, hydrogen fluoride, or hydrogen, which may be released as the salt cools).

- **Vacuum drying** – Once the liquid salt has been drained, the residual salt and any substantial amount of tritium can be removed by increasing the temperature of the discharged fuel while creating a vacuum. This process is similar to that used for the LWR fuel assemblies prior to dry storage. The difference between vacuum drying for FHR fuel and LWR fuel is that the drying temperatures for the former can reach 1000 °C or more. For the LWR fuel, there are tight constraints on peak drying temperatures to avoid the formation of radical zirconium hydrides in the clad that cause weakness. For irradiated graphite, heating would be required to increase the temperatures of graphite to discharge any retained salt or other waste elements.

- **Gas Drying** – An inert gas like helium could be used to remove volatile salts from the fuel and graphite. Despite the effectiveness of the method for LWF fuel, it is expected to be less effective then vacuum drying for FHR fuel, as the gas flow may cause wear of the fuel through vibration and fretting.

- **Washing** – To dissolve any residual salts that remains in the system, the discharged fuel and graphite waste can be washed with water-based (or alternative) liquids. If a water solution is chosen, subsequent fuel drying will be essential, and the wash water may contain tritium.

6.4 Treatment and Disposal of FLiBe Salt

The management of salt waste streams can be divided into whole-salt streams (unseparated salts) and separated salt streams where specific species such as fission products, F and Cl, are discharged. It may be preferred to separate the salts into different components due to the numerous benefits it offers for recycling and waste form fabrication. From a recycling perspective, it would be favorable to recycle Li-7-enriched salt back into future FHRs due to the salt's high cost [9]. Moreover, the purification process to prepare the salt for use requires hazardous solvents, so minimization of that activity is of interest for safety [9].

Direct liquid fluoride salt disposal in a geological repository could be possible but comes with the risk that the water-soluble salts could penetrate the groundwater if the repository containment is breached. A more likely alternative is to immobilize or stabilize the salt in a waste form which is chemically resistant towards dissolution and subsequent transport from a repository to groundwater [9]. To immobilize the whole fluoride salt in a single waste form that satisfies waste acceptance criteria for a high-level waste repository would be challenging. For one, those waste acceptance criteria may not exist in every region due to minimal data or experience with disposal of salt wastes. For salt-based high-level waste, typical waste form options includes glasses, ceramics, glass ceramics, glass-bonded ceramics (alternatively called glass-composite waste forms), and ceramic-metal composites (alternatively called cermets), which can be seen in Figure 6-7 [9].
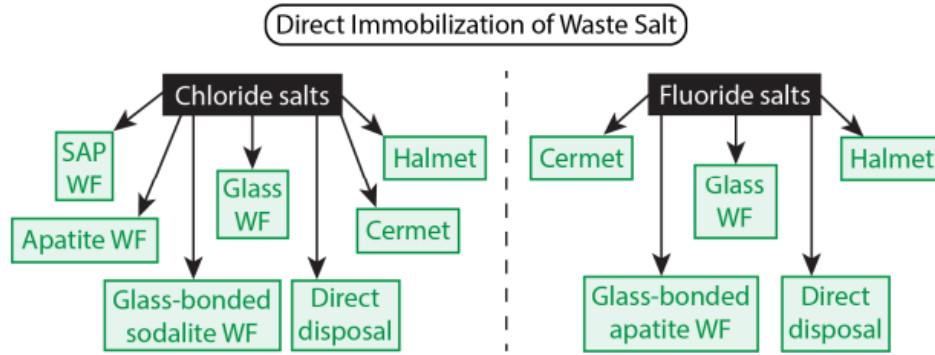
Figure 6-7
Summary of unseparated salt waste streams along with disposal pathways [10]. Image in Public Domain courtesy of Oak Ridge National Laboratory

6.4.1 Off-Gas Waste Processing, Capture, and Immobilization Method

The type of waste streams exiting the reactor will be dependent on several design options, including [11]:

- Type of reactor (e.g., salt-cooled or salt fueled)
- Type of salt (e.g., chloride salts could develop Cl_2 and volatile species such as UCl_4, fluoride salts could develop F_2 and volatile species such as UF_6)
- Neutron spectrum, fissile and fertile material types, and fuel processing strategy, which result in different fission product distributions
- Real-time separations technologies, such as pathways for removing fission products, could introduce other gases and volatiles

Most of the current studies related to the off-gas treatment are for salt-fueled reactors [10, 11]. Due to the lack of data availability for FHR systems, this study assumes that the main types of off-gas streams will be consistent across the different reactor concepts (i.e., particulates, aerosols, reactive gases, tritium, noble gases, residual halides, nitrogen, and oxygen).

During reactor operation, particulates and aerosols could be produced in a variety of ways. In dissolved fuel MSRs, there has been evidence of corrosion from the graphite moderator. Carbon floated and could become entrained in a purge gas passing over the salt due to its low density compared to the salt [6, 10-12]. A molten hydroxide scrub with a melting temperature of 318 °C or more for NaOH, could be used to capture these gases. A submerged-bed aqueous hydroxide scrubber may be used as an alternative [10, 11].

Tritium can be produced directly in the primary coolant by neutron reactions with Li and Be, primarily from the reaction of Li-6, and it can exist in salts as tritium fluoride (TF), as a dissolved ion (+T), or as tritium gas (HT or T_2) [12]. While being transported in the primary loop, the produced tritium can either be confined by the porous graphite materials in the primary loop, escape through the primary coolant surface into the cover gas, infiltrate through the reactor vessel or piping, or infiltrate through the heat exchanger tubing. The technologies for removing tritium include helium-hydrogen sparging, ultrasonic degassing, spray-droplet based

disengagement, and cathodic stripping, all of which heavily rely on the mass transfer of tritium in the salt [12]. However, for the most part, the effectiveness of these methods in a FHR has not been proven to a high level of confidence due to the complexity in quantifying tritium release [10].

The residual halides originate from salt impurities such as Br_2, fission products (e.g., I_2), and salt breakdown products (e.g., Cl_2 or F_2). For halide capture, various metal-impregnated/functionalized sorbents have been studied. The zeolite family of minerals is one of the most commonly researched groups of solid sorbents for halides. To remove these species from a gas stream, another alternative is to utilize an aqueous-based caustic scrub or a nonaqueous molten hydroxide scrub [10].

The primary coolant boundary will initially have a thin oxide coating across all of its metallic surfaces. The primary coolant cover gas will contain small amounts of moisture and oxygen. Using a sacrificial carbon anode, oxides can be directly electrochemically reduced to form carbon dioxide in order to eliminate the oxygen impurity from the primary molten salt [12].

Most of the noble gases eventually emerge into the gas plenum above the salt pool due to the low solubility of Kr and Xe in the primary salt. Some of the gas can also permeate into the structural materials, such as graphite [12]. The primary removal methods for these elements are the dissolution process in liquid nitrogen which then can be separated in purification columns [10]. For FHRs, most of the gases will eventually emerge into the gas plenum above the salt pool [12] (see Table 21 for the summary of off-gas waste processing, capture, and immobilization methods).

6.5 References

1. Subki, H., Advances in small modular reactor technology developments. 2020.

2. K.P. LLC, *Generic FHR Core Model*. 2022; Available from: https://kairospower.com/generic-fhr-core-model/.

3. IAEA, Status report 70 - Pebble Bed Modular Reactor (PBMR). 2011, IAEA.

4. Engineering, D.o.N., Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant. 2014, University of California, Berkeley: U.S.

5. Oh, C.H., E.S. Kim, and M. Patterson, *Design option of heat exchanger for the next generation nuclear plant*. Journal of Engineering for Gas Turbines and Power, 2010. **132**(3).

6. Seifried, J.E., et al., A general approach for determination of acceptable FLiBe impurity concentrations in Fluoride-Salt Cooled High Temperature Reactors (FHRs). Nuclear Engineering and Design, 2019. **343**: p. 85-95.

7. (IAEA), I.A.E.A., Waste from Innovative Types of Reactors and Fuel Cycles. A Preliminary Study, in IAEA Nuclear Energy Series. 2019, IAEA: International Atomic Energy Agency (IAEA).

8. Forsberg, C. and P.F. Peterson, Spent nuclear fuel and graphite management for salt-cooled reactors: Storage, safeguards, and repository disposal. Nuclear Technology, 2015. **191**(2): p. 113-121.

9. Riley, B.J., et al., *Molten salt reactor waste and effluent management strategies: A review.* Nuclear Engineering and Design, 2019. **345**: p. 94-109.

10. Riley, B.J., et al., *Identification of potential waste processing and waste form options for molten salt reactors*. 2018, Oak Ridge National Lab.(ORNL), Oak Ridge, TN (United States).

11. Andrews, H.B., et al., *Review of molten salt reactor off-gas management considerations.* Nuclear Engineering and Design, 2021. **385**: p. 111529.

12. Zhang, J., Impurities in primary coolant salt of FHRs: Chemistry, impact, and removal methods. Energy Technology, 2019. **7**(10): p. 1900016.
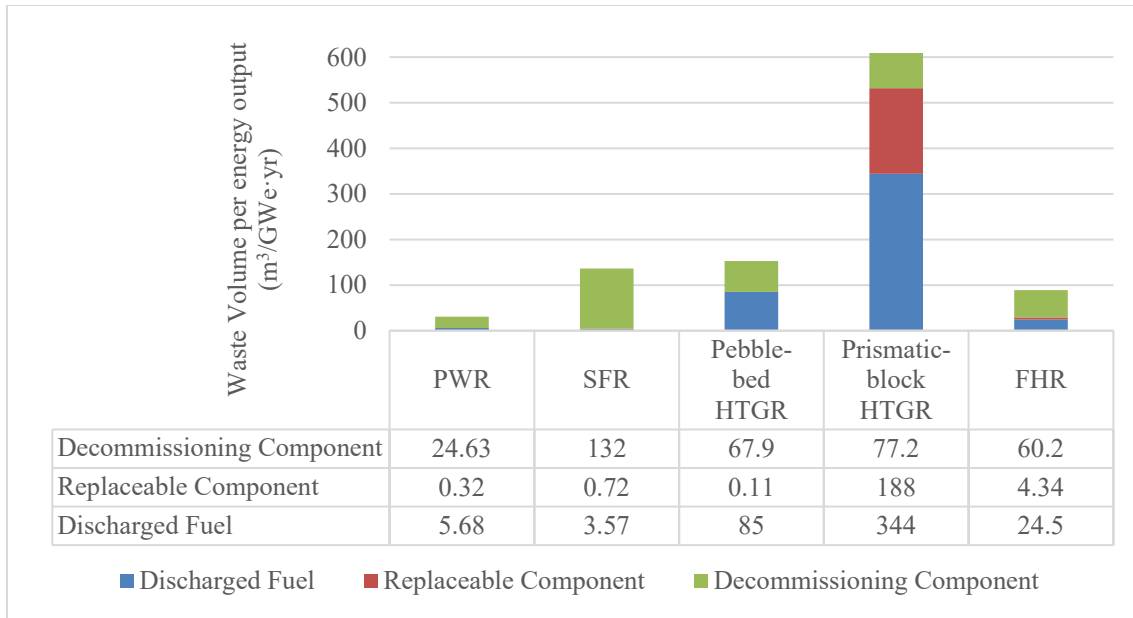
7
IMPACTS OF ADVANCED REACTORS ON THE BACK-END OF THE FUEL CYCLE

With differences in initial enrichment, burn-up, fuel composition, and neutron spectra, the discharged fuel attributes of the AR concept variants selected are understandably distinct from the industry standard PWR fuel. These differences can impact storage, transportation, and disposal strategies, may require differing technologies, and could provide benefits or detriments depending on a number of variables.

An important note for the comparison of these diverse advanced reactor technology concepts is that they represent only a subset of the multiple configurations that can be developed. The impact of this on the exploratory study is primarily driven by the selection of the SFR variant. The SFR concept chosen produces the least volume of spent nuclear fuel compared to the rest of the reactors due to its design as a unique breed-and-burn reactor which allows for high levels of burnup. This is not necessarily representative of every SFR and also provides a challenge for direct comparison with thermal reactor concepts. Actinides are produced from the driver fuel assemblies and deeply burn in-situ relative to the fuel, which enables higher fuel utilization. The SFR also produces the highest volume of decommissioning waste, mainly contributed by the liquid sodium coolant. Other than the SFR variant selected, the other ARs considered in this report produce larger volumes of discharged nuclear fuel per unit of energy produced compared to a typical PWR, due to the use of TRISO fuel.

Although both HTGR and the FHR variants considered have higher burn-up rates than the PWR (i.e., >120 GWd/tHM and 180 GWd/tHM respectively, compared to 49 GWd/tHM), the physical characteristics of the TRISO fuel used for these advanced reactors such as presence of carbon- and ceramic-based materials in the graphite matrix pebble fuel contribute to the increase in spent fuel volume and classification. Whether as a contaminant on graphite or a constituent of nitrides, N-14 may be present in these fuel forms. This can result in the formation of C-14, which can increase classification of LLW or ILW graphite streams to long-lived ILW, which requires disposal as HLW in some regions. While both the pebble-bed HTGR and FHR use TRISO fuel in pebble form, the FHR variant considered generates less spent fuel due to its fuel form composition and enrichment.

For the prismatic-block HTGR design variant considered, the apparent increase in discharged fuel volume corresponds to the graphite fuel blocks that contain the fuel compacts. The nature of the refueling method, which assumes replacement of half the core, reflectors, and control rods every two years, also yields the highest volume of replaceable component waste [18]. It should be noted that this is not a strict requirement of a prismatic HTGR design. As can be seen from the Advanced Gas Reactors (AGRs) in the UK, non-fuel graphite elements within a reactor core environment can be used for longer than this replacement period, and have been in more aggressive environments than helium. Except for the control rods that are stainless steel clad B_4C, the core and reflector components are predominately composed of graphite materials. which require additional processes in treating and disposing the carbonaceous waste [1].

	PWR	SFR	Pebble-bed HTGR	Prismatic-block HTGR	FHR
Decommissioning Component	24.63	132	67.9	77.2	60.2
Replaceable Component	0.32	0.72	0.11	188	4.34
Discharged Fuel	5.68	3.57	85	344	24.5

■ Discharged Fuel ■ Replaceable Component ■ Decommissioning Component

Figure 7-1
Waste volume comparisons (discharged fuel, replaceable components, and decommissioning components) between AR concept variants in comparison with PWR

As can be seen in Figure 7-1, these graphite reflectors, structural elements, and the prismatic block which contains the fuel compact constitute the majority of the waste produced in the prismatic HTGR variant considered. This illustrates the sensitivity of waste stream volume to operational and design choices. In the other two TRISO fueled reactor concepts, which use similar graphite reflectors and structural elements, the quantity of waste form that is replaceable components is considerably less. These two design variants use longer-lived graphite reflectors, which reduces equivalent waste volume by around an order of magnitude. In the event this specific waste must be disposed of as HLW, this can result in significant cost savings compared to a more frequent replacement scheme.

All reactors show a similar pattern in the modeled decay heat profile represented in Figure 7-2. The FHR producing about twice as much decay heat during the first year compared to the HTGRs, and SFR, and at a year is equivalent to the PWR. After 10 years (around when LWR fuel is removed from a spent fuel pool), the decay heat for all AR concept fuels is lower than the PWR fuel, by up to 50%. In the interim storage timespan up to 100 years, the decrease in decay heat accelerates for each of the AR fuels. The TRISO fuels from the HTGRs produces around two thirds the heat of the PWR, the FHR fuel about one third, and the SFR fuel produces only around one fifth of the heat of the PWR fuel. Beyond 100 years, out to 10,000 years is often considered the disposal period. These trends change slightly within this time range, with the SFR fuel beginning to plateau in decay heat reduction after a few hundred years. By 10,000 years, the SFR fuel produces about two thirds the heat of PWR fuel, while the FHR fuel produces 15% as much heat, and the HTGR fuels produce about 10% as much heat.

These decay heat differences can drive important considerations for storage (in the 10–100 year range) and disposal (100 to 10,000 year range). With decay heat production 33% to 80% less than PWR fuel, these AR fuels may have very different cooling and loading patterns for interim storage. This indicates that existing technologies, if capable of handing the differences in fuel

form and geometry, are likely capable of handling the heat loads anticipated within their intended lifetime. For disposal, reduced decay heat impacts disposal waste form design, and could impact repository design.

The activity profile follows a similar behavior as the heat load, but with some key differences. From approximately 100 years onwards, total activity for the FHR, HTGRs, and SFR begins to deviate further with a sharp decline in activity for SFR fuel while the decrease for PWR fuel is more moderate. This behavior mirrors the deviation the SFR makes in decay heat, which is sensible. The deeply burning configuration of the SFR does not result in the same fuel isotopics upon discharge as for the three concepts which use TRISO fuel in a thermal neutron spectrum. As for the FHR and HTGRs, the calculated total activity diverges, close to one order of magnitude by 10,000 years.
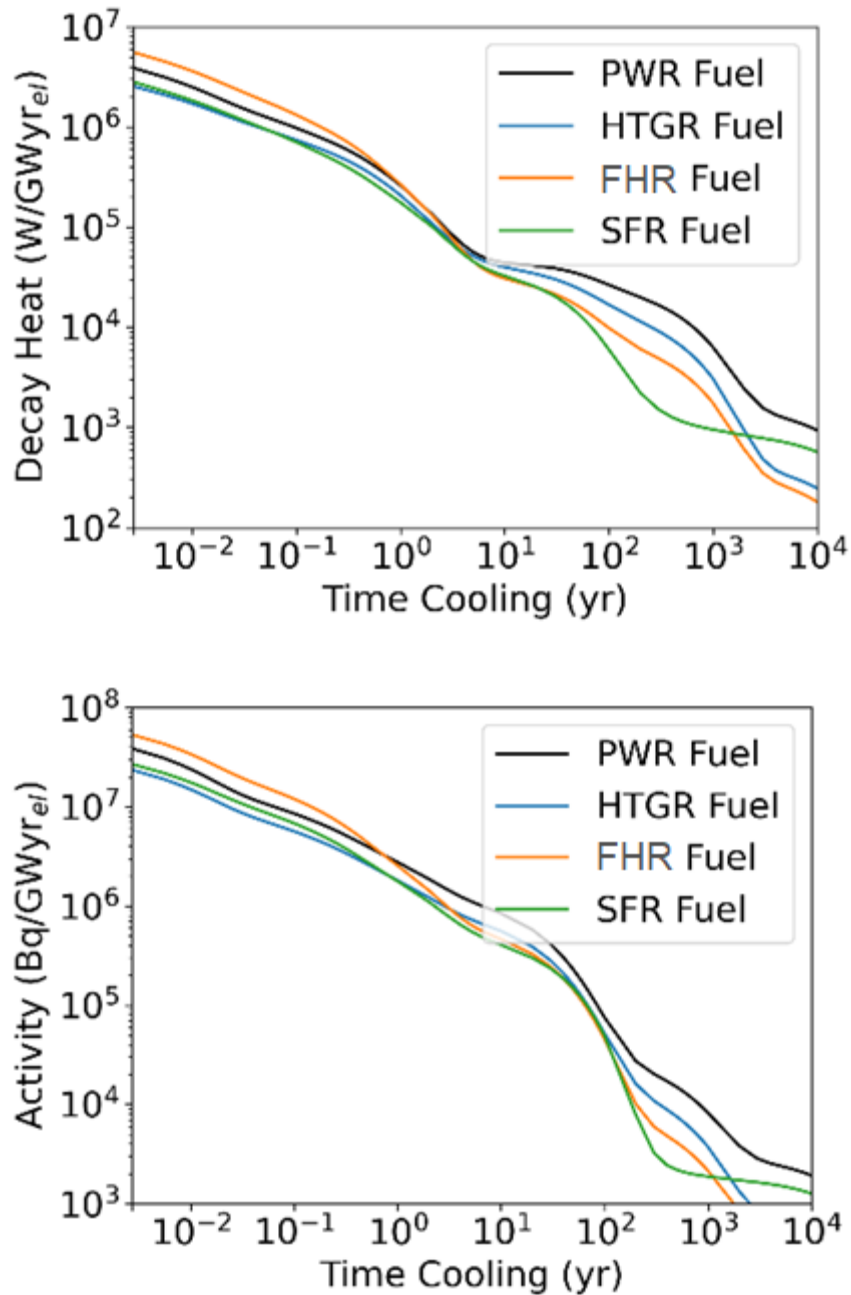
Figure 7-2
Calculated decay heat (top) and activity profile (bottom) between the advanced reactors

Per the methods noted in Section 2, isotopes of note were chosen from repository standards for clay, granite, salt, and a human intrusion scenario. The modeled isotopic composition was downselected to these isotopes of note for display in Figure 7-3 which contains the concentration of each isotope for each AR concept variant relative to concentration within PWR fuel, 300 years after discharge. The FHR produces the least activity (between 0.88× and 0.258×) for isotopes noted in the three groundwater pathways, with exceptions of C-14 (1.925 ×) and Cl-36 (negligible in PWR), which are relevant for disposal in clay and granite bedrocks. The HTGR

fuels contain higher total activity than PWR fuels for C-14 (5.668×), Cl-36 (negligible in PWR), Th-230 (1.583×) and Ra-226 (1.582×), which are relevant to groundwater pathway scenarios. For disposal in salt, Ra-226 is the only relevant isotope for which HTGR fuels yields higher concentrations than the PWR fuel. Conversely, SFR fuel produces much greater activities than PWR fuel for all radio-isotopes of concerns for disposal, with the only exception being C-14 which is not produced in the SFR. As noted previously, the distinct behavior of the SFR fuel is representative of the differences in reactor technology configuration. The fast spectrum environment this fuel is fissioned in creates different fissions and decay chains, which leads to the stark difference in radionuclide activity.

Concerning the radionuclides of concern for the human intrusion scenario, the FHR fuel yields consistently lower activity than PWR fuel for all isotopes. On the other hand, HTGR produces slightly higher activities for the radionuclides from the U decay series: Pu-238 (1.527×), Th-230 (1.583×), Ra-226 (1.582×), and Rn-222 (1.582×) and less activities in Cs-137 (0.728×), Th-229 (0.684×), and considerably less Pu-239 (0.186×), Pu-240 (0.284×), and Am-241 (0.552×). Due to the power density differences between the TRISO fuel designs and the PWR, this difference in activity is sensible. SFR fuel yields significantly higher activities for all radionuclides of concern compared to PWR fuel, with the exception of Am-241, which is fissioned by the fast neutrons.
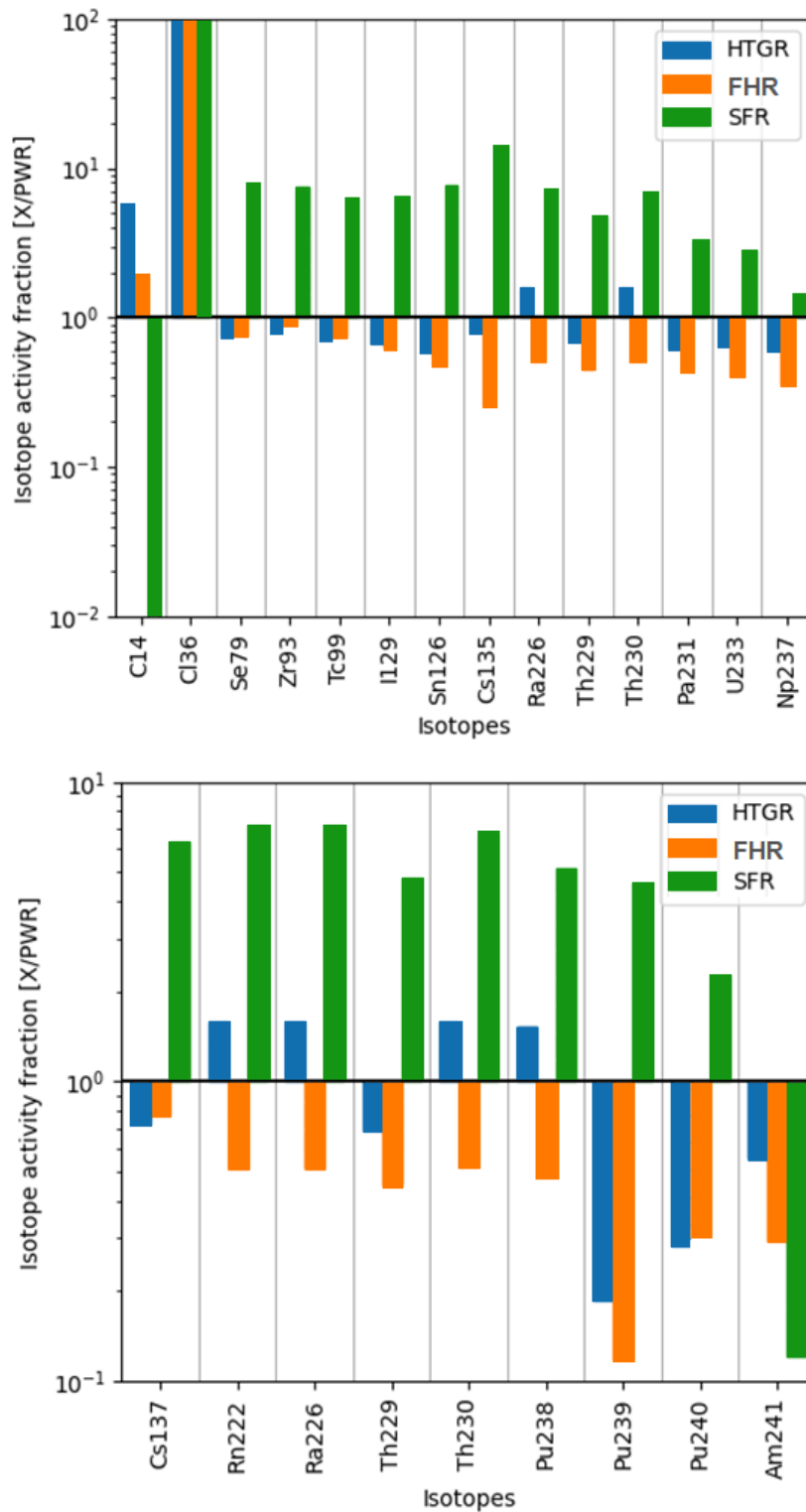
Figure 7-3
Comparison in isotope composition between the ARs and PWR at 300 years for all groundwater pathway scenario (GW-Clay, GW-Granite, GW-Salt) (left) and human intrusion scenario (HI) (right). Isotope activities in fraction of PWR reactor (>1 means isotope in ARs has higher activity than PWR and vice versa for <1).

7.1 References

1. IAEA, Status report 101 – Gas Turbine High-Temperature Reactor (GTHTR300C). 2011, IAEA.

8
RECYCLING SCENARIOS REVIEW

The majority of the existing nuclear fuel passes through the corresponding nuclear reactor only once, after which the nuclear fuel is considered waste. The industry standard practice to date for the reprocessing and re-use of constituent isotopes within the discharged fuel has been the solvent extraction process known as plutonium uranium redox extraction (PUREX), in combination with the production of MOX fuel rods that combine uranium and plutonium oxides. This single recycling step process produces fresh MOX fuel and vitrified high-level waste, which is designed for deep geological disposal. More advanced recycling scenarios are briefly discussed here and include multi-recycling possibilities for the AR concept variants selected for this study. A high-level discussion is provided on the impact of such recycling scenarios on waste streams, potential waste matrices, and other aspects of long-term waste management.

The fuel reprocessing technologies under development in support of innovative reactors may also have potential for nearer term application to optimize the capacity and performance of a geological repository. Recycling, fuel treatment, conditioning, and transmutation technologies have the potential to reduce the quantity and activity of waste requiring geological disposal. These technologies are not alternatives to geological disposal but help reduce the cost and optimize the use of a geological repository.

While there are multiple approaches to separating the components of nuclear fuel, two categories of reprocessing technology are more mature than others: aqueous processing and pyro processing. The aqueous PUREX process separates uranium and plutonium from a solution resulting from the dissolution of fuel assemblies in nitric acid. This solution contains more than 99% of the fission products and transuranic elements (TRUs). It is then treated by an organic solvent (kerosene) with tri-n-butyl phosphate (TBP) to extract U, Pu, and sometimes minor actinides. Pyro processing does not involve dissolution of fuel in an acid solution. Rather, the fuel is chopped and suspended in baskets in a molten salt bath through which an electric current flow. Most of the spent fuel radionuclides, including uranium, TRUs and fission products, dissolve into the salt. Whereas most of the fission products remain in the salt, uranium and TRUs are removed from the salt through deposition on different cathodes.

8.1 SFR

Spent fuel from sodium fast reactors may be convenient to reprocess using pyro processing. This involves sodium removal from the chopped fuel pins through distillation with cesium, followed by electrochemical dissolution in a molten salt. In a two-step distillation, uranium is removed first followed by removal of the remaining uranium together with plutonium and minor actinides. The bonded sodium will contain I-129 which then requires a HLW waste form [1]. Treatment of the molten salt involves extraction of a fraction of the salt which is then purified by removing uranium, plutonium and minor actinides. The purified salt is then adsorbed onto zeolite, thermally converted to sodalite together with the excess salt. The final product can be used as an additive in borosilicate glass. This process also generates C-14 from N-14 impurities in the cover gas [1].

8.2 Zirconium Recovery/Recycle and Uranium Removal from Metallic Waste

A potential zirconium recovery methodology for the U-Zr alloy waste involves volatilization with chloride. It is based on the so-called chloride volatilization method developed in the zirconium industry, where Zr is recovered from ore by mixing with C and reacting with Cl to produce $ZrC1_4$. The applicability of this method for nuclear alloy wastes was tested and involved chlorination and volatilization of Zr to finally recover zirconium as $ZrC1_4$ [2]. The end product would be a potential resource of Zr provided a sufficient removal of radioactive contamination can be achieved [2].

In this method the temperature conditions (i.e., low temperatures and high temperatures) and addition of O_2 to the Cl_2 gas were found to both affect the efficiency rate of the chloride volatilization process [2]. At a high temperature (1000 °C) and without any addition of O_2 to the chlorination process, the volatility ratios (V_z) for Zr and U were 100% and 96%, respectively, which resulted in a decontamination factor (DF) of 1 [41]. Specifically, no separation effect was found at high temperature without the addition of oxygen. DF increased from 1 to 3.8 when O_2 was added due to the low value of V_z, which may make it an unfavorable option from a production point of view. A maximum DF of 12.5 was achieved when the chlorination method is done at low temperature (400 °C) with the addition of O_2 [2]. The recovery of zirconium and removal of uranium was shown to be more efficient at low temperature than at high temperature. These findings indicate metallic fuels from SFRs have potential for reprocessing based on modifications of existing industrial processes.

8.3 HTGR

In general, HTGR technology developers indicate intent to operate in an open fuel cycle; if fuel recycling is to become part of the cycle, the fuel kernel must be separated from the coating layers to facilitate access of chemical reagents to the fuel particles during reprocessing. The HTGR designs will produce complex wastes, mainly associated with the activated C-14 waste produced by the dismantlement of graphite fuel blocks and the crushing of fuel compacts [1].

The following steps are considered prior to applying aqueous reprocessing to carbide or oxide fuel within the TRISO kernel: (i) mechanical extraction of fuel compacts from the fuel blocks, (ii) removal of graphite from the compacts, and (iii) removal of the carbon and SiC coating layers from the fuel kernels by high-temperature oxidation or by carbo-chlorination and separation of the fuel kernels [1].

For spent fuel from the pebble bed reactors, the spherical fuel elements and the coated fuel particles can be crushed by welding techniques. Following the crushing of the fuel particles, the fragmentation products can be dissolved in nitric acid similar to PUREX, though further research and demonstration was recommended [1].

8.4 FHR (MSR-Variant)

Discharged fuel from dissolved fuel molten salt reactors may be reprocessed using pyro processing. With this process the uranium and transuranic elements from fuel in a molten salt medium are recovered under high temperature using electrochemical methods or molten salt extraction [40]. Discharged fuel from a FHR like the variant considered in this investigation may be more suitable to the same approach as reprocessing of TRISO fuel from the HTGR variants.

Additional steps to account for removal of intruded salt may be necessary before the removal of the graphite binder phase.

The reprocessing of fuel from an MSR has not been well established. One of the challenges is how to confine tritium and how to manage the presence of Zr that could add complexity to reprocessing. However, many of the topics requiring further investigation are similar to classical pyrochemical solid fuel reprocessing studies, especially the separation of actinides and lanthanide [1].

8.5 References

1. (IAEA), I.A.E.A., Waste from Innovative Types of Reactors and Fuel Cycles. A Preliminary Study, in IAEA Nuclear Energy Series. 2019, IAEA: International Atomic Energy Agency (IAEA).

2. Sato, N., et al. Recovery of Zirconium and Removal of Uranium from Alloy Waste by Chloride Volatilization Method. in Proceedings of the IEEK Conference. 2001. The Institute of Electronics and Information Engineers.

9
CONCLUSIONS

This study investigates the impact on the back end of the nuclear fuel cycle of four advanced reactor concept variants spanning three technology categories: an SFR, a pebble-bed HTGR, a prismatic-block HTGR, and a FHR. The study considers the volume of waste produced, its decay heat production, activity, elemental, and radiological composition. Waste volumes generated from these reactors are expressed in cubic meter per gigawatt year of electricity production (m^3/GWe-yr) and are categorized into spent fuel, replaceable components (includes graphite reflectors, coolant fluid, and similar materials), and decommissioning components (e.g., reactor pressure vessel, steam generator).

In terms of spent nuclear fuel only, the SFR variant considered produces the least discharged fuel volume compared to the PWR (around 2/3 as much), while the HTGR variants considered produce 15× (from the pebble bed) and 60× (from the prismatic-block) the volume of PWR discharged fuel, and the FHR considered produces 4.3× that of PWR. However, all advanced reactors variants considered have a larger total waste volume when including waste from decommissioning, consumables and maintenance. The larger contributions in waste volume are mainly from the presence of carbon- and ceramic-based materials within the TRISO fuel kernels, graphite fuel block and reflectors. These waste streams vary considerably based on operational and design choices. The elective replacement scheme for graphite elements within the prismatic HTGR variant which are not present in the other designs clearly drive a significant increase in waste volume. This sensitivity to operational choices is present in many AR technology options. All the non-fuel waste streams have resultant activity which may be classified as LLW or ILW. Further work is required to separate LLW from ILW, and to determine a suitable disposal pathway. Regardless, the presence of longer-lived isotopes, such as C-14, within the discharged graphite wastes may result in a higher waste classification.

Discharged fuel from the HTGRs variants considered produce less decay heat which could allow the maximization of geological disposal capacity within thermal design limits, or support more efficient interim storage and disposal in the nearer term. The heat production from discharged FHR fuel is slightly higher than from PWR fuel during the first 10 years, then drops slightly below that of PWR until after 1000 years it exceeds again PWR. The lower activity and heat load appear to be a direct consequence of the higher burn-up that the ARs can achieve compared to a PWR.

Waste volumes from advanced reactors can potentially be reduced by waste treatment and separation techniques. For the HTGRs and FHR variants, the TRISO fuel forms can be separated into two parts: TRISO fuel kernels and the carbon from the graphite binder phase. Fuel processing and carbon conditioning must be considered separately after separation. If the bulk graphite can be successfully separated from the prismatic-block HTGR fuel compacts without any contamination, then this would allow the graphite to be disposed as LLW or ILW, which reduces volume for disposal in the geological repository. It remains to be seen if this approach will be viable, due to the potential activation of nitrogen contamination in the graphite block.

Based on the modeled isotopic composition of the discharged fuels, a subset of isotopes were selected for review. These were based on scenarios developed by a variety of international organizations for reference repository performance assessments. Three groundwater scenarios were reviewed, and isotopes of note from exposure in host rocks of clay, granite, and salt were identified from the literature. A similar review was performed for a human intrusion scenario based on a core sample brought to a lab. Results for the groundwater pathway show that, except for SFR fuel, all the advanced reactors variants which used TRISO fuel (i.e., HTGRs and FHR) discharged fuel containing lower amounts of notable radioisotope (e.g., Se-79, Tc-99, I-129) activities compared to PWR, except for C-14 and Cl-36. This is not a rigorous evaluation of the mobility of these isotopes within any modeled repository environment, but an indication of the relative concentration of isotopes that drive risk. For the human intrusion scenario reviewed, results showed that 300 years after the fuel has been released from the reactor, FHR fuel contains lower activities for all notable radioisotope compared to PWR spent fuel. The HTGR variant fuels displayed slightly higher activities for radionuclides from the U decay series such as Pu-238, Th-230, Ra-226, and Rn-222; conversely, lower activities were found for Cs-137, Th-229, Pu-239, Pu-240 and Am-241. For the SFR fuel all radionuclides of concern had higher activities than PWR, except for Am-241. For the AR variants with lower specific activities, the degree of reduction does not appear to be significant with regards to the consequence of release, as most are still within an order of magnitude of the activities in discharged PWR fuel.

Many differences clearly exist in the attributes of discharged advanced reactor fuel. This derives from initial enrichment, burn-up, chemical form, and more. Decay heat and specific activities may deviate by factors of two to five, isotopic composition varies within an order of magnitude, and volume of potentially challenging graphite waste occurs that has not been addressed in many regions. Historical experience with many potential AR fuels exists at the laboratory scale, but economic scale-up of these systems to commercial levels has yet to be demonstrated. This indicates that more effort is needed to mature technical solutions to industry applicability. Further regulatory challenges may exist, requiring the collection of more data. Closing these commercialization and regulatory gaps will likely require collaboration between many organizations around the world, for which there exist many avenues for success.

Export Control Restrictions

Access to and use of this EPRI product is granted with the specific understanding and requirement that responsibility for ensuring full compliance with all applicable U.S. and foreign export laws and regulations is being undertaken by you and your company. This includes an obligation to ensure that any individual receiving access hereunder who is not a U.S. citizen or U.S. permanent resident is permitted access under applicable U.S. and foreign export laws and regulations.

In the event you are uncertain whether you or your company may lawfully obtain access to this EPRI product, you acknowledge that it is your obligation to consult with your company's legal counsel to determine whether this access is lawful. Although EPRI may make available on a case by case basis an informal assessment of the applicable U.S. export classification for specific EPRI products, you and your company acknowledge that this assessment is solely for informational purposes and not for reliance purposes.

Your obligations regarding U.S. export control requirements apply during and after you and your company's engagement with EPRI. To be clear, the obligations continue after your retirement or other departure from your company, and include any knowledge retained after gaining access to EPRI products.

You and your company understand and acknowledge your obligations to make a prompt report to EPRI and the appropriate authorities regarding any access to or use of this EPRI product hereunder that may be in violation of applicable U.S. or foreign export laws or regulations.

About EPRI

Founded in 1972, EPRI is the world's preeminent independent, non-profit energy research and development organization, with offices around the world. EPRI's trusted experts collaborate with more than 450 companies in 45 countries, driving innovation to ensure the public has clean, safe, reliable, affordable, and equitable access to electricity across the globe. Together, we are shaping the future of energy.

3002023916

EPRI
3420 Hillview Avenue, Palo Alto, California 94304-1338 • USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com