ASSESSMENT OF THE CURRENT STATE OF KNOWLEDGE ON STORAGE AND TRANSPORTATION OF MOLTEN SALT REACTOR WASTE—FINAL REPORT

Prepared for

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Prepared by

George Adams, Patrick LaPlante, and Yi-Ming Pan

Southwest Research Institute®
Center for Nuclear Waste Regulatory Analyses

January 2023

EXECUTIVE SUMMARY

Research and development in Molten Salt Reactors (MSRs) continues to be extensive; however, operational experience is limited. The Molten Salt Reactor Experiment (MSRE) was conducted decades ago, and it remains the primary source of operational experience, having demonstrated some unique challenges with MSR salt storage.

Several companies are pursuing development of MSRs; however, many of the designs are preliminary. The open literature describes the layout of components as well as the type of fuel and salt but does not typically characterize the MSR salt waste nor provide detail on storing, transporting, or processing the waste. Salt waste varies by MSR design. Therefore, the current and potential MSR designs were analyzed based on current industry trends to characterize salt waste streams and to identify potential challenges associated with the safe storage, transportation, and processing of MSR waste. Understanding the challenges can help to determine information needs in the context of existing regulatory frameworks and guidance.

Many of the MSRs are in the early stages of development and detailed information is limited on any one of them. Therefore, this report focuses on groups of MSR designs that represent a range of molten salts and molten salt operations. Organizing them by group can shed light on how similar designs may handle processing, storage, and transportation of MSR salt waste. Group 1 MSRs maintain the fuel separately from a primary coolant salt. Group 2 MSRs use fluoride fuel salt and extend from the MSRE. Group 3 MSRs are fast-spectrum MSRs using a chloride fuel salt. Group 4 MSRs are two-fluid breeders having both a fluoride fuel salt and blanket salt.

During operations, fission products will build up in the molten salt, possibly requiring the salt to be processed to remove these waste products. For the two-fluid breeders in Group 4, continuous online processing is integral to the operation. For the remaining MSRs, the need for processing is unclear. At least one thermal-spectrum thorium MSR design in Group 2 is expected to evolve from offline batch processing to continuous online processing (Wang, 2021), though this type of technology is not expected near-term in the US. Other MSRs in Groups 1, 2, and 3 may require at least some salt cleanup to remove fission products that can interfere with reactor operations. Replacing used fuel salt may be an alternative to processing; Kairos Power is considering this approach for its Hermes test reactor in Group 1 (Kairos Power, 2021).

The extent to which molten fuel salt would be processed after discharge from a reactor is also not available in the public documentation for many of the MSR designs because the reactor vendors are not emphasizing waste management aspects of the designs. For some MSRs, actinides can be recovered and reused as fuel. This is an important concept because radioelements such as plutonium dominate the waste fuel salt activity after about 300 years. Recovering actinides would leave a salt waste stream consisting of shorter-lived fission products. Additionally, salts will likely require immobilization long-term, due to their hygroscopic nature. Therefore, processing the molten salt could reduce the total volume of waste that may need to be stored and could also reduce the amount of waste remaining during long term storage or disposal timeframes, and ultimately could allow for more efficient waste form production. It is feasible that some or all potential waste processing operations, including waste form development, could be carried out at the reactor site or at another site, such as a centralized processing facility which would service more than one MSR. Processing and managing irradiated reactor fuel involves working with materials that generate very high radiation levels. Controls such as remote handling and adequate shielding are needed as described in technical reports that detail the MSRE experience.

Limited publicly available information was found on the composition of MSR salt waste and plans for storing this waste, including waste forms. In some designs such as the Group 2 MSRs, the vendor plans to store the salt waste onsite indefinitely or until reactor decommissioning. The Group 2 MSRs are similar to the MSRE in that there is the potential for fluorine generation during long term storage of salt waste. Although mitigation measures are described for the MSRE and in other technical reports, no information was found on how this would be addressed for the new MSR designs.

Limited information was found on transporting MSR salt waste. The most recent information comes from two test reactors currently under development, but no details were available—only high-level plans. A prior Oak Ridge National Laboratory (ORNL) evaluation of MSRE salt waste disposition alternatives provides more detailed conceptual design information related to transportation of salt waste. It includes the use of an existing transportation cask and discusses how fluorine gas could be addressed in the shielded container design. However, ORNL did not conduct a detailed evaluation nor address whether the proposed approach would meet package specifications.

This report reviews several MSR designs. In some cases, high-level plans for processing, storing, and transporting MSR salt waste were found, but with limited details related to the specific designs. Research has continued for decades on MSRs and some of this research is included in this report. The most recent application of this research is with two test reactors currently under development, and they are both discussed in this report. Still, limited information is available even for these reactors on processing, storing, and transporting MSR salt waste. Therefore, this report includes a list of insights, summarized here:

Early interaction between industry and the NRC should be encouraged to identify information or technologies which could be relevant to licensing/certification reviews. Topics that may be relevant include:

1. Plans for co-located chemical processing facilities because some waste management strategies include the recovery of actinides and separation of fission products in a co-located facility.

2. Information on tritium management strategies because many MSRs would generate significantly more tritium than currently operating reactors. Information is needed on the expected amount of tritium produced, capabilities for capturing it, and methods for storing and disposing of the tritium waste.

3. Salt waste streams need to be characterized in terms of their composition and assessed for any potential chemical and radiological hazards during processing, storage, and transportation.

4. The potential for long-term storage of MSR salt waste as the MSRE revealed challenges with storing salt waste for decades.

5. The NRC should consider forward-looking knowledge management on phenomena which could impact design analysis like fluorine gas production in irradiated fluoride

salts and tritium production. Investigations of any knowledge gaps should be conducted if necessary.

6. Any new proposed designs or design features that could require significant staff resources for regulators to review due to their relative newness should be identified early and communicated with the NRC.

7. Plans for transporting salt waste that may contain fission products and actinides. If the intent is to transport the waste as part of a sealed core unit, then the means for doing so need to be developed while that core unit is being designed.

8. New glass or ceramic waste forms for the immobilization of MSR salt waste. Due to the properties of the different salts being proposed for use, it would be difficult to immobilize all salt types in a single waste form.

9. Designs for containers for the transportation and storage of used chloride fuel salts from chloride salt MSRs.

REFERENCES

Kairos Power. “Hermes Non-Power Reactor Preliminary Safety Analysis Report.” HER-PSAR-001. Revision 0. Kairos Power. September 29, 2021. <https://www.nrc.gov/docs/ML2127/ML21272A378.pdf> (Accessed 9 December 2022).

Wang, B. “China’s Molten Salt Nuclear Reactors.” Next Big Future. August 23, 2021. <https://www.nextbigfuture.com/2021/08/chinas-molten-salt-nuclear-reactors.html> (Accessed 20 September 2022).

TABLE OF CONTENTS

LIST OF TABLES

ACKNOWLEDGMENTS

This report was prepared to document work performed by the Center for Nuclear Waste Regulatory Analyses (CNWRA®) for the U.S. Nuclear Regulatory Commission (NRC) under Contract No. 31310018D0001. The activities reported here were performed on behalf of the NRC Office of Nuclear Regulatory Research. The report is an independent product of CNWRA and does not necessarily reflect the views or regulatory position of the NRC.

The authors thank the NRC staff for constructive comments on a draft version of this report, Osvaldo Pensado for his technical review, and David Pickett for his programmatic review. The authors also thank Arturo Ramos for providing formatting and word processing support in preparation of this document.

QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT DATA

DATA: There are no original CNWRA-generated data in this report. Sources of other data should be consulted for determining the level of quality of those data.

ANALYSES AND CODES: No codes were used in the analyses contained in this report.

ABBREVIATIONS/ACRONYMS

ACU	Abilene Christian University
Am	americium
BeF_2	beryllium fluoride
CANDU	CANada Deuterium Uranium
cemet	ceramic-metal composite
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
Cm	curium
CMSR	Compact Molten Salt Reactor
CNRS	National Centre for Scientific Research
CNWRA®	Center for Nuclear Waste Regulatory Analyses
CWF	ceramic waste form
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
EPA	U.S. Environmental Protection Agency
FCF	Fuel Conditioning Facility
FHR	fluoride salt-cooled high-temperature reactor
FSB	Fee, Seed, and Breed
FS-MSR	fast spectrum molten salt reactor
GAIN	Gateway for Accelerated Innovation in Nuclear
HALEU	high-assay low-enriched uranium
HLW	high-level radioactive waste
HWR	heavy water reactors
IAEA	International Atomic Energy Agency
IET	Integrated Effects Test
IFR	Integral Fast Reactor
IMSR®	Integral Molten Salt Reactor
INL	Idaho National Laboratory
KP	Kairos Power
LANL	Los Alamos National Laboratory
LEU	low enriched uranium
LFTR	Lithium Fluoride Thorium
LiF	lithium fluoride
LLRW	low-level radioactive waste
LWR	light water reactor
MCFR	Molten Chloride Fast spectrum Reactor
MCRE	Molten Chloride Reactor Experiment
MCSFR	Molten Chloride Salt Fast Reactor

MOSART	MOlten Salt Actinide Recycler and Transmuter
MSR	Molten Salt Reactor
MSRE	Molten-Salt Reactor Experiment
MSRP	Molten Salt Reactor Program
MSRR	Molten Salt Research Reactor
MW_{th}	megawatt thermal power
MW_e	megawatt electric power
Np	neptunium
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
Pu	Plutonium
SEU	slightly enriched uranium
SINAP	Shanghai Institute of Applied Physics
SLZ	salt-loaded zeolite
SMR	small modular reactor
SNF	spent nuclear fuel
SRS	Savannah River Site
SSR	Stable Salt Reactor
Ti	titanium
TiT_2	titanium tritide$_2$
TMSR	Thorium Molten Salt Reactor
TMSR-LF	Thorium Molten Salt Reactor liquid-fueled
TRISO	tristructural isotropic
TRL	technical readiness level
TRU	transuranic
WaTSS	Waste to Stable Salt
WIPP	Waste Isolation Pilot Plant

1 INTRODUCTION

1.1 Background

To help prepare for regulatory interactions and potential license applications for Molten Salt Reactor (MSR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff is investing effort to identify the potential technical and regulatory challenges associated with the storage and transportation of the associated salt waste and potential waste forms. Unlike light-water reactors (LWRs), MSRs use molten salt as the coolant and/or fuel and would therefore require management of novel waste streams. In addition, it is feasible that salt processing[1] would be needed to remove fission products, recover fissile and fertile material, and immobilize the waste salt. There is no NRC regulatory precedent for licensing such operations. Therefore, it is important to identify the current experience with MSR salt storage, transportation, and processing as well as the applicability of current spent fuel management approaches and technologies to the storage and transportation of the MSR salt waste (and waste forms), including potential high-level waste generated.

Much of the experience with MSRs comes from the Molten Salt Reactor Experiment (MSRE) in the 1960s. Many of the MSR designs described in this report build on this experience. However, modern MSRs are in the early stages of development and limited information is available on processing, storage, and transportation of the molten salt. MSRs that are the furthest along are two test reactors. One is the Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) and the second is the Kairos Power (KP) Hermes Test Reactor. Both are described in this report along with any publicly available information that could be found on other MSRs under development.

This report primarily addresses fluoride salt MSRs because most of the information was available for this type. The use of a fluoride salt in an MSR extends from the MSRE to many of the current designs. Wherever possible, this report includes information for chloride-salt MSRs; however, discussions are limited because limited information is publicly available.

This report is organized as follows. Chapter 1 describes the background and purpose of this document. Chapter 2 summarizes MSR developments from the MSRE to the current designs. Chapter 3 describes experience in MSR salt storage, transportation, and processing. While Chapter 3 centers primarily on the MSRE work at Oak Ridge National Laboratory (ORNL), it also includes some discussion of the Fuel Conditioning Facility (FCF) at Idaho National Engineering Laboratory (now INL). Chapter 4 describes the current and potential MSR designs as they affect the processing, storage, and transportation of MSR salt. Chapter 5 provides insights from this study, and Chapter 6 presents the summary and conclusions.

1.2 Purpose and Scope

The objective of this project is to collect and review information from the open literature on experience with storage and transportation of MSR waste and salt fuel processing operations. The review includes available information on the nature of the fuel, waste streams, and associated management (including waste forms), and any storage and transportation experience both domestic and international. Included in this assessment is the current state of

[1] Salt and fuel processing in the context of this report does not include preparation of fresh salt fuel for use in the reactor.

knowledge of the U.S. Department of Energy (DOE) and commercial plans for spent fuel and high-level waste management. The review was conducted with the goal of identifying and assessing potential challenges associated with the safe storage, transportation, and processing of MSR salt waste.

2 SUMMARY OF MSR DEVELOPMENT

This section describes DOE research and current MSR developments, both domestic and international. Section 2.1 provides some background on DOE research extending from the 1950s to the present, Section 2.2 focuses on MSR development by U.S.-based companies, and Section 2.3 describes MSR development in countries outside the U.S.

2.1 Department of Energy Research

DOE research on MSR technology started with the aircraft reactor experiment in the 1950s, followed by the MSRE in the 1960s. With a resurgence of interest in MSR technology for power generation, the DOE presently supports a Molten Salt Reactor Program (MSRP). These research and development efforts related to MSR technology are summarized in the following sections.

2.1.1 The Molten Salt Reactor Experiment

The design and history of the MSRE has been described in detail in various reports (Peretz, 1996; Notz, 1988; Haubenreich and Engel, 1970; Robertson, 1965). The MSRE was an 8 MW reactor that was operated at ORNL from 1965 through 1969 to demonstrate Molten Salt Breeder Reactor technology. The reactor used a liquid fuel formed by dissolving UF_4 in a carrier salt composed of a mixture of LiF, BeF_2, and ZrF_4. The fuel salt circulated through a reactor vessel, a fuel salt pump, and a primary heat exchanger at temperatures above 600 °C (1,112 °F). In the reactor, the salt was forced through channels of graphite to provide the geometry and moderation necessary for a nuclear chain reaction. Heat was transferred from the fuel salt to a secondary coolant salt in the primary heat exchanger. The secondary coolant salt was a mixture of LiF (66%) and BeF_2 (34%).

Each of the salt loops included drain tanks so the salt could be drained out of either circuit by gravity (Peretz, 1996). A third drain tank connected to the fuel salt loop was provided for storing a batch of flush salt. The flush salt, similar in composition to the coolant salt, was used to condition the fuel salt loop after it had been exposed to air and to flush the fuel salt loop of residual fuel salt and contaminants before accessing the reactor circuit for maintenance or experimental activities. Each salt drain tank was suspended in a furnace assembly with heaters to maintain the salt melted (i.e., above the solidus temperature). The capacity of each tank was sufficient to contain the entire inventory of fuel or flush salt. The reactor and drain tank cells were designed as sealed containment systems (Peretz, 1996). When reactor operations ended in December of 1969 the salts were allowed to solidify at ambient temperature and remain in the tanks. The tanks would be maintained pending the availability of a geologic repository or other viable disposal option. Disposal options considered and the development of an analysis of alternatives (Peretz, 1996) identifying specific scenarios for management of the MSRE salts is further described in Chapter 3.

2.1.2 The Molten Salt Reactor Program

With the industry's recent interest in small modular reactors and the use of molten salts in some reactor designs, DOE renewed its focus on MSR research under its Molten Salt Reactor Program. Paviet (2022) states the vision for this program is to work with industry stakeholders to address the challenges of MSR technology and enable MSRs to enter the commercial market. There are currently two primary MSR designs[2]: a salt-fueled (liquid-fueled) design and a salt-cooled design. In a salt-fueled design, the salt is a combination of fissile salt such as UF_4 or UCl_3 with nonradioactive fluoride-based or chloride-based diluent or carrier salts. The salt serves as both the fuel and the reactor (or primary) coolant. The MSRE, for example, was a salt-fueled design with a fluoride-based carrier salt. In a salt-cooled design, the fuel can be in a solid form with a molten salt as coolant. For example, fluoride salt-cooled high-temperature reactor (FHR) designs may use tristructural isotropic (TRISO) graphite-matrix coated fuel particles in a liquid fluoride salt coolant (Forsberg and Peterson, 2015). Currently, most of the industry designs are salt-fueled designs. Therefore, this report focuses more heavily on salt-fueled designs than salt-cooled designs.

Riley et al. (2018) describe research applicable to unseparated and separated salt waste streams for both chloride-based[3] and fluoride-based MSRs. The purpose of separating salt waste is to recover components that can be recycled such as ^{7}Li and ^{37}Cl, to remove nonradioactive components that can dilute the waste, and to remove components that may present a health hazard such as BeF_2. Riley et al. (2018) list the most promising waste form options for separated and unseparated salt waste to include glass-bonded sodalite and apatite ceramics, ceramic-metal composites (cemets), and phosphate glass. They assert that glass-bonded sodalite is ideal for immobilizing mixed chloride-based salt wastes; whereas, glass-bonded apatite is better for fluoride wastes.

McFarlane et al. (2019) describe hazards associated with MSR processes. Relevant to this study are the hazards they tabulate for activities after shutdown in Table 3f of their report, and hazards associated with waste form preparation in Table 3g. These tables include a technical readiness level (TRL)[4] qualitative assessment [TRL 1 (low) to 9 (high)] and a qualitative severity judgment[5] [Rank 1 (low) to 5 (high)]. All the activities after shutdown listed in Table 3f have a TRL level 2 indicating the current state of knowledge is low. This includes activities such as onsite storage of chloride-based and fluoride-based spent salt as well as packaging the salt and transporting it to off-site facilities. Many of the waste form preparation activities listed in Table 3g were deemed TRL level 2. However, a higher TRL level 5 was assigned to actinide recovery of chloride salts by pyroprocessing as well as actinide recovery of fluoride salts by hydrofluorination. Actinide recovery of fluoride salts by dissolution in acid was deemed TRL level 8, reflecting greater experience with this activity.

2.2 U.S. Developments

MSRs under development in the U.S. are listed in Table 1. This table provides a brief overview of the project and any information related to salt waste that could be found. Note that two of the designs in Table 1 are fast spectrum molten salt reactors (FS-MSRs). Holcomb et al. (2011)

---

[2] Note however, the Moltex Stable Salt Reactor (SSR) does not fit directly into either of these two classifications as described in Table 2.
[3] No chloride-based MSR has been brought to criticality, so chloride-based MSRs are conceptual (Riley et al., 2018).
[4] The TRL is a qualitative assessment of the current state of knowledge for the activities.
[5] The qualitative severity judgment ranks the potential severity of hazards to operations staff.

describe features of FS-MSRs that have potential fuel cycle applications, including the ability to consume actinides from used LWR fuel. Furthermore, DOE (n.d.) describes the agency's support for Exodys Energy (formerly Elysium Industries) through its Gateway for Accelerated Innovation in Nuclear (GAIN) program. Exodys Energy is developing a molten chloride fuel salt from actinides chemically recovered from used nuclear fuel. It should be noted that many of these reactor designs are under development and that available details regarding fuel salts, coolants, and waste management options may not be current or may subsequently change.

Table 1 Summary of U.S. Projects involving MSRs

Project	Description
Molten Chloride Fast Reactor (MCFR)—Terra Power	The MCFR is conceptualized as a fast spectrum, liquid-fueled, chloride salt reactor capable of 600 to 2,500 MW_{th} (Kramer, 2018). It is designed with online refueling for continuous operation and capable of using multiple fuels to include depleted and natural uranium as well as spent fuel (DOE, 2021). As described by Latkowski (2021), a variety of waste disposal options are being considered to include: • Direct disposal in a salt repository (no ^{37}Cl recovery), • SynRoc[6] with recovery of ^{37}Cl, and • Conversion to accepted waste forms such as vitrification to Fe-phosphate glass.
Fast Chloride-Molten Salt Reactor *FC-MSR)—Exodys Energy (formerly Elysium Industries)	Currently focused on designing, licensing, and constructing a 10 MW_{th} liquid-fueled, Fast Chloride-Molten Salt Reactor (FC-MSR) non-power demonstration unit and a nuclear waste-to-fuel salt conversion facility (DOE, 2022)
Kairos Power (KP) – Fluoride High Temperature Reactor (FHR)	Kairos Power (KP) is iteratively designing and developing a fluoride salt-cooled high temperature reactor (KP-FHR) it designates KP-X[7] (KP, 2022; KP, 2019). This is a commercial 140 MW_e pebble-bed reactor using TRISO coated fuel particles and a FLiBe[8] molten salt coolant. NRC (2020) states that the use of FLiBe as coolant will generate a mixed hazardous waste that will have to be addressed during operation and decommissioning. Furthermore, NRC (2020) states that although TRISO fuel has excellent fission product retention capabilities, some fission products may be

[6] SynRoc is a ceramic (also referred to as a synthetic rock) used to incorporate high-level radioactive waste (HLW) elements into its crystal structure thereby immobilizing them (World Nuclear Association, 2019).
[7] KP is also developing the 35 MW_{th} Hermes Test Reactor as part of this iterative process.
[8] FLiBe is a mixture of lithium fluoride (LiF) and beryllium fluoride (BeF_2), with a nominal chemical composition of $2LiF{:}BeF_2$ (NRC, 2020).

Table 1 Summary of U.S. Projects involving MSRs	
Project	**Description**
	released to the coolant and therefore, cleanup of the coolant may be required during operation. Consistent with its iterative development strategy, KP is currently focused on the Hermes Test Reactor that will be used to gain operational knowledge and experience with this technology as development proceeds towards KP-X development. The NRC received an application for the Hermes construction permit for the Test Reactor on September 29, 2021 and has issued a final safety evaluation report (FSER).
Lithium Fluoride Thorium Reactor (LFTR) – Flibe Energy	The LFTR is a thermal spectrum, liquid-fueled, graphite-moderated two-fluid breeder design using fluoride salts (i.e., FLiBe) for the primary fuel and blanket. The primary fuel salt contains fissile ^{233}U and the blanket salt contains fertile ^{232}Th. The blanket salt is processed during operations to replenish ^{233}U in the primary fuel salt. As described by Flibe Energy (2022), the LFTR produces only as much ^{233}U as it consumes. Furthermore, Flibe Energy (2022) asserts that 83% of the waste products generated can be safely stabilized within 10 years.
Molten Salt Research Reactor (MSRR) – Abilene Christian University (ACU)	The MSRR is a one-megawatt thermal power (MW_{th}) molten fluoride salt research reactor design with uranium tetrafluoride dissolved in the fuel salt. The reactor has a loop configuration that operates at high temperature and low pressure similar to the MSRE. The proposed nominal fuel salt composition is 67.2% LiF, 27.8% BeF_2 and (5%) UF_4. ACU anticipates the fuel would be approximately 19.75 percent enriched in U-235 with lithium enriched greater than 99.99 percent Li-7. The proposed coolant salt is composed of 67LiF-33BeF_2. The proposed MSRR is to be located within a building on the university campus (ACU, 2022). The NRC received an application for a construction permit for the MSRR on August 12, 2022.

2.3 International Developments

Several international projects have also been under development recently. Table 2 lists these projects with the country leading the development effort.

Table 2. Summary of International Projects involving MSRs		
Project	**Country**	**Description**
FUJI Molten Salt Reactor—Thorium Tech Solution	Japan	This reactor may no longer be under development. The original intent was to build a 160 MW_e liquid fluoride thorium reactor using technology from the MSRE (Halper, 2013). This reactor is a liquid-fueled design combining thorium with spent fuel. As described by Halper (2013), beryllium is a controlled material because of its toxicity, and it does not work well with plutonium. Consequently, the company was investigating the use of sodium in the molten salt (e.g., FLiNaK) instead of the more typical FLiBe salt used in the MSRE.
Integral Molten Salt Reactor (IMSR®)—Terrestrial Energy	Canada	The IMSR is a liquid-fueled MSR design with an output of 400 MW_{th} or 190 MW_e. It uses standard assay low enriched uranium (LEU) of less than 5% ^{235}U and a graphite moderator. The core has a seven-year lifespan and is started with carrier fluoride salts plus UF_4 as slightly enriched uranium (SEU)[9]. Makeup fuel salt of 4.95% enrichment is added over the seven--year period resulting in a final salt volume that is about 50% larger than at the start (Choe et al., 2018). Each plant would have space for two reactors. When a core is replaced, it is left for fission products to decay and then removed for off-site processing (World Nuclear Association, 2021).
MOlten Salt Actinide Recycler & Transmuter (MOSART)	Russia	Feynberg (n.d.) describes the MOSART reactor design as a liquid-fueled fast spectrum MSR that does not use a graphite moderator. It is fueled by transuranic (TRU) waste from light water reactors (LWRs). It uses a FLiBe salt that is processed using reductive extraction of actinides into liquid bismuth (Ignatiev, 2017).

[9] SEU is less than 2% ^{235}U.

Table 2. Summary of International Projects involving MSRs		
Project	**Country**	**Description**
Compact Molten Salt Reactor (CMSR)—Seaborg	Denmark	As described in Seaborg Technologies (2022), the CMSR is currently in concept verification with detailed design beginning in 2024. It is envisioned as a liquid-fueled MSR developed as a Power Barge capable of delivering up to 800 MW_e. After its 12-year fuel cycle, the fuel is returned to the supplier and short-lived fission products are separated and sent to storage with the remaining fuel salt mixed into new CMSR fuel. The CMSR is being developed as a waste burner fueled with spent LWR fuel and thorium. In the future, thorium, plutonium, and minor actinides are planned as fuel sources (World Nuclear Association, 2021).
Thorium Molten Salt Reactor (TMSR) -- Shanghai Institute of Applied Physics (SINAP)	China	TMSR liquid-fueled (TMSR-LF) reactors are being designed and developed as ^{235}U-Th fueled reactors with the fuel dissolved in a FLiBe molten salt (Wang, 2021)[10]. TMSR-LF1 is a 2 MW_{th} pilot plant that recently received commissioning approval (Nuclear Engineering International, 2022a), TMSR-LF2 would be a 10 MW_{th} experimental reactor, and TMSR-LF3 would be a 100 MW_{th} demonstration reactor. Initially, all the salt would be discharged after 5 to 8 years and processed in batch form to remove fission products and minor actinides which would be sent to temporary storage. However, SINAP is developing a continuous process that would recycle salt, uranium and thorium. This process would also remove fission products that can be sent to interim storage or geologic disposal (Wang, 2021).
Stable Salt Reactor (SSR)—Moltex Energy	United Kingdom, Canada	The Moltex SSR differs from most other molten salt designs because the molten salt fuel is held in conventional fuel tubes that are cooled by the primary coolant molten salt (Nuclear Engineering International, 2022b; Scott, 2020; and Scott, 2016). Heat is transferred from the primary coolant to a secondary coolant molten salt for powering turbines and generating electricity. Moltex has developed the Waste to Stable Salt (WaTSS) process to create fuel for the reactor from

[10] Wang (2021) describes MSRs as liquid-fueled instead of salt-fueled and solid-fueled instead of salt-cooled.

Table 2. Summary of International Projects involving MSRs		
Project	**Country**	**Description**
		highly radioactive, long-lived CANada Deuterium Uranium (CANDU) waste. This process generates a highly radioactive but relatively short-lived salt waste stream in addition to the fuel for the SSR. For the SSR-W version, the fuel salt is 25% $PuCl_3$ with 30% UCl_3 and 45% KCl contained in fuel tubes; primary coolant salt is ZrF_4-KF stabilized with ZrF_2 and secondary coolant salt is a nitrate salt buffer.
ThorCon MSR—Martingale	Indonesia	ThorCon is designing a 500 MW_e MSR that can be contained within a barge. It is a scale-up of the MSRE (ThorCon, 2022a). The project is in the early stages although ThorCon (2022b) describes that much of the design phase has been completed. A pre-fission test facility is planned for testing the concept (i.e., without fuel salt). Afterwards, shipyard construction of the MSR is planned. The TMSR-500[11] would be graphite moderated and use a HALEU fuel salt (NaF-BeF_2-ThF_4-UF_4) with an enrichment of 19.75% ^{235}U (Manik et al., 2020). ^{7}Li fluoride is avoided due to cost; every four years the entire primary loop is changed out, returned to a recycling facility, decontaminated, disassembled, inspected, and refurbished (World Nuclear Association, 2021).
Molten Salt Fast Reactor (MSFR) —The National Centre for Scientific Research (CNRS)	France	As described by IAEA (n.d.), CNRS has been developing a liquid fueled 1,500 MW_e MSFR using a LiF molten salt containing actinide fluorides of Th, ^{233}U, ^{235}U, and/or Pu. A fuel processing unit extracts a few liters per day of fuel salt for fission product removal and then returns the cleaned fuel salt to the reactor. In addition, Boussier et al. (2012) describe additional offline salt processing to extract lanthanides.

[11] Manik et al. (2020) refers to the ThorCon MSR as TMSR-500, but other references do not specifically use this name convention. Therefore, within this report, the term ThorCon MSR will be used instead of TMSR-500.

3 EXPERIENCE RELATED TO STORAGE, TRANSPORTATION, AND PROCESSING OF MSR SALT

3.1 Molten Salt Reactor Experiment

Following shutdown of the MSRE, ORNL observed the production of fluorine gas from the irradiation of the solid fuel salt (Peretz, 1996; Haubenreich, 1970). Irradiation produced fluorine radicals that then combined to form F_2 that would diffuse to the salt surface. Further investigation found that fluorine production was inhibited by elevating the temperature. To prevent the accumulation of fluorine in the drain tanks, ORNL instituted an annual annealing operation where the drain tank temperature was raised to 149 °C (300 °F), well below the melting temperature, for about one week. Annealing continued until 1989 (Ablequist, 2021). In 1994, ORNL confirmed uranium (UF_6) migration into the off-gas system which created a criticality concern. ORNL then installed a reactive gas removal system to remove the uranium from the off-gas system. Defueling occurred between 2001 and 2008 where fuel salts were melted and chemically treated. The molten salts were fluorinated to remove uranium that was transferred to another building for storage. Removal of fuel salt in the two fuel drain tanks and flush salt in the flush tank was attempted using a heated probe but transfer pipes reportedly became clogged, transfer operations were aborted (ORSSAB, 2010), and residual salts presently remain in the tanks pending resolution of an implementable path to final disposition (McMillan, 2019).

Over the intervening years since the shutdown of the reactor, ORNL has evaluated options for dispositioning the MSRE fuel salt waste including disposal in a geologic repository for commercial high-level radioactive waste, disposal at the Waste Isolation Pilot Plant (WIPP) in New Mexico, or on-site disposal at ORNL (Peretz, 1996; Notz, 1988; 1985). Final disposition is described here because the disposition option affects the storage and transportation activities that would be required or not. Because of limited options for disposal and regulatory complexities, continued storage of the salt waste material has been necessary for several decades.

An early evaluation of disposition options (Notz, 1988) concluded that the fuel salt could not be left in its present form and location permanently; however, ORNL expected the material could be stored in its present form for 20 or 30 years or possibly longer. ORNL suggested the fuel salt should be melted and repackaged into smaller containers, to facilitate handling and to provide a basis for future, permanent disposal. During such an operation, a fluorine getter[12] could be added, which would eliminate the need for periodic reheating. The repackaged material could then be stored in a manner similar to remotely-handled transuranic waste (RH-TRU), an operation already conducted routinely at ORNL. ORNL expected that, in the repackaged form, interim storage could be done safely for many years.

In the following decade, after the U.S. Environmental Protection Agency (EPA) placed the ORNL site on the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) National Priorities List for cleanup, ORNL conducted an evaluation of disposal alternatives for the MSRE salt fuel in accordance with CERCLA requirements for remedial actions (Peretz, 1996). That analysis considered the following alternatives: (i) permanent disposal in the drain tanks; (ii) disposal of all key contaminants in the federal repository,

[12] A getter is a metallic compound used to capture a gas. Gas molecules are absorbed by the getter or combine with it chemically.

(iii) disposal of separated uranium; (iv) disposal of the key contaminants in the salt residue in WIPP; (v) disposal of the key contaminants in the salt residue in the federal repository; (vi) reuse of the salt; and (vii) interim storage. For the geologic disposal alternatives evaluated, disposal at WIPP was found to be technically compatible (waste was similar to TRU waste; repository was salt) however existing programmatic and regulatory constraints presented challenges (e.g., WIPP was limited to defense waste, disposal of high-level radioactive waste was not allowed, and the MSRE salt fuel had not been formally classified by DOE). The same challenges in utilizing WIPP would be faced by any commercial (non-defense) application of MSR technology.

Following evaluation of the alternatives, ORNL developed a series of preferences including the following: (i) if reuse of the fuel and/or flush salts is a serious option then it should be selected; (ii) if any of the geologic disposal alternatives prove technically and programmatically implementable, every reasonable effort should be made to implement that alternative. In particular, ORNL noted, efforts should be made to resolve the regulatory and programmatic obstacles to disposal at WIPP; (iii) at the present time only the interim storage option is likely to be implementable. ORNL further noted that interim storage provided adequate protection of human health and the environment, but also indicated that at some time an ultimate disposition must be identified to end the cost of perpetual care and monitoring. ORNL determined storage of fluorinated, stabilized salt could meet the quantitative requirements for disposal at WIPP or could serve as the feed for any of the other processes evaluated. Thus, ORNL recommended interim storage as being consistent with ultimate disposition and being fully implementable at the time. ORNL further noted that permanent disposal of the salts in their drain tanks (i.e., *in situ*) was not recommended and that the long-lived actinides in the salt should be placed in geologic disposal. They also suggested that the penetrating radiation from fission products and uranium decay daughters will require the waste be remotely handled. While federal high-level waste repository waste acceptance criteria were not available during the time of the ORNL alternatives analysis, subsequent evaluations have identified potential challenges to disposal of untreated MSR salts in a federal high-level waste repository. These challenges are related to hazardous material content, corrosivity, and waste form performance (Riley et al., 2018). More recent developments indicate ORNL is re-evaluating the feasibility of *in-situ* entombment and possibly revising the CERCLA record of decision regarding the removal action (McMillan, 2019).

Specific scenarios for management of the MSRE salts in the ORNL alternatives analysis (Peretz, 1996) included reusing at Los Alamos National Laboratory (LANL); separating the uranium and chemically stabilizing the salt; calcining and possibly eventually vitrifying the salt at INL; blending the salt into the feed stream to the Defense Waste Processing Facility at Savannah River Site (SRS); and constructing an electrorefining system or other new process at ORNL or another site. The description in the ORNL alternatives analysis of applying electrorefining to the MSRE salt waste includes detailed steps, the resulting waste materials from each step, and how such an approach is related to and could be integrated with the EBR-II electrorefining process at INL.

Further processing involving other sites such as INL would involve transportation. The ORNL alternatives analysis presumed a container for salt removed from the drain tanks that was consistent with packaging defined in the WIPP waste acceptance criteria (e.g., the 72-B transportation cask). Although weights and shielding requirements were not defined, ORNL presumed that expected package surface dose rates would be consistent with the expected use of the 72-B cask. The 72-B[13] cask is part of a transportation system developed to ship wastes from a number of DOE sites to WIPP, including the shipment of Remote-Handled Transuranic (RH-TRU) waste from ORNL to WIPP. ORNL noted transportation, particularly arrangements between states for acceptance of waste shipments, could significantly affect the implement ability of several options. They noted that shipping packages or casks would be needed to implement any of the ultimate disposition strategies. The actual status of the 72-B cask, or other containers and casks that might be used to ship salt or salt components, were not thoroughly explored by ORNL at the time.

3.2 Electrochemical Separations at the Fuel Conditioning Facility

The Fuel Conditioning Facility (FCF) at Idaho National Engineering Laboratory (now INL) has had a long history of operation (DOE, 1996). Between 1964 and 1969, the facility was called the Fuel Cycle Facility, and it demonstrated technology for recycling fuel into the adjacent Experimental Breeder Reactor (EBR)-II. About 700 spent nuclear fuel assemblies were recycled using a pyrometallurgical (melt refining) process. Between 1969 and 1994, the facility was used for a variety of applications, including storage of spent EBR-II fuel and refabrication of fuel for use in EBR-II. The FCF includes two operating cells, one with an air atmosphere for handling intact fuel and the other with an inert argon atmosphere for conducting operations, including electrorefining, with exposed nuclear materials. Presently, the FCF's primary mission is to support treatment of DOE-owned sodium-bonded metal fuel. The FCF also supports work to refine the technical feasibility of pyroprocessing for treating used nuclear fuel for DOE's Fuel Cycle Research and Development Program. Pyroprocessing may be used to process salt waste and is a family of technologies involving high-temperature chemical and electrochemical methods for separation, purification, and recovery of fissile elements from used nuclear fuel (INL, 2021).

The waste salt from electrochemical reprocessing (or pyroprocessing), used to separate fissile material from EBR-II spent fuel, is made into a ceramic waste form (CWF). Because of the high chloride content of the salt, it cannot be directly vitrified using the conventional borosilicate approach (Lee et al., 2019). Halogen solubility limits in sodium aluminoborosilicate glasses are very low. According to Priebe and Bateman (2008), the salt has to be ground to a fine particle size (45-250 μm) and mixed with Zeolite 4A ground to the same size. The salt and zeolite are heated to 500 °C for about 18 hours, during which time the salt and zeolite are continually mixed. The salt-loaded zeolite (SLZ) is cooled and then mixed with borosilicate glass frit with a comparable particle size range. The SLZ glass mixture is transferred to a crucible, which is heated to 925 °C [1,697 °F]. This process converts the zeolite to a final sodalite form. The glass thoroughly encapsulates the sodalite, producing a dense leach-resistant final waste form. The quality of the final CWF is established by visual observation, density, product consistency test, X-ray diffraction, and scanning electron microscopy (Johnson et al., 1999).

[13] The RH-TRU 72-B transportation cask received a Certificate of Compliance from the NRC in 2015 (expires in January 2025) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19324E132)

4 CURRENT AND POTENTIAL DESIGNS AS THEY AFFECT PROCESSING, STORAGE, AND TRANSPORTATION OF MSR SALT

This section focuses on four groups of designs[14] that represent a range of molten salts and molten salt operations that are relevant to this study [see World Nuclear Association (2021) for more discussion on these designs]. These groups, shown in Table 3, are 1) the fixed-fuel design in which the molten salt is contained in cylindrical metal tubes and the solid fuel design (also known as the salt-cooled design) in which the fuel may be maintained as TRISO graphite-matrix coated fuel particles; 2) the single-fluid, fluoride-salt design that operates in the thermal spectrum, 3) the single-fluid, chloride salt design that operates in the fast spectrum, and 4) the two-fluid breeder reactor design.

The intent of Table 3's organization is to capture MSR designs that have similarities. Typically, limited information was found on any one MSR, so organizing MSRs by group can yield insights on how similar designs may handle processing, storage, and transportation of MSR salt waste. Group 1 MSRs maintain the fuel separate from a primary coolant salt. Group 2 MSRs use fluoride fuel salt and extend from the MSRE experience.[15]. Group 3 MSRs use a chloride fuel salt. Group 4 MSRs have both a fluoride fuel salt and blanket salt.

Table 3. Groups of MSR Designs and Concepts

Group	Designs and Concepts	Reactor
1	Fixed-Fuel, Solid Fuel	SSR (Moltex), KP-X and Hermes Test Reactor [Kairos Power (KP)]
2	Single-Fluid, Fluoride Salt, Thermal Spectrum	MSRE, MSRR (ACU), TMSR-LF (SINAP), IMSR (Terrestrial Energy), FUJI (Thorium Tech Solutions), TMSR-500 (ThorCon), CMSR (Seaborg)
3	Single-Fluid, Chloride Salt, Fast Spectrum	MCFR (Terra Power), FC -MSR (Exodys Energy)
4	Two-Fluid, Breeder	Lithium Fluoride Thorium Reactor (LFTR) (Flibe)

No information was found for the Moltex SSR, KP-X , FUJI, and FC - MSR (Exodys Energy) reactors with regard to processing, storage, and transportation of the molten salt and these are therefore not discussed in this section.

The CNRS MSFR and the MOSART reactor from Table 2 are not included in this table. Little information was found for the CNRS MSFR concept, and its development is further in the future (i.e., after 2050) (IAEA, n.d.). Because insufficient information was found for the MOSART reactor to be classified into one of the groups, it is not discussed in this report.

[14] The term "designs" is used throughout this chapter. Many of these designs are in the conceptual stage and it is difficult to understand how far some of the designs have progressed. Instead of trying to distinguish between a design and a concept, the simpler term "designs" is used throughout.
[15] Although these designs use a fluoride fuel salt, many of them do not use the FLiBe salt that was part of the MSRE. ThorCon proposes a sodium-beryllium fluoride (BeF2-NaF) salt; the IMSR may use FLiBe or a sodium rubidium fluoride fuel salt instead (World Nuclear Association, 2021).

4.1 Processing the Molten Salt

This section describes three areas of molten salt processing. The first subsection summarizes the recovery of actinides and the removal of fission products from the molten salt. It covers processing the molten fuel salt while reactor operations are underway, as well as processing the molten fuel salt as a waste stream. The second subsection discusses electrochemical treatment of molten salt fuel waste. The third subsection describes tritium management. Holcomb et al. (2013) note that the potential for tritium release was cited as a major issue in the WASH-1222 report (U.S. Atomic Energy Commission, 1972), and this report was instrumental in the decision to discontinue the original U.S. Molten Salt Reactor Program (MSRP).[16] Therefore, tritium capture and management of tritium waste are important aspects to consider and address in MSR designs.

Much of the discussion in this section is focused on fluoride salt rather than chloride salt because limited information was found on chloride-salt MSR designs. Many of the fluoride-salt MSR designs use FLiBe salt of the form $2LiF_2$-BeF_2-XF_4. The "X" may be ^{235}U or ^{233}U for fuel salts, or ^{232}Th for the blanket salt (e.g., in an LFTR).[17] FLiBe is chemically stable over a broad temperature range and is unaffected by neutrons or radiation; therefore, it is common in MSR designs. However, tritium generation is significant in salts containing light elements lithium and beryllium. Also, during operations, fission products build up in the molten salt and the salt may need to be processed to remove these waste products. Used MSR fuel may also need to be processed to recover actinides that can be recycled for fuel. Processing the molten salt can reduce the total volume of waste that may need to be stored and can also reduce the amount of time it needs to be stored.

4.1.1 Recovery of Actinides and Removal of Fission Products

This section describes processing operations for MSRs in each of the four groups shown in Table 3. Continuous online processing is described for the LFTR (Group 4) and the MCFR[18] (Group 3), salt cleanup is described for the FHR (Group 1), and processing waste fuel salt on core changeout is described for the IMSR (Group 2). Continuous online processing is integral to the LFTR (Group 4), but may be less important for other MSRs, especially those in Group 1 in which the fuel salt is separate from the primary coolant salt. For example, the KP Hermes Test Reactor, which is an FHR in Group 1, does not use continuous online processing of the molten salt.[19] Also, the ACU MSRR in Group 2 does not use continuous online processing. Other MSRs in Groups 1 and 2 may require at least some salt cleanup to remove fission products.[20]

---

[16] The original Molten Salt Reactor Program (MSRP) was an MSR research program conducted at Oak Ridge National Laboratory. It included the Molten Salt Reactor Experiment (MSRE) in the 1960s and extended to research on the Molten Salt Breeder Reactor (MSBR) in the early 1970s.

[17] Some fuel salt designs differ from this formula. For example, the TMSR-LF1 uses a LiF-BeF_2-ZrF_4-UF_4-ThF_4 fuel salt (Wang, 2021). The FUJI uses a ThF_4-UF_4 fuel salt and FLiBe coolant (World Nuclear Association, 2021).

[18] This acronym is referring to the MCFR in general and not specifically to the Terra Power MCFR. When the Terra Power MCFR is discussed, "Terra Power" is explicitly added in the description (i.e., Terra Power MCFR).

[19] The Hermes Test Reactor does not use an online processing system or salt cleanup system for the FLiBe reactor coolant. Instead, coolant is withdrawn and replaced to restore it to specifications. KP (2021b) states that the FLiBe reactor coolant accumulates radionuclides from fission products that escape from defective layers of the TRISO fuel and transmutation products from FLiBe impurities including uranium.

[20] EPRI (2015) states that the FHR (i.e., a type of MSR in Group 1) is designed for a once-through fuel cycle with processing limited to salt cleanup.

Molten salt processing involves the recovery of actinides[21] and removal of fission products[22] from the molten salt. Processing may be performed continuously online during reactor operation to remove fission products that can interfere with reactor operation (Riley et al., 2018; Holcomb et al., 2011).[23] Processing may also occur in batch mode and may occur after reactor shutdown; for example, during core changeout and would also include immobilization processes for waste management. After an MSR is shut down, the molten fuel salt becomes a waste stream. Actinides can be recovered from this waste stream and used for makeup feed to a new core or another MSR. Recovering actinides permits the remaining shorter-lived fission product waste to be stored for a shorter duration.[24,25]

Online salt processing is often referred to as salt cleanup or salt conditioning. In some designs such as the LFTR, online salt processing is integral to normal operations and requires a small chemical processing facility co-located with the reactor (EPRI, 2015).[26,27] Within this chemical processing system, some fission products are removed from the fuel salt using a potassium hydroxide neutralization system. Other fission products are removed using reductive extraction. As described by EPRI (2015), when removed from the molten salt by the chemical processing system, the fission products exist in a metallic state in bismuth and need to be oxidized and placed in a disposal form before shipment from the site. EPRI (2015) states that small amounts of material are produced and that disposal plans would not constitute a major issue with reactor operations. The same report notes, however, that there are certain hazards associated with molten salt processing. First, processing fuel salt can result in hydrogen reacting with fluorine unless processes are designed to maintain hydrogen and fluorine separately.[28] Additionally, safely handling highly radioactive materials can be a challenge as these materials move from one fluid stream to another.

Salt cleanup is important not only for salt-fueled MSRs but also potentially for salt-cooled MSRs such as the FHR.[29] FHRs have a molten salt primary coolant in contact with solid fuel. Defective fuel can leak radionuclides into the primary salt coolant (Holcomb et al., 2013). Nongaseous radionuclides remain in the salt but can be removed using a salt cleanup system such as liquid

---

[21] Actinides are radioactive elements having atomic numbers 89 to 103 (actinium through lawrencium) and include naturally occurring uranium and thorium as well as synthetically produced plutonium.
[22] A primary emphasis in this section is the removal of fission products from the molten fuel salt. However, fission products can also be present in the primary coolant salt of an FHR if defective fuel is present. Therefore, salt cleanup is described for the FHR.
[23] As described by Holcomb et al. (2011) it is particularly important to remove fission products from the salt in thermal-spectrum systems because of parasitic neutron capture resulting from fission products with large capture cross sections. The cross sections are lower in the fast spectrum range.
[24] EPRI (2015, p. 3-18) lists requirements for LFTRs in Section 3.4.1. It states, "…developer discussion with utility customers indicates a strong desire to reduce the production rate of long-lived waste. Since actinides are considered problematic in terms of heat generation, radiotoxicity, and lifetime, an approach that reduces actinide waste to the maximum degree possible is desirable."
[25] One example is the LFTR, in which the waste is predominately fission products rather than actinides. Flibe Energy (2022) describes storing the remaining waste from an LFTR for only about 300 years to allow radioactivity to decay.
[26] For LFTRs, the blanket molten salt is also processed to obtain ^{233}U. The ^{233}U is then added to the primary fuel salt to replenish fissile material. EPRI (2015) states that the main function of the chemical processing system is to remove uranium and protactinium from the blanket salt and return uranium to the fuel salt. The secondary function is to remove fission products from the fuel salt and further process them.
[27] In addition to the LFTR, there are other designs that process the fuel salt at an on-site facility near the reactor during normal operations. For example, for the Molten Salt Fast Reactor (MSFR), Huer et al. (2014) describe fuel salt cleaning in which fission products are removed via batch processing of small fuel salt samples at an approximate rate of 10 to 40 L/day [2.6 to 10.6 gal/day].
[28] Hydrogen and fluorine react to form highly corrosive hydrogen fluoride. Hydrogen fluoride forms hydrofluoric acid when mixed with water.
[29] NRC (2020) describes the potential for needing a salt cleanup system in an FHR.

bismuth reductive extraction, which was demonstrated on a laboratory scale during the original MSRP. Holcomb et al. (2013) state that the salt cleanup technology for FHRs needs to be further developed and that it will likely have substantial technology differences from those previously demonstrated on a laboratory scale. Specifically, they state, "…in order to perform the cleanup operation on a side stream of primary coolant, a means to provide a controlled amount of primary salt into the cleanup facility and then return it to the reactor needs to be demonstrated" (Holcomb et al., 2013, p. 31).

Choe et al. (2018) propose an alternative reductive extraction technology that can be used to remove TRUs [i.e., neptunium (Np), plutonium (Pu), americium (Am), and curium (Cm)] [30] from the IMSR waste fuel salt. This is important because TRUs such as Pu dominate the waste fuel salt activity after about 300 years. Prior reductive extraction work involved liquid bismuth as a carrier metal with lithium as a reductant, but liquid bismuth is both toxic and corrosive. Instead, molten aluminum can be used as both carrier and reductant (Choe et al., 2018). In this process, the waste fuel salt is fluorinated to remove uranium. Afterwards the fuel salt is contacted by molten aluminum to extract TRUs. TRUs are removed from the molten aluminum, converted to fluorides, and then returned to the process as makeup feed. The waste stream would contain fission products and virtually no TRUs (Choe et al., 2018).

Similar waste minimization concepts exist for molten chloride fast reactors (MCFRs). Ottewitte (1992) describes not only removing actinides from the waste salt stream but also removing select long-lived fission products such as iodine. Iodine would be returned to the reactor for transmutation.[31] Continuous online processing of the molten fuel salt at an onsite facility would reduce the transportation of radioactive shipments (processing would occur onsite). Also, interim storage is not required for cool-down or preparation of the waste for shipment, and waste separated from the fuel would no longer pose a criticality concern (Ottewitte, 1992). Within the processing facility, the shorter-lived fission product waste stream can be further processed into an optimal form. It may be concentrated or diluted and "…further transformed into the most desirable chemical state, shape, size, or configuration to meet shipping and/or storage requirements" (Ottewitte, 1992, p. 8).

Although online processing may be important in some MCFRs, TerraPower's MCFR concept does not include online processing. Although online processing is not part of their design, TerraPower's MCFR includes mechanical filtering of insoluble fission products (Latkowski, 2021). This use of a simpler salt processing approach is supported by Holcomb et al. (2011). They state that fast-spectrum reactors can support simpler salt-processing approaches or a batch processing approach with relatively long salt-processing intervals. In comparison to fluoride salts, chloride salts have higher solubility for actinides, which allows the MCFR to accommodate a higher fuel loading and be better able to maintain criticality while fission products build up (Holcomb et al., 2011).

4.1.2 Electrochemical Treatment of Molten Salt Fuel Waste

Electrochemical treatment (i.e., electrorefining and pyroprocessing) is described by several authors (Arm et al., 2020; McFarlane et al, 2019; Riley et al., 2018; Peretz, 1996) as a way to conduct various separations on molten salt fuel waste. Separations that would be applied to a

[30] Transuranics (or TRUs) are elements having atomic numbers greater than uranium (i.e., greater than 92). TRU waste contains these elements with half-lives greater than twenty years in concentrations greater than 100 nanocuries per gram of waste (excluding HLW). "The WIPP Land Withdrawal Act specifically excludes high-level waste and spent nuclear fuel from the definition, as neither is allowed to be disposed of at the WIPP" (EPA, 2022).
[31] In this context, transmutation is referring to converting the element into a more stable, less harmful form.

specific MSR fuel salt waste would depend on the processing objectives which, in turn, would be influenced by the reactor design, the fuel cycle, and available disposal options. Peretz (1996) broadly describes the purpose of electrochemical treatment in the context of evaluating alternatives for processing MSRE waste as the extraction of actinides and fission products from the fuel/flush salt, concentrating the radioactive constituents in a small volume of high-level waste and leaving a large volume [fluoride] salt residue containing trace amounts of radioactivity. Electrorefining is well matched for processing a halide salt containing uranium, other actinides, and fission products, and would decontaminate the salt matrix sufficiently for it to be disposed of as low-level waste (Peretz, 1996). Other wastes include a zirconium metal waste, a chloride salt contaminated with plutonium and other radioactive materials, a waste form that incorporates plutonium and other materials in a bismuth ingot, and a metal waste incorporating the ^{90}Sr. Storage of the salt until ^{90}Sr has decayed to an acceptable level is suggested as a path to a low-level waste form.

In more recent descriptions of electrorefining in the context of preparing for potential future MSR projects, Arm et al. (2020) indicate the pyrochemical process has been amply demonstrated for chloride salt. They note that capture of fission products within zeolite ion exchange media has been demonstrated for the pyrochemical process, while they are separated as salts in the reductive separation process. In the latter case, they note a final waste form is likely available but needs to be explicitly identified. In evaluating the potential hazards associated with MSR processing, McFarlane et al. (2019) describe that the collected materials must be stored appropriately for possible recycling (in the case of the carrier salts) or waste disposal (fission products). They further caution that electrochemical processing involves working with spent fuel (and the associated very high radiation levels); therefore, such operations must be performed remotely (McFarlane et al. 2019).

4.1.3 Tritium Capture

Holcomb et al. (2013) describe technology challenges facing FHRs; however, aspects of their study related to FHR salts are applicable to other MSRs as well. For example, they recognize the need for tritium control and tritium capture technologies. Tritium is formed through interaction of neutrons with lithium and beryllium in the salt coolant.[32] Because MSRs produce significantly more tritium than LWRs (Holcomb et al., 2013; EPRI, 2015[33]; U.S. Atomic Energy Commission, 1972[34]; Grabaskas et al., 2020[35]), there is a need for tritium capture technology and for managing the tritium waste.

Because tritium is light, it may diffuse through the walls of piping and other components such as heat exchangers and, if not captured, could be released to the environment as tritiated water. WASH-1222 (U.S. Atomic Energy Commission, 1972) and Grabaskas et al. (2020) identify various options for tritium control such as using coatings on metal surfaces to inhibit diffusion through piping or heat exchangers, or using a double-walled heat exchanger having an yttrium

[32] FLiBe salt in a nuclear reactor will produce tritium via neutron absorption in ^{6}Li (NRC, 2020).
[33] EPRI (2015) states that tritium generation is an inevitable consequence of the presence of lithium and beryllium in the salt mixture.
[34] WASH-1222 (U.S. Atomic Energy Commission, 1972) compares tritium production in the proposed MSBR to that of other reactors. The MSBR concept was developed during the original Molten Salt Reactor Program (MSRP) at Oak Ridge National Laboratory (ORNL) during the 1970s and followed the MSRE work conducted earlier in the 1960s. WASH-1222 states that tritium would be produced in an MSBR at the rate of about 2,400 Ci/day in a 1000 MWe reactor. The rate would be about 40 to 50 Ci/day in a light water, gas-cooled, or fast-breeder reactor.
[35] The use of a lithium- or beryllium-containing salt could potentially cause large tritium generation rates in MSRs (Grabaskas et al., 2020).

getter layer to capture the tritium.[36] Holcomb et al. (2013) describe the development of an effective tritium containment strategy as a leading task to enable the safe operation of FHRs.

The development of an effective tritium containment strategy may also be important for other MSR designs that use a lithium-beryllium salt. The MSRs in Table 3 operating in the thermal spectrum and using a lithium-beryllium salt are the KP-X and Hermes Test Reactor (Group 1), the MSRR and TMSR-LF (Group 2), and the LFTR (Group 4). For fast spectrum reactors such as the MCFR in Group 3, tritium generation is much lower and therefore tritium containment may be less of a concern. Ottewitte (1992) compared the tritium generation from an MSBR[37] to that of an MCFR and concluded the MCFR produces orders of magnitude less tritium.

MSR vendors have made little information publicly available on their plans for handling tritium waste once it is captured and removed from an MSR[38]; however, Grabaskas et al. (2020) suggest it may be stored onsite or transported to a different facility for processing and storage. One area of experience with tritium storage comes from processing tritium waste generated from CANDU heavy water reactors (HWRs). For these reactors, tritium is processed into a metal hydride (or tritide). Similarly, Grabaskas et al. (2020) suggest that MSR operators could process the tritium waste into a metal hydride for storage. They state that titanium (Ti) is a popular choice for the metal and there is experience at the Savannah River Site storing tritium as titanium tritide (TiT_2).

4.2 Storage of MSR Salt Waste

Much of the experience with storage of MSR salt waste comes from the MSRE. After the reactor shutdown in 1969, the fuel salt was drained into two tanks. Several years later, uranium was removed from the fuel salt, but the residual fuel salt, contaminated with fission products, remains today in the drain tanks. Also, after shutdown, a flush salt was run through the reactor and piping and then drained into a tank where it remains. Some challenges occurred while storing the salt waste in these drain tanks:

2. The residual fuel salt constantly generates fluorine gas from radiolysis (McMillan, 2019).

    1. After several years, ORNL experienced challenges transferring the salt waste to other containers. ORNL experienced drain line clogging, possibly from the buildup of corrosion products, and postponed the effort to transfer the salt (ORSSAB, 2010).

    2. A clear path for final disposition of the salt waste has not been identified after storing the salt waste for decades, leading to a continuation of storage. Peretz (1996) describes disposition alternatives, which are summarized in Section 3.1. McMillan

[36] A getter is a chemical trap. The double-walled heat exchanger prevents contact between the primary and secondary fluids and has a central section containing the "getter," a rare earth element such as yttrium to capture or trap the tritium. As described by Holcomb et al. (2013), rare earth elements such as yttrium form high-temperature, stable hydrides that trap tritium. Grabaskas et al. (2020) also describe other double-walled heat exchanger designs using a purge fluid instead of a getter.

[37] The MSBR is a thermal spectrum reactor concept with a molten salt fuel including UF_4 and ThF_4 dissolved in $LiF – BeF_2$ [See WASH-1222 (U.S. Atomic Energy Commission, 1972) for more information].

[38] KP (2021b) includes some information on tritium management for the Hermes Test Reactor. The Tritium Management System is used to capture and dispose tritium. As stated by KP (2021b), following service, tritium capture materials are stored in sealed canisters. They would be shipped from the site in packages meeting Department of Transportation regulations. KP intends to use Type A and Type B packaging canisters in accordance with 10 CFR 71.51.

(2019) provides a more recent status update including reconsideration of *in-situ* decommissioning (entombment) as an option under study.

Various sources describe fuel salt storage in general terms but do not specifically identify solutions to problems encountered from the MSRE. For example, World Nuclear Association (2021) states the IMSR sealed core is replaced after seven years and left for fission products to decay, but there is no discussion on the potential for fluorine gas generation or how it would be managed. Additionally, there is no discussion of tritium waste management.[39]

The following subsections compile information on salt waste storage for individual MSR designs. Many MSRs listed in Table 3 are in the early design stage, so information was not available on salt waste storage and the management of stored salt waste. Also, specifics on the management of salt waste are not provided because information related to the waste form composition[40] could not be found.

4.2.1 Group 1 MSRs

The Group 1 MSRs are characterized by a primary coolant salt separated from the fuel. The primary coolant salt may contain radionuclides from damaged fuel, but the concentration would be much lower than that in the molten fuel salts of the other groups. The radioactivity is expected to be lower in this coolant salt, and therefore, storage of the primary coolant salt would not have the same level of complexity expected for fuel salts and experienced during storage of the MSRE fuel salt.

4.2.1.1 Kairos Power Solid-Fueled Reactor

NRC accepted for review in November 2021 the construction permit application for the Hermes test reactor, which Kairos Power has proposed to build at the East Tennessee Technology Park Heritage Center site in Oak Ridge and begin operating by 2026 (KP, 2022). The NRC staff completed its review of the environmental report for the Hermes test reactor (KP, 2021a) and prepared a draft environmental impact statement for comment (NRC, 2022). NRC (2022) states that at the end of operations, FLiBe would be collected in storage containers and will solidify as the salts cool with a low radionuclide gamma activity such that radiation decomposition of the FLiBe would not be of concern during long-term storage. NRC (2022) also notes that some trace amounts of tritium could diffuse out of the storage containers, but radiation monitoring equipment would be in place to ensure safe storage until the waste material is removed from the site or placed in a certified dry cask storage system, if necessary. The solidified FLiBe could be classified as Class C low-level radioactive waste (LLRW) due to the presence of ^{14}C at greater than 0.8 Ci/m^3 but less than 8.0 Ci/m^3 along with other radionuclides controlled by the technical specifications provided in Table 14.1-1 of the preliminary safety analysis report (KP, 2021b).

---

[39] The IMSR uses fluoride carrier salts (World Nuclear Association, 2021; Choe et al., 2018). Initially, no Li or Be will be used in the carrier salt; however, World Nuclear Association (2021) states the carrier salt may change to FLiBe, requiring tritium management.

[40] Riley et al. (2018) characterizes the salt waste in general, stating that after irradiation of actinide-bearing fluoride/chloride salts used as coolants or fuels, the salts will contain fission products, TRUs, and the fuel component (e.g., uranium or thorium).

4.2.2 Group 2 MSRs

Group 2 MSRs build on the experience from the original MSRP with thermal spectrum fluoride-salt MSR designs. For these MSRs no information was found on how they would address (or whether they would need to address) the problems experienced with MSRE fuel salt storage described at the beginning of Section 4.2. Like the MSRE, in some of the MSR designs, such as the MSRR, the salt is not processed prior to storage in tanks onsite. Other MSR designs, such as the IMSR, may include processing to recover TRUs from the salt waste when the reactor core is replaced. This processing would leave shorter-lived fission products as a salt waste stream. Other MSR designs such as the TMSR-LF would have batch or continuous online processing to remove thorium and uranium. This processing would leave a waste salt stream of minor actinides[41] and fission products for storage. Note, however, the minor actinides may be processed in the future in a fast spectrum TMSR-LF.

4.2.2.1 Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR)

The proposed ACU MSRR conceptual design (ACU, 2022) incorporates a fuel handling system. As depicted by ACU, the fuel handling system would be capable of handling and storing used salt in a separate area of the facility outside the reactor enclosure and would incorporate safety features to address containment, criticality, and direct radiation (e.g., shielding). The fuel handling system includes storage tanks connected to a fuel receiving area (the entry point for fresh fuel shipments) and the reactor fuel drain tank (which facilitates fuel loading and removal from the reactor vessel). The MSRR is designed to reach shutdown through passive draining of fuel salt into a subcritical configuration within the reactor drain tank below the reactor vessel. ACU proposes transfer technology utilizing pressurized inert gas and heated transfer lines between vessels and is considering incorporating salt freeze plugs and pressure controls to minimize mechanical valves. The fuel handling system design will be finalized in the ACU operating license application. ACU is working with DOE to obtain fuel cycle services for the MSRR similar to other university research reactors.

Long-term and temporary storage of solid radioactive waste is expected by ACU until appropriate disposal arrangements are implemented. ACU is proposing an approximate operations period that ends in 2042. Under this proposal, used fuel would not be removed from reactor coolant salts and, therefore, would not be stored separately. The entire reactor coolant inventory, including fuel, would be returned to the DOE during decommissioning. The ACU proposal is not reliant on any further processing of the used fuel salt.

4.2.2.2 Compact Molten Salt Reactor (CMSR)

As stated by Seaborg Technologies (2022), short-lived fission products are separated from the fuel salt and sent to storage; the remaining fuel salt is mixed in with new fuel. There are no details on this waste stream, but Seaborg Technologies states that it is radiologically similar to radioactive hospital waste and that it can be handled by conventional methods. World Nuclear Association (2021) states that fission products are extracted online, but no details are provided.

4.2.2.3 Integral Molten Salt Reactor (IMSR)

Choe et al. (2018) describe various options for managing molten fuel salt waste for the IMSR. The IMSR has a seven-year design life for each core unit. For a once-through fuel cycle, the

[41] Minor actinides are actinide elements other than the major actinides uranium and plutonium.

entire fuel salt volume may be removed and sent to a nearby salt storage vault when the core unit is replaced; however, this results in a substantial volume of fuel salt accumulating over an approximately 60-year facility lifetime. This volume can be reduced if some of the removed fuel salt is redirected to the next core unit or to additional IMSRs.[42] However, reusing molten fuel salt will cause fission products to build up.[43]

4.2.2.4 Thorium Molten Salt Reactor – Liquid Fueled (TMSR-LF)

The SINAP TMSR liquid-fueled reactors (or TMSR-LFs) are currently under development with the TMSR-LF1 having been constructed and receiving commissioning approval in China (Nuclear Engineering International, 2022a). Although information is not available on the characteristics of the salt waste and any requirements for its storage, Wang (2021) states that initially all the fuel salt would be removed after five to eight years and processed to remove fission products and minor actinides for storage. The TMSR-LF development would then move from this batch processing to continuous processing of the molten fuel salt, in which uranium and thorium would be recovered and minor actinides and fission products would be separated. World Nuclear Association (2021) describes the TMSR-LF development as a fully closed thorium-uranium fuel cycle with the future development of a TMSR-LF fast reactor optimized for burning minor actinides.

4.2.2.5 ThorCon MSR

IAEA (2020) includes some discussion on ThorCon's plans for storing salt waste. Regular maintenance occurs at four-year intervals. A service ship (or CanShip) visits to exchange fuel casks and Cans.[44] The Can is removed after four years, but the fuel salt is reused for another four years. After eight years, the fuel salt is spent and is transferred to a fuel cask in a vault where it can be stored indefinitely[45] with passive airflow cooling (IAEA, 2020).

IAEA (2020) provides a comparison of waste generated by the ThorCon MSR to that of the decommissioned Connecticut Yankee boiling water reactor from generating 450 MWe for 28 years. IAEA (2020, p. 30) concluded that "Generating this much energy with ThorCon would not even fill two of 12 tanks in the 5 m long vault inside the hull."

4.2.3 Group 3 MSRs

Group 3 MSRs use a chloride molten fuel salt. No information was found on the composition of the waste salt stream. TerraPower does not plan to use continuous online processing for its MCFR, but it may recover ^{37}Cl and is evaluating some options for storing the waste salt, either directly or in some converted form (e.g., glass or SynRoc). Other future MCFRs may use continuous online processing to recover actinides and long-lived fission products for fuel, leaving shorter-lived fission products as a waste stream for storage (Ottewitte, 1992; Holcomb et al., 2011).

---

[42] Choe et al. (2018) describe this concept as a Feed, Seed, and Breed (FSB) fuel cycle.
[43] Choe et al. (2018) state that fission products will build up in the molten fuel salt during the first few core cycles before reaching an equilibrium.
[44] A Can contains the reactor, the primary loop heat exchanger, and the primary loop pump (IAEA, 2020).
[45] IAEA (2020) also states the spent fuel salt is stored for the plant lifetime (80 years) in the vault (a 5-meter section at the end of the hull).

4.2.3.1 TerraPower Molten Chloride Fast Reactor (MCFR)

TerraPower is early in the development stage of the MCFR. In partnership with Southern Company, it has recently built an Integrated Effects Test (IET) to develop its MCFR design and help scale its molten chloride technology to commercial-sized reactors (TerraPower, 2022). In addition, with Southern Company it is developing the Molten Chloride Reactor Experiment (MCRE) at INL as an MCFR demonstration reactor. Latkowski (2021) describes some options for molten chloride salt waste including direct disposal in a salt repository with no ^{37}Cl recovery, conversion of the waste to possibly an iron-phosphate glass (i.e., vitrification), or recovering ^{37}Cl and incorporating metal oxides into SynRoc. No details were provided on any of these three options relating to the composition of the salt waste stream, packaging, or storage requirements.

4.2.3.2 Fast Spectrum Reactors

Although details on salt storage were not available for TerraPower's MCFR design, Holcomb et al. (2011) shed some light on the fast spectrum MSR waste stream. As described previously, reductive extraction would be used to separate fission products from the fuel salt in a fluoride-salt reactor. In contrast, Holcomb et al. (2011) state that chloride salt reactors may employ electrochemical separation, zeolite ion-exchange capture, and chloride volatility processing. In either case, for fast spectrum reactors, longer-lived fission products can be returned to the fuel salt. As stated by Holcomb et al. (2011) this leaves a relatively small volume of separated fission products that can be left in salt form and allowed to solidify and decay in short-term storage.

4.2.4 Group 4 MSRs

Group 4 MSRs are two-fluid breeders having a fuel salt and blanket salt. The Flibe Energy LFTR is currently under development and uses continuous online processing, which is integral to these designs.

4.2.4.1 Lithium Fluoride Thorium Reactor (LFTR)

No details were found on the composition of the waste product streams; however, Flibe Energy (2022) states that waste products from salt processing are predominantly fission products rather than actinides. The fission products decay more rapidly than actinides so having fewer long-lived waste products can result in less waste during long term storage or disposal timeframes. Flibe Energy (2022) stated that only 17 percent of waste products require long-term storage and estimated that after about 300 years the radiological hazard of fission products is substantially reduced by radioactive decay. No information was found on management of the salt waste, including how the waste would be stored.

4.3 Transportation of MSR Salt Waste

The evaluation of MSRE fuel salt waste disposition alternatives (Peretz, 1996) provides concept and design information related to MSR fuel salt waste transportation. To allow for possible additional processing, handling, and transportation, ORNL proposed filling an inner Hastelloy-N can with molten salt. The can would be placed inside a shielded container, and three shielded waste containers could then be placed inside an RH-TRU canister.[46] ORNL presumed an RH-TRU 72-B transportation cask could be used for transportation of the loaded RH-TRU canisters but did not conduct a detailed evaluation nor address whether the proposed approach would meet package specifications.

For transportation of fluoride salt containing the gamma source from cesium or the ^{232}U daughter chain, ORNL (Peretz, 1996) noted the evolution of radiolytic fluorine must be controlled. ORNL proposed a getter, such as soda-lime, to be used for this purpose. To facilitate a getter system, each salt can would be filled with salt to 75% of its internal volume. The remaining internal volume would be used to vent the fluorine gas through the getter and a HEPA filter. They noted the quantity of fluorine gas could be calculated and verified, thereby allowing the amount of getter to be determined. ORNL indicated additional effort was needed to determine whether such a transportation package could be certified.

More recent information on transportation of MSR salt waste is from test reactors that are currently under development. Although test reactors are further along in the development process, no details were found on transportation cask design or management of fluorine evolution during transportation. For many of the other MSRs, no information could be found on transportation of salt waste, but a few of them included some general descriptions. For example, for the IMSR, World Nuclear Association (2021) states the sealed core unit is replaceable. After removing it, it is allowed to cool and then it is removed for offsite processing. There is no discussion on the transportation cask planned or the destination for the waste salt. Similarly, for the ThorCon MSR, World Nuclear Association (2021) states the entire primary loop would be changed out every four years and returned to a centralized recycling facility.[47] Specifics on how this would be accomplished are not provided.[48] For the Seaborg Technologies CMSR, after a 12-year cycle, the fuel would be returned to the supplier where short-lived fission products would be separated and sent to storage (Seaborg Technologies, 2022). There is no discussion on how the fuel salt would be transported or where the short-lived fission products would be sent and how this would be accomplished. ThorCon and Seaborg Technologies plan to develop the MSR as a power system contained within a barge. Therefore, plans for transporting MSR waste may vary depending on the location of the barge and the specific safety requirements associated with that location.

The following subsections describe information on transportation plans that is available for two test reactors that are currently under development.

[46] More specifically and as stated by Peretz (1996), the can would have 1.27 cm [0.5 in] thick walls to allow for corrosion if uranium is separated from the salt by fluorination in the can. The can would then be inserted into a shielded container with the approximate outer dimensions of a “55-gallon” drum. This container would provide 5.1 cm [2.0 in] of steel in each direction as a radiation shield. ORNL expected three of these shielded salt waste containers could be placed into an RH-TRU canister. An RH-TRU canister is made of 0.64 cm [0.25-in] thick carbon steel and is 307 cm [121 in] long and 66 cm [26 in] in diameter.
[47] Cans (i.e., the container for the reactor, primary loop heat exchanger, and primary loop pump) are changed every four years and removed after a four-year cooldown period; fuel salt is replaced every eight years (IAEA, 2020).
[48] However, IAEA (2020) includes some information related to management of the spent fuel salt. IAEA (2020) states the spent fuel salt would be held in casks inside a vault. It may be transferred at a later time to a special transfer cask, removed by crane and offloaded to a service ship (i.e., a CanShip) where it would be taken for processing or dry cask storage. No specifics were provided on the types of casks that are planned.

4.3.1 Kairos Power Solid-Fuel Reactor

The used solidified FLiBe primary coolant salt generated at the Hermes test reactor facility would be stored onsite until decommissioning and then disposed as solid radioactive waste. Kairos Power plans to use existing certified packages to transport the solid FLiBe LLRW to Waste Control Specialists in Texas by truck (KP, 2021a).

4.3.2 Molten Salt Research Reactor (MSRR)

The ACU construction permit application (ACU, 2022) provides only limited information about transportation[49] of fuel salt waste, but indicates that waste systems would be operated in accordance with procedures so that the final waste form would be acceptable for transport in Department of Transportation or NRC-certified shipping containers.

5 INSIGHTS

Storage and transportation of MSR fuel waste are affected by the final disposition option and by the broader approach to the fuel cycle for MSRs. Available information suggests that under the existing LWR waste management statutory and regulatory framework, MSR fuel waste would likely be destined for disposal in a federal high-level radioactive waste repository and waste would likely require further onsite or offsite processing to meet disposal requirements. Also, because some MSRs can burn plutonium and minor actinides, they can both recycle fuel materials and potentially reduce the longevity of the hazard in final waste forms destined for a repository (Orano, 2022; Forsburg et al., 2001; McKay, 1981).

The final disposition of existing MSR salt waste, as well as existing LWR spent fuel, has been hampered, in part, by delays in licensing a federal high-level radioactive waste repository. This, in turn, has resulted in the need for extended storage and deferred transportation. The MSRE experience illustrates the need to factor storage, transportation, and disposal considerations into proposed designs and operational practice so that produced salt waste is in a form and packaging that facilitates post-shutdown disposition of the material while MSR structures, systems, and components needed for salt waste removal are still operational.

The following insights were developed consistent with the MSRE experience after reviewing available information on MSR designs.

3. Plans for co-located chemical processing facilities: Some MSR designs may have a salt cleanup system co-located with the reactor to remove fission products from the fuel salt or primary coolant during normal operations.[50] Some waste management strategies include the recovery of actinides and separation of fission products in a co-located chemical processing facility. Information is needed on the composition of the waste stream, the amount of waste produced, plans for storing or transporting the waste, and any further onsite or offsite processing needed.

    1. Tritium management strategies: Many of the MSRs would generate significantly more tritium than LWRs (particularly those operating in the thermal region and using a lithium-beryllium salt), and therefore an effective tritium containment, processing,

[49] Part of the reason information is limited may be because ACU is working with DOE to obtain fuel cycle services for the MSRR, similar to other university research reactors.

[50] An example of an MSR design that does not require online processing is the ThorCon MSR. As described by World Nuclear Association (2021), processing takes place in a centralized plant at the end of core life.

and disposal strategy is needed. Information is needed on the expected quantity of tritium produced, capabilities for capturing it, and methods for storing and disposing of the tritium waste. Very little information was found on tritium management strategies.

2. Plans for transportation of MSR salt waste: The MSR designs would require transportation of salt waste. Fuel salt waste may contain fission products and actinides. Plans for transporting the salt waste need to be integrated into the design of the MSR because there needs to be a way to handle and package the waste for shipment. If the intent is to transport the waste as part of a sealed core unit, then the means for doing so need to be developed as that core unit is being designed.

3. Characterization of salt waste streams: Salt waste streams need to be characterized for composition and any potential hazards during processing, storage, and transportation. This may include the potential for fluorine generation. Continuous generation of fluorine was identified as a problem during storage of the waste salt from the MSRE.

4. Long-term storage of MSR salt waste: In some of the MSR designs, the salt waste may be stored for several years (e.g., until decommissioning or until final disposition). The MSRE revealed challenges with storing the waste for decades. Information is needed on how the salt waste would be safely stored to ensure the operational integrity of equipment used to manage (i.e., contain and transfer) the stored waste.[51] Also, concerns such as the potential for fluorine gas generation need to be addressed.

5. Due to the solubility of MSR salt waste, there may be a need to immobilize the salt in a waste form such as glass or ceramic. Due to the different salts being proposed for use, it would be difficult to immobilize all salt types in a single waste form.

6 SUMMARY AND CONCLUSIONS

There continues to be extensive MSR research and development, but operational experience is limited. The MSRE was conducted decades ago and it remains the primary source of experience, having demonstrated some unique challenges with MSR salt storage, such as: the production of fluorine gas from irradiation of solidified fluoride-based fuel salts, the potential remobilization of uranium in off-gas control systems, and challenges associated with aging structures and equipment under long-term storage conditions.

Several companies are developing MSRs; most designs are liquid-fueled designs in which the fuel is dissolved in the molten salt and the salt serves as both the fuel and the primary coolant. Many of the designs are preliminary and the open literature describes the layout of components as well as the type of fuel and salt, but does not typically characterize the salt waste nor provide detail on storing, transporting, or processing the MSR salt waste. Salt waste varies with MSR design. We analyzed the current and potential MSR designs based on industry trends to attempt to characterize salt waste streams and help identify potential challenges associated with its safe

[51] Note the technology for managing molten chloride salts is not as advanced as that for fluoride salts. Holcomb et al. (2011, p. xii) stated "…the corrosion processes for chlorine are more complex than those for fluorine. Consequently, the knowledge base for structural materials tolerant of chloride-based salts is not as mature as that for fluoride-based salts. A confident structural material selection cannot yet be performed for a chloride salt-based FS-MSR."

storage, transportation, and processing. Available information on each design discussed in this report is summarized in Table 4.

Processing molten fuel salt achieves two primary objectives: 1) removing fission products that degrade the operation of the reactor and 2) separating long-lived radioactive material from the salt waste. Processing may be performed continuously online, or it may be performed in batch mode, for example, on a waste stream after the fuel salt is drained from the reactor. It may be a more complex process to recover elements for reuse in the fuel, or it may be a much simpler process focused more on salt cleanup.

Continuous online processing is intended for some thermal spectrum, fluoride-salt, thorium MSRs such as those in Group 2 (i.e., TMSR-LF) and Group 4 (i.e., LFTR). For example, continuous online processing is integral to the LFTR, and SINAP is developing plans for continuous online processing for their TMSR-LFs.[52] For other MSRs, such as TerraPower's MCFR in Group 3, only salt cleanup is envisioned with mechanical filtering of insoluble fission products. Online processing or salt cleanup may be envisioned for other MSRs in Groups 1 and 2, but no information was found on the extent to which these operations are needed.

Limited non-proprietary information was found on the composition of MSR salt waste and plans for storing this waste. In some designs such as the Group 2 MSRs, the vendor plans to store salt waste onsite indefinitely or until reactor decommissioning. The Group 2 MSRs are similar to the MSRE, and problems were encountered with storing MSRE waste for decades. No information was found on how similar problems could be avoided with the new MSR designs.

Limited publicly available information was also found on transporting MSR waste. The most recent information comes from two test reactors currently under development, but no details were available -- only high-level plans. For example, ACU (2022) states the MSRR salt waste will be transported in an acceptable form using Department of Transportation or NRC-certified shipping containers. A prior ORNL evaluation of MSRE salt waste disposition alternatives (Peretz, 1996) provides more detailed concept and design information related to transportation. That evaluation describes how salt waste could be transferred to a shielded container and how these containers could be placed inside a canister that would fit into an existing transportation cask. It also describes how fluorine gas could be addressed in the shielded container design. ORNL did not conduct a detailed evaluation nor address whether the proposed approach would meet package specifications.

There was also sparse publicly available information on development of waste forms outside of DOE research and development. Because MSR salt wastes are soluble, it is likely they will need to be immobilized in waste forms such as glasses and ceramics, either at the reactor site or elsewhere. It would also be difficult to immobilize the entire chloride-based or fluoride-based salt in a single waste form that meets disposal restrictions for an HLW repository. Depending on the treatment operations, a portion of the MSR fuel salt may qualify as low-level radioactive waste

[52] The ThorCon MSR and Seaborg Technologies CMSR are also thermal-spectrum, fluoride-salt, thorium MSRs, but limited information was found for these MSRs on salt cleanup or salt processing that would be needed during normal operations. World Nuclear Association (2021) states that fission products are extracted online for the Seaborg Technologies CMSR. IAEA (2020) states that the noble gas fission product ^{135}Xe is removed during normal operation of the ThorCon MSR to decrease neutron absorption while World Nuclear Association (2021) states that no online processing occurs for this MSR. No other information was found that describes the need for online processing or cleanup of the molten salt. This may reflect that the ThorCon MSR and Seaborg Technologies CMSR are in earlier stages of development compared to the TMSR-LF.

(LLW) based on the radionuclide concentrations in the salt waste stream. This could expand the options for final disposition outside of a deep-mined geologic repository (Kitcher, 2020). This could reduce reliance on deep geologic repositories.

In summary, several MSR designs were reviewed. In some cases, high-level plans for processing, storing, and transporting MSR waste were found, but specific details were not made available. Research has continued for decades on MSRs and some of this research is included in this report. The most recent application of this research is with two test reactors that are currently under development, and they are both discussed in this report. Very limited information is publicly available even for these reactors on processing, storing, and transporting MSR salt waste, including other potential waste forms. Insights from industry trends are presented in this report. From these insights, early communication between industry and the NRC should be encouraged to support more efficient reviews of applications. The evaluation of current industry trends to characterize MSR salt waste streams and identify potential challenges associated with the safe storage, transportation, and processing of MSR salt waste can help to determine information needs in the context of existing regulatory frameworks and guidance. To prepare for design analysis of MSR designs and supporting technologies, the NRC should also consider safety and risk considerations of relevant phenomena where knowledge gaps may exist.

Table 4. Summary of Designs as they Affect Processing, Storage and Transportation of MSR Salt Waste			
MSR	**Processing**	**Storage**	**Transportation**
	Group 1		
KP-X and Hermes Test Reactor	Hermes Test Reactor: FLiBe reactor coolant salt is removed and replaced as needed to maintain technical specifications of the salt. KP-X: No Information	Hermes Test Reactor: FLiBe salt waste is stored onsite in storage containers until decommissioning. KP-X: No Information	Hermes Test Reactor: Solid FLiBe waste is planned for transport to Waste Control Specialists by truck. KP-X: No Information
Moltex SSR	No Information	No Information	No Information[53]
	Group 2		
CMSR	Limited Information – Fuel salt is processed at the end of a 12-year cycle to remove short-lived fission products for storage. Fission products may also be extracted online.	No Information	No Information – Fuel salt is returned to the supplier but no details are provided on how the salt would be transported.
FUJI	No Information	No Information	No Information
IMSR	No Information – A possibility is to process the waste stream to recover TRUs.	No Information – A possibility is to discharge the entire salt volume to a nearby salt vault. Another possibility is to reuse some of the salt in new cores.	No Information
MSRE	Removed uranium from the stored fuel salt after several years of storage.	Stored in tanks for decades.	Transportation offsite has not occurred but an ORNL alternatives analysis (Peretz, 1996) describes an approach to transfer fuel salt

[53] Scott (2016) describes the design as including a cartridge-like core module (containing fuel assemblies) that would be road transportable. It is not clear if the core would be transported as a unit at the end of core life or if individual assemblies would be removed, repackaged, and transported.

Table 4. Summary of Designs as they Affect Processing, Storage and Transportation of MSR Salt Waste			
MSR	**Processing**	**Storage**	**Transportation**
			into cans that would fit into canisters that they presumed could be shipped in existing transportation casks.
ACU MSRR	No processing used fuel salt was proposed. ACU expects to reach an agreement with DOE to take the used fuel salt.	Molten fuel salt maintained onsite in storage tanks until reactor decommissioning.	Limited Information – Final waste form would be acceptable for transport in Department of Transportation or NRC-certified shipping containers.
SINAP TMSR-LF	Initially use batch processing of the molten fuel salt but subsequently plan to develop continuous online processing.	Limited Information – Minor actinides and fission products are separated out for storage.	No Information
ThorCon MSR	No Information – At the end of an eight-year cycle, uranium may be recovered from the spent fuel salt.	Molten fuel salt would be stored indefinitely in casks in a vault on the barge.	No Information – Possible future removal of the spent fuel salt via a CanShip for processing or dry cask storage.
Group 3			
TerraPower MCFR	No online processing but would include mechanical filtering of insoluble fission products.	No Information - Options are being evaluated to include the recovery of ^{37}Cl with the possible use of vitrification, SynRoc, or direct disposal in a salt repository.	No Information
Exodys Energy FC - MSR	No Information	No Information	No Information

Table 4. Summary of Designs as they Affect Processing, Storage and Transportation of MSR Salt Waste			
MSR	**Processing**	**Storage**	**Transportation**
Group 4			
Flibe LFTR	Continuous online processing to recover ^{233}U and remove fission products.	No Information	No Information

7 REFERENCES

Abelquist, E. and T. Morgan. “Decommissioning Challenges at the Molten Salt Reactor Experiment Site.” Presentation at the 2021 Virtual Molten Salt Reactor Workshop, October 12, 2021. Oak Ridge, Tennessee: United Cleanup Oak Ridge, LLC. 2021. <https://msrworkshop.ornl.gov/wp-content/uploads/2021/11/09_MSREMoltenSaltReactorWorkshopPresentationOctober202127SEP2021.pdf> (Accessed 9 October 2022).

ACU. “Abilene Christian University Molten Salt Research Reactor Preliminary Safety Analysis Report.” Rev 0. ADAMS Accession No. ML22227A203. Abilene, Texas: Abilene Christian University NEXT Lab. 2022.

Arm, S.T., D.E. Holcomb, R.L. Howard, and B. Riley. “Status of Fast Spectrum Molten Salt Reactor Waste Management Practice.” PNNL-30739. Richland, Washington: Pacific Northwest National Laboratory. December 2020. <https://www.pnnl.gov/main/publications/external/technical_reports/PNNL-30739.pdf> (Accessed 5 January 2023).

Boussier, H., S. Delpech, V. Ghetta, D. Heuer, D.E. Holcomb, V. Ignatiev, E. Merle-Lucotte, and J. Serp. “The Molten Salt Reactor (MSR) in Generation IV: Overview and Perspectives.” GIF Symposium – San Diego (California). November 14-15, 2012. <https://www.researchgate.net/publication/275020090_The_Molten_Salt_Reactor_in_Generation_IV_Overview_and_Perspectives> (Accessed 4 October 2022).

Choe, J., M. Ivanova, D. LeBlanc, S. Mohaptra, and R. Robinson. “Fuel Cycle Flexibility of Terrestrial Energy’s Integral Molten Salt Reactor (IMSR).” 38th Annual Conference of the Canadian Nuclear Society and 42nd Annual CNS/CNA Student Conference. June 3-6, 2018. <https://www.terrestrialenergy.com/wp-content/uploads/2018/09/TerrestrialEnergyPaperCNS2018PDF.pdf> (Accessed 21 September 2022).

DOE. “Southern Company and TerraPower Prep for Testing on Molten Salt Reactor.” Office of Nuclear Energy. November 29, 2021. <https://www.energy.gov/ne/articles/southern-company-and-terrapower-prep-testing-molten-salt-reactor> (Accessed 22 September 2022).

DOE. “Environmental Assessment: Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at Argonne National Laboratory – West.” DOE/EA-1148. Washington DC: U.S. Department of Energy. 1996. <https://www.energy.gov/nepa/downloads/ea-1148-final-environmental-assessment> (Accessed 9 October 2022).

DOE. “Elysium Industries/Idaho National Laboratory (INL)/Argonne National Laboratory (ANL) - RFA-17-14592, Synthesis of Molten Chloride Salt Fast Reactor Fuel Salt from Spent Nuclear Fuel.” n.d. <https://gain.inl.gov/SiteAssets/2017%20Voucher%20Abstracts/RFA-17-14592%20ELYSIUM.pdf#search=elysium> (Accessed 5 October 2022).

DOE. “GAIN announces second round FY 2022 Nuclear Energy Voucher recipients” <https://gain.inl.gov/SiteAssets/2022VoucherAbstracts-2ndRound/Abstracts/ElysiumTechnicalAbstract_FINAL_22-2.pdf> (Accessed 1 February 2023)

EPA. “What is Transuranic Radioactive Waste?” 2022. <https://www.epa.gov/radiation/what-transuranic-radioactive-waste> (Accessed 17 December 2022).

EPRI. “Program on Technology Innovation: Technology Assessment of a Molten Salt Reactor Design – The Liquid-Fluoride Thorium Reactor (LFTR).” 3002005460. October 2015. <http://www.thmfgrcs.com/Assessment_of_a_Molten_Salt_Reactor_Design.pdf> (Accessed 16 November 2022).

Feynberg, O. “MSRs Development in Russia.” National Research Center “Kurchatov Institute.” n.d. <https://indico.ictp.it/event/8725/session/1/contribution/4/material/slides/0.pdf> (Accessed 21 September 2022).

Flibe Energy. “LFTR Technology.” 2022. < https://flibe-energy.com/lftr/> (Accessed 16 November 2022).

Forsberg, C. and P.F. Peterson. “Spent Nuclear Fuel and Graphite Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal.” *Nuclear Technology.* Vol. 191. pp. 113–121. 2015.

Forsberg, C.W., D.F. Williams, E. Greenspan, M.W. Golay, and R. Moir. “Generation IV Roadmap Activity Description of Generation IV Reactor and Fuel Cycle Molten Salt Reactors (MSRs) for Production of Electricity with Fissile, Fertile, and Fission Products Dissolved in a Fluoride Salt.” Oak Ridge, Tennessee: Oak Ridge National Laboratory. 2001. <https://technicalreports.ornl.gov/cppr/y2001/misc/110694.pdf> (Accessed 10 October 2022).

Grabaskas, D., T. Fei, and J. Jerden. “Technical Letter Report on the Assessment of Tritium Detection and Control in Molten Salt Reactors.” ANL/NSE-20-15. Argonne National Laboratory. ML20157A155. 2020. <https://www.nrc.gov/docs/ML2015/ML20157A155.pdf> (Accessed 15 December 2022).

Halper, M. “A Plan to Turn Japan’s Nuclear Past Into Its Future With Molten Salt Reactors.” Weinberg Next Nuclear. March 22, 2013. <https://web.archive.org/web/20170309131404/http://www.the-weinberg-foundation.org/2013/03/22/a-plan-to-turn-japans-nuclear-past-into-its-future-with-molten-salt-reactors/> (Accessed 20 September 2022).

Haubenreich, P.N. “Fluorine Production and Recombination in Frozen MSR Salts After Reactor Operation.” ORNL-TM-3144. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1970. <http://moltensalt.org/references/static/downloads/pdf/ORNL-TM-3144.pdf> (Accessed 9 October 2022).

Haubenreich, P.N. and J.R. Engel. “Experience With the Molten-Salt Reactor Experiment.” *Nuclear Applications & Technology.* Vol. 8. pp. 118–136. 1970. <https://web.archive.org/web/20150129012419/http://www.energyfromthorium.com/pdf/NAT_MSREexperience.pdf> (Accessed 7 September 2022).

Holcomb, D.E., G.F. Flanagan, G.T. Mays, W.D. Pointer, K.R. Robb, and G.L. Yoder. “Fluoride Salt-Cooled High-Temperature Reactor Technology Development and Demonstration Roadmap.” ORNL/TM-2013/401. Oak Ridge, Tennessee: Oak Ridge National Laboratory. September 2013. <https://info.ornl.gov/sites/publications/files/Pub45913.pdf> (Accessed November 28, 2022).

Holcomb, D.E., G.F. Flanagan, B.W. Patton, J.C. Gehin, R.L. Howard, and T.J. Harrison. “Fast Spectrum Molten Salt Reactor Options.” July 2011. ORNL/TM-2011/105. <https://info.ornl.gov/sites/publications/files/Pub29596.pdf> (Accessed 5 October 2022).

Heuer, D., E. Merle-Lucotte, M. Allibert, M. Brovchenko, V. Ghetta, and P. Rubiolo. “Towards the thorium fuel cycle with molten salt fast reactors.” Annals of Nuclear Energy, Vol. 64, 2014, pp. 421-429.

IAEA. “Status Report – ThorCon (Thorcon US, Inc.) USA/Indonesia.” International Atomic Energy Agency. June 22, 2020. <https://aris.iaea.org/PDF/ThorCon_2020.pdf> (Accessed 27 December 2022).

IAEA. “MSFR (CNRS, France).” International Atomic Energy Agency. n.d. <https://aris.iaea.org/PDF/MSFR.pdf> (Accessed 4 October 2022).

Ignatiev, V. “Developing the Next Generation of Molten Salt REactor Systems in Russian Federation.” National Research Center “Kurchatov Institute.” Molten Salt Reactor Workshop, PSI, Switzerland. January 24, 2017. <https://www.gen-4.org/gif/upload/docs/application/pdf/2017-05/08_victor_ignatiev_russia.pdf> (Accessed 21 September 2022).

INL. “Fuel Conditioning Facility, Waste Forms Separations.” Idaho Falls, Idaho: Idaho National Laboratory. 2021. <https://factsheets.inl.gov/FactSheets/Fuel%20Conditioning%20Facility_2022.pdf> (Accessed 10 October 2022).

Johnson, S.G., S.M. Frank, T.P. O’Holleran, M.H. Noy, T. Disanto, K.M. Goff, and K.J. Bateman, “Characterization and Durability Testing of a Glass-Bonded Ceramic Waste Form,” Ceramic Transactions, Vol. 93, 1999, p. 313.

Kitcher, E.D. “A White Paper: Potential Disposition Options for a Liquid-Fueled Molten Salt Reactor at INL,” INL/EXT-20-57831.” March 2020.

KP. “Kairos Power – Company Overview.” 2022. <https://r6.ieee.org/scv-lm/wp-content/uploads/sites/88/Kairos-Power-Company-Overview-2022-slides-1.pdf> (Accessed 3 October 2022).

KP. “Hermes Non-Power Reactor Environmental Report.” HER-ER-001. Revision 0. Kairos Power. October 31, 2021a. <https://www.nrc.gov/docs/ML2130/ML21306A133.pdf> (Accessed 9 December 2022).

KP. “Hermes Non-Power Reactor Preliminary Safety Analysis Report.” HER-PSAR-001. Revision 0. Kairos Power. September 29, 2021b. <https://www.nrc.gov/docs/ML2127/ML21272A378.pdf> (Accessed 9 December 2022).

KP. “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor.” Topical Report KP-TR-003-NP, Revision 1. Kairos Power. July 2019. <https://www.nrc.gov/docs/ML1921/ML19212A756.pdf> (Accessed 3 October 2022).

Kramer, K. "TerraPower's Molten Chloride Fast Reactor Technology." ETEC Nuclear Suppliers Workshop. TerraPower. October 2, 2018. <https://www.researchgate.net/publication/334171092_TerraPower%27s_Molten_Chloride_Fast_Reactor_Technology> (Accessed 22 September 2022).

Latkowski, J. "TerraPower's Molten Chloride Fast Reactor (MCFR)." TerraPower. February 22, 2021. <https://www.google.com/url?sa=t&rct=j&q=&esrc=s&source=web&cd=&ved=2ahUKEwiZjdKH66j6AhX7IGoFHXzPB4oQFnoECAcQAQ&url=https%3A%2F%2Fwww.nationalacademies.org%2Fevent%2F02-22-2021%2Fdocs%2FDB0D308269688B2BD7B1AF60BAA143D48890C2DE80BB&usg=AOvVaw3K5U5sy6cZ9uG4ooDSdf_L> (Accessed 22 September 2022).

Lee, Ki Rak et al. "Investigation of physical and chemical properties for upgraded SAP (SiO_2-Al_2O_3-P_2O_5) waste form to immobilize radioactive waste salt," Journal of Nuclear Materials, Vol. 515, 2019, p. 382.

Manik, F.R.L., F. Suharyana, Riyatun Anwar, and A. Khakim. "Safety Analysis TMSR-500 in Terms of the Temperature Reactivity Coefficient of the Fuel and the Moderator Using the MCNP6 Software." ICAMBF 2020. <https://iopscience.iop.org/article/10.1088/1742-6596/1912/1/012010/pdf> (Accessed 5 October 2022).

McFarlane, J., P. Taylor, D. Holcomb, and W.P. Poore. "Review of Hazards Associated with Molten Salt Reactor Fuel Processing Operations." ORNL/TM-2019/1195. 2019. <https://info.ornl.gov/sites/publications/Files/Pub126864.pdf> (Accessed 19 September 2022).

McKay, H.A.C. "Elimination of Waste Actinides by Recycling Them to Nuclear Reactors." IAEA Bulletin, Vol. 23, No.2. Vienna, Austria: International Atomic Energy Agency. 1981. <https://www.iaea.org/sites/default/files/publications/magazines/bulletin/bull23-2/23204894649.pdf> (Accessed 10 October 2022).

McMillan, B. "Molten Salt Reactor Experiment Project Initiatives." Oak Ridge, Tennessee: Oak Ridge National Laboratory. 2019. <https://www.energy.gov/sites/prod/files/2019/11/f68/ORSSAB%20MSRE%20Presentation%20-%20Nov%202019.pdf> (Accessed 9 October 2022).

Notz, K.J. "Decommissioning of the Molten Salt Reactor Experiment a Technical Evaluation." ORNL/RAP-17. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1988. <https://technicalreports.ornl.gov/1988/3445602722702.pdf> (Accessed 9 October 2022).

Notz, K.J. "Extended Storage-in-Place of MSRE Fuel Salt and Flush Salt." ORNL/TM-9756. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1985. <http://moltensalt.org/references/static/downloads/pdf/ORNL-TM-9756.pdf> (Accessed 9 October 2022).

NRC. NUREG-2263, "Environmental Impact Statement for the Construction Permit for the Kairos Hermes Test Reactor - Draft Report for Comment." September 26, 2022. <https://www.nrc.gov/docs/ML2225/ML22259A126.pdf> (Accessed 9 December 2022).

NRC. “Safety Evaluation of the Kairos Topical Report KP-TR-005-P, Revision 1, ‘Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor.” June 1, 2020. <https://www.nrc.gov/docs/ML2014/ML20148M230.pdf> (Accessed 3 October 2022).

Nuclear Engineering International. “China Approves Launch of Liquid-salt Thorium Reactor.” August 11, 2022a. <https://www.neimagazine.com/news/newschina-approves-launch-of-liquid-salt-thorium-reactor-9920912> (Accessed 3 October 2022).

Nuclear Engineering International. “The Waste-burning Stable Salt Reactor.” March 21, 2022b. <https://www.neimagazine.com/features/featurethe-waste-burning-stable-salt-reactor-9563796/> (Accessed 4 October 2022).

ORANO. “Integration of MSRs in LW-SMR Fleets to Close Their Fuel Cycle and/or Manage waste.” Presentation at IAEA Technical Meeting. Châtillon, Hauts-de-Seine, France: ORANO. 2022. <https://conferences.iaea.org/event/321/attachments/13273/20589/D2_25_France_Integration%20of%20MSRs%20of%20LW-SMR%20fleets_Isabelle.pdf> (Accessed 10 October 2022).

ORSSAB. “Molten Salt Reactor Continues to Test Skills and Patience.” *Advocate*, Issue 38. Oak Ridge, Tennessee: Oak Ridge Site Specific Advisory Board. 2010. <https://web.archive.org/web/20130222013818/http://www.oakridge.doe.gov/em/ssab/Publications/Advocates/4-10.pdf> (Accessed 9 October 2022).

Ottewitte, E. “Cursory First Look at the Molten Chloride Fast Reactor as an Alternative to the Conventional BATR Concept.” 1992. <http://egeneration.org/wp-content/Repository/Feasibility_and_concept_study/MCFR_BATR.pdf#:~:text=A%20Molten%20Chloride%20Fast%20Reactor%20%28MCFR%29%20concept%20with,Continuous%20processing%20and%20refueling%20would%20minimize%20reactor%20downtime.> (Accessed 20 December 2022).

Paviet, P. “The Fuel Cycle of a Molten Salt Reactor.” RIC Conference. March 9, 2022. <https://ric.nrc.gov/Docs/Abstracts/pavietp-hv-w19.pdf> (Accessed 8 September 2022).

Peretz, F.J. “Identification and Evaluation of Alternatives for the Disposition of Fluoride Fuel and Flush Salts from the Molten Salt Reactor Experiment at Oak Ridge National Laboratory, Oak Ridge, Tennessee.” ORNL/ER-380. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1996. <https://www.osti.gov/scitech/servlets/purl/441122> (Accessed 9 October 2022).

Priebe, S. and K. Bateman, “The Ceramic Waste Form Process at Idaho National Laboratory,” Nuclear Technology, Vol. 162, 2008, p. 199.

Riley, B.J., J. McFarlane, G.D. DelCul, J.D. Vienna, C.I. Contescu, L.M. Hay, A.V. Savino, and H.E. Adkins. “Identification of Potential Waste Processing and Waste Form Options for Molten Salt Reactors.” NTRD-MSR-2018-000379, PNNL-27723. 2018. <https://info.ornl.gov/sites/publications/Files/Pub114284.pdf> (Accessed 13 September 2022).

Robertson, R.C. “MSRE Design and Operations Report, Part 1, Description of Reactor Design.” ORNL-TM-728. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1965. <https://www.osti.gov/servlets/purl/4654707> (Accessed 9 October 2022).

Scott, I. “Modular Stable Salt Reactors – A Simpler Way to Use Molten Salt Fuel.” Saint John, New Brunswick, Canada: Moltex Energy. 2016. <https://www.nuclearinst.com/write/MediaUploads/Events/SMR2016/Session%202/Ian_Scott_Moltex_Energy_LLP.pdf> (Accessed 1 November 2022).

Scott, I. “Stable Salt Reactors – A New Platform Technology in Nuclear Fission.” October 15, 2020. <https://msrworkshop.ornl.gov/wp-content/uploads/2020/11/24_Scott_Moltex_SSR_ORNL1.pdf> (Accessed 31 December 2022).

Seaborg Technologies. “The Energy Landscape of Tomorrow–The Compact Molten Salt Reactor Will Complement Other Sustainable Energy Sources.” 2022. <https://www.seaborg.com/the-reactor> (Accessed 21 September 2022).

TerraPower. “TerraPower’s Molten Chloride Fast Reactor Technology: Nuclear for a Changing Energy Sector.” 2022. <https://www.terrapower.com/wp-content/uploads/2022/06/TP_2022_MCFR_Technology.pdf> (22 December 2022).

ThorCon. “Powering up our world with cheap, reliable, CO2-free electric power, now.” 2022a. <https://thorconpower.com/> (Accessed 5 October 2022).

ThorCon. “Indonesia ThorCon 3.5 GW fission power project.” 2022b. <https://thorconpower.com/project/> (Accessed 5 October 2022).

U.S. Atomic Energy Commission. “An Evaluation of the Molten Salt Breeder Reactor.” WASH-1222. <https://www.osti.gov/servlets/purl/4372873> (Accessed 15 December 2022).

Wang, B. “China’s Molten Salt Nuclear Reactors.” Next Big Future. August 23, 2021. <https://www.nextbigfuture.com/2021/08/chinas-molten-salt-nuclear-reactors.html> (Accessed 20 September 2022).

World Nuclear Association. “Molten Salt Reactors.” May 2021. <https://world-nuclear.org/information-library/current-and-future-generation/molten-salt-reactors.aspx#:~:text=During%20the%201960s%2C%20the%20USA%20developed%20the%20molten,years%20to%201969%20%28the%20MSR%20programme%20ran%201957-1976%29.> (Accessed 1 November, 2022).

World Nuclear Association. “Synroc Wasteform.” April 2019. <https://www.world-nuclear.org/information-library/nuclear-fuel-cycle/nuclear-wastes/synroc.aspx> (22 December 2022).