



Technical Letter Report  
TLR-RES/DE/REB-2022-04

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***Preliminary Assessment of Considerations for the Front-End  
Activities for Molten Salt Reactors***

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Date: April 2022

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## EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) has developed an overarching plan to support the licensing and certification activities for advanced non-light-water reactor (non-LWR) reactor technologies. The non-LWR Vision and Strategy Near-Term Implementation Action Plans (NRC, 2017) directly supports NRC's responsibilities under the Nuclear Energy Innovation Capabilities Act of 2017 (U.S. Congress, 2017), which includes objectives to enable civilian research and development of advanced nuclear energy technologies by private and public institutions, and to expand theoretical and practical knowledge of nuclear physics, chemistry, and materials science, among other purposes.

The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) has been monitoring both the U.S. Department of Energy (DOE) and industry's activities related to the development of advanced non-LWR technologies, including those related to the potential licensing and certification of molten salt reactors (MSRs). To date, the NRC is aware of five commercial vendors that have expressed interest or have been actively engaged in the development and potential near-term licensing and certification of MSR technologies, namely:

- the Integral Molten Salt Reactor under development by Terrestrial Energy USA,
- the Fluoride Salt-Cooled High Temperature Reactor under development by Kairos Power,
- the Lithium Fluoride Thorium Reactor under development by Flibe Energy,
- the Molten Chloride Fast Reactor under development by TerraPower and Southern Nuclear, and
- the Molten Salt Research Reactor under development by Abilene Christian University.

The objective of this report is to provide a preliminary assessment of potential technical and regulatory considerations related to the management of unirradiated (fresh) fissile and fertile fuel materials for these near-term MSR technologies, including activities related to their enrichment, production, and transportation to a reactor site (i.e., front-end activities). The preliminary assessment was structured to provide the following:

- General background information related to prior experience on operating molten salt reactors, and their associated fuel enrichment, production and transportation operations (Section 2),
- General background on near-term MSR technologies and associated fuel materials, and the current state of knowledge on fuel enrichment, production and transportation options considered by their commercial vendors (Section 3),
- Preliminary assessment of potential technical and regulatory considerations related to the enrichment, production and transportation of fuel materials for near-term MSR technologies under development (Section 4), and
- Summary and recommendations for follow-on activities to support NMSS activities related to MSR fuel cycles (Section 5).

## **ACKNOWLEDGMENTS**

The authors thank N. Hansing, T. Boyce, and R. Iyengar for their valuable support and encouragement during the course of the effort and for their comments on the manuscript. The authors also wish to thank B. Beasley, W. Kennedy, W. Jessup, A. Muniz, R. Rivera, M. Sutton and C. Van Wert for their review and input regarding near-term molten salt technologies.

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## ACRONYMS

ADAMS	Agencywide Documents Access and Management System
ADU	ammonium diuranate
AEC	U.S. Atomic Energy Commission
ANP	Aircraft Nuclear Propulsion
ARDP	Advanced Reactor Demonstration Program
ARE	Aircraft Reactor Experiment
AUC	ammonium uranyl carbonate
DOE	Department of Energy
DOT	Department of Transportation
EPRI	Electric Power Research Institute
Euratom	European Atomic Energy Commission
FHR	Fluoride Salt-Cooled High Temperature Reactor
GAIN	Gateway for Accelerated Innovation in Nuclear
HALEU	high-assay low-enriched uranium
IAEA	International Atomic Energy Agency
IMSR	Integral Molten Salt Reactor
ISA	integrated safety analyses
LEU	low-enriched uranium
LFTR	Lithium Fluoride Thorium Reactor
LWR	light-water reactor
MCFR	Molten Chloride Fast Reactor
MCRE	Molten Chloride Reactor Experiment
MOSART	Molten Salt Actinide Recycler and Transmuter
MSBR	Molten Salt Breeder Reactor
MSR	molten salt reactor
MSRE	Molten Salt Reactor Experiment
MSRR	Molten Salt Research Reactor
NMSS	Office of Nuclear Material Safety and Safeguards
NNL	National Nuclear Laboratory (UK)
NRC	U.S. Nuclear Regulatory Commission
OPG	Ontario Power Generation
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PCP	packaging certification program
PEL	permissible exposure limit
RAMPAC	Radioactive Materials Packaging (database)
RRIP	Research Reactor Infrastructure Program
SAMOFAR	Safety Assessment of the Molten Salt Fast Reactor
SMR	small modular reactor
SNM	special nuclear material
TRISO	tristructural isotropic

## CHEMICAL ELEMENTS AND COMPOUNDS

Be	beryllium	N	nitrogen
BeF <sub>2</sub>	beryllium fluoride	NH <sub>3</sub>	ammonia
BeO	beryllium oxide	NH <sub>4</sub> F	ammonium fluoride
Bi	bismuth	NH <sub>4</sub> HF <sub>2</sub>	ammonium bifluoride
BiCl <sub>3</sub>	bismuth chloride	Na	sodium
Br	bromine	NaCl	sodium chloride
C	carbon	NaF	sodium fluoride
CCl <sub>4</sub>	carbon tetrachloride	NaNO <sub>3</sub>	sodium nitrate
CCl <sub>x</sub> F <sub>(4-x)</sub>	freon	NO <sub>x</sub>	nitrogen oxide
C <sub>3</sub> Cl <sub>6</sub>	hexachloropropene	O	oxygen
C <sub>2</sub> H <sub>2</sub>	acetylene	OPyC	outer pyrolytic carbon
C <sub>3</sub> H <sub>6</sub>	propylene	Pb	lead
C <sub>6</sub> H <sub>12</sub> N <sub>4</sub>	hexamethylenetetramine	PbCl <sub>2</sub>	lead chloride
CH <sub>4</sub> N <sub>2</sub> O	urea	Pu	plutonium
CH <sub>3</sub> SiCl <sub>3</sub>	methyltrichlorosilane	PuF <sub>3</sub>	plutonium fluoride
CO	carbon monoxide	PuO <sub>2</sub>	plutonium oxide
CO <sub>2</sub>	carbon dioxide	Si	silicon
COCl <sub>2</sub>	phosgene	SiC	silicon carbide
Ca	calcium	Solar salt	mixture of NaNO <sub>3</sub> and KNO <sub>3</sub>
CaCl <sub>2</sub>	calcium chloride	Th	thorium
Cd	cadmium	ThF <sub>4</sub>	thorium fluoride
CdCl <sub>2</sub>	cadmium chloride	Th(NO <sub>3</sub> ) <sub>3</sub>	thorium nitrate
Cl	chlorine	ThO <sub>2</sub>	thorium oxide (thoria)
F	fluorine	Th(OH) <sub>4</sub>	thorium hydroxide
FLiBe salt	mixture of LiF and BeF <sub>2</sub>	ThOF <sub>2</sub>	thorium oxyfluoride
FLiNaK salt	mixture of LiF, NaF, and KF	TRISO	tristructural isotropic
H	hydrogen	U	uranium
HBr	hydrogen bromide (hydrobromic acid if aqueous)	UC <sub>2</sub>	uranium dicarbide
HCl	hydrogen chloride (hydrochloric acid if aqueous)	UCO	uranium oxycarbide
HF	hydrogen fluoride (hydrofluoric acid if aqueous)	UCl <sub>3</sub>	uranium trichloride
HI	hydrogen iodide (hydroiodic acid if aqueous)	UF <sub>4</sub>	uranium tetrafluoride
HNO <sub>3</sub>	nitric acid	UF <sub>6</sub>	uranium hexafluoride
IPyC	inner pyrolytic carbon	UO <sub>2</sub>	uranium dioxide
KNO <sub>3</sub>	potassium nitrate	UO <sub>3</sub>	uranium trioxide
LiF	lithium fluoride	U <sub>3</sub> O <sub>8</sub>	uranium oxide
MgCl <sub>2</sub>	magnesium chloride	UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub>	uranyl nitrate
		UO(OH) <sub>2</sub>	uranyl hydroxide
		UO <sub>2</sub> F <sub>2</sub>	uranyl fluoride
		UO <sub>2</sub> Cl <sub>2</sub>	uranyl chloride
		Zr	zirconium
		ZrF <sub>4</sub>	zirconium tetrafluoride

# 1 INTRODUCTION

## 1.1 Regulatory Driver

The U.S. Nuclear Regulatory Commission (NRC) has developed an overarching plan to support the licensing and certification activities for advanced non-light-water reactor (non-LWR) reactor technologies. The non-LWR Vision and Strategy Near-Term Implementation Action Plans (IAPs) (NRC, 2017) directly support NRC's responsibilities under the Nuclear Energy Innovation Capabilities Act of 2017 (U.S. Congress, 2017), which includes as objectives to enable civilian research and development of advanced nuclear energy technologies by private and public institutions, and to expand theoretical and practical knowledge of nuclear physics, chemistry, and materials science, among other purposes.

The NRC's Office of Nuclear Material Safety and Safeguards (NMSS) has been monitoring both the U.S. Department of Energy and industry's activities related to the development of advanced non-LWR technologies, including those related to the potential licensing and certification of molten salt reactors (MSRs). To date, the NRC is aware of five (5) vendors that have expressed interest or have been actively engaged in the development and potential near-term licensing and certification of MSR technologies, namely:

- the Integral Molten Salt Reactor under development by Terrestrial Energy USA,
- the Fluoride Salt-Cooled High Temperature Reactor under development by Kairos Power,
- the Lithium Fluoride Thorium Reactor under development by Flibe Energy,
- the Molten Chloride Fast Reactor under development by TerraPower and Southern Nuclear, and
- the Molten Salt Research Reactor under development by Abilene Christian University.

As part of its regulatory mission, NMSS has been particularly interested in the vendor-proposed plans for activities related to the front-end and back-end of MSR fuel cycles, including the enrichment, production and transportation of unirradiated (fresh) fissile and fertile fuel materials, and the storage and transportation of irradiated (spent) fuel and carrier salt materials. The timely understanding of potential technical and regulatory issues related to the licensing, certification, and operation of facilities and systems for materials management is an important component for supporting activities and timelines related to reactor licensing and operation.

## **1.2 Report Objectives**

The objective of this report is to provide a preliminary assessment of potential technical and regulatory considerations related to the management of unirradiated (fresh) MSR fissile and fertile fuel materials prior to reactor operations, including activities related to their enrichment, production, and transportation to a reactor site. The preliminary assessment was structured to provide the following:

- General background information related to prior experience on operating molten salt reactors, and their associated fuel enrichment, production and transportation activities (Section 2),
- General background on near-term MSR technologies and associated fuel materials, and the current state of knowledge on fuel enrichment, production and transportation options considered by their commercial vendors (Section 3),
- Preliminary assessment of potential technical and regulatory considerations related to the enrichment, production and transportation of fuel materials for near MSR technologies under development (Section 4), and
- Summary and recommendations for follow-on activities to support NMSS activities related to MSR fuel cycles (Section 5).

## 2 BACKGROUND AND PRIOR OPERATING EXPERIENCE

### 2.1 Molten Salt Reactors

The concept of a homogeneous, molten salt-cooled nuclear reactor was first envisioned in 1948 when scientists were evaluating high temperature fluid-fueled reactors that are not constrained by the limitations of solid fuels and conventional coolants (Williams, et al., 2017). Molten salt reactor (MSR) concepts are based on using a molten salt mixture as a primary nuclear reactor coolant, while the fuel can be either directly dissolved in the coolant (i.e., liquid-fueled) or can be in a separate solid form (i.e., salt-cooled). MSRs can be operated in either thermal or fast neutron spectra, as simple fissile convertors or breeder reactors (e.g., utilizing both fissile and fertile fuels), as well as accelerator-driven sub-critical reactors or transuranic element burners.

Molten salt coolants have very high heat capacities and heat transfer properties, as well as being thermally and radiation stable, which make them suitable coolants compared to water used in current light-water reactors (LWRs) (Le Brun, 2007). Their high boiling point and low vapor pressures allow for reactor operation at lower pressure (i.e., atmospheric) and higher temperature than current light-water reactor technologies. This, in turn, provides increased power conversion efficiencies (approximately 50 percent more kWh(e)) due to the higher thermal efficiency.

MSRs provide additional inherent safety relative to current LWR technologies (Elsheikh, 2013). The fuel-salt mixtures used in fluid-cooled MSRs exhibit large negative temperature coefficients of reactivity due to their high thermal expansion; therefore, if the reactor overheats, the reactivity in the core slows down. In addition, the core unit does not require active engineered safety systems for decay heat removal (i.e., MSRs rely on passive cooling). Since the fuel-salt mixture is in liquid form, during a design-basis accident it can be quickly drained out of the reactor into a passively cooled dump tank. MSR designs can implement an actively-cooled salt plug at the bottom of the core kept at a temperature below the freezing point of the salt. If the reactor core temperature rises beyond a critical point, the plug melts, and the liquid fuel is immediately evacuated, pouring into a sub-critical geometry in a catch basin. Finally, since MSRs operate at atmospheric pressure and use no coolant water, the risk of a steam or hydrogen-led explosion during a design-basis accident is not credible.

Liquid-fueled MSRs also allow for increased flexibilities for fissile and fertile material utilization through continuous fission product removal via physical and pyrochemical processes. The fuel composition can be periodically adjusted online to minimize outages. Significant research efforts are focused on reactor design and supplementing thermodynamic and electrochemical data of fuel-salt halide mixtures, particularly those based on chlorides.

## 2.2 Prior Operating Experience on Molten Salt Reactors

Oak Ridge National Laboratory (ORNL) has historically led efforts to support the development of MSR technologies, as shown in Table 2-1, with two designs being built and operated (Fredrickson et al., 2018). The following discussion provides a brief overview of those technologies and associated front-end activities for the enrichment, production, mixing, and transportation of their fuel-salt mixtures.

**Table 2-1 Molten Salt Reactor Development Activities at Oak Ridge National Laboratory (Fredrickson et al., 2018)**

<b>Aircraft Nuclear Propulsion Program (1949 – 1961)</b>	Aircraft Reactor Experiment: 2.5 MWt operated Aircraft Reactor Test: 60 MWt proposed
<b>Two-Fluid Molten Salt Breeder Reactor (1961 – 1970)</b>	Molten Salt Breeder Reactor: 1,000 MWe design
<b>Molten Salt Reactor Experiment (1960 – 1975)</b>	8 MWt operated
<b>Molten Salt Converter Reactor (1961 – 1965)</b>	1,000 MWe design
<b>Molten Salt Breeder Experiment (1969 – 1970)</b>	150 MWt proposed
<b>Molten Salt Reactor Demonstration (1972)</b>	350 MWe proposed
<b>Single-Fluid Molten Salt Breeder Reactor (1969 – 1978)</b>	1,000 MWe design
<b>Denatured Molten Salt Reactor (1977 – 1980)</b>	1,000 MWe design

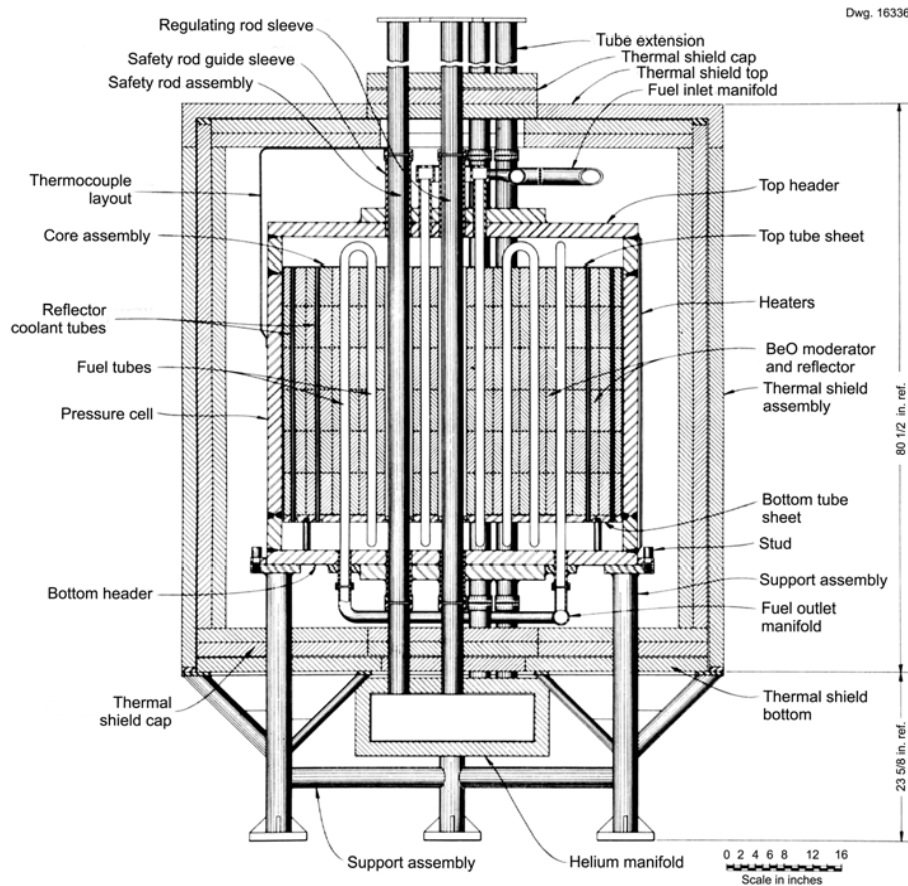
### 2.2.1 Aircraft Reactor Experiment

The Aircraft Nuclear Propulsion (ANP) project at ORNL was established in 1949, at the request of the Atomic Energy Commission (AEC), to provide technical support to the Air Force's efforts on propulsion of supersonic aircraft (Cottrell et al., 1955). In response to the ANP, ORNL proposed to design and operate a reactor, named the Aircraft Reactor Experiment (ARE) in which a molten metal would flow through the core and then through a heat exchanger where it would heat the air for propulsion.

The ARE reactor assembly consisted of an Inconel pressure shell in which beryllium oxide (BeO) moderator and reflector blocks were stacked around fuel tubes, reflector cooling tubes, and control assemblies. The reactor core relied on sodium metal as coolant for the reflector and pressure shell as well as heat transfer medium by which moderator heat was readily transmitted to the fuel stream. The excess heat was removed from the sodium metal via a heat exchanger. The fuel was a mixture of sodium fluoride (NaF) and zirconium tetrafluoride (ZrF<sub>4</sub>), with sufficient uranium tetrafluoride (UF<sub>4</sub>) added to make the reactor critical (see Table 2-2). Figure 2-1 and Figure 2-2 show a schematic and flow diagram for the ARE core. The design and operation of the ARE is extensively described by Cottrell et al. (1955). The ARE was operated from November 3-12, 1954 for a total of 96 MWh. The ANP project was abruptly terminated in June 1961.

**Table 2-2 Aircraft Reactor Experiment – Design Parameters and Relevant Dates (Rosenthal, 2010)**

<b>Maximum Power (Thermal)</b>	2.5 MWt
<b>Fuel-Salt</b>	53 percent NaF, 41 percent ZrF <sub>4</sub> , 6 percent UF <sub>4</sub> (93 percent enriched <sup>235</sup> U)
<b>Fuel Melting Point</b>	532 °C [990 °F]
<b>Fuel Inlet Temperature</b>	663 °C [1225 °F]
<b>Fuel Outlet Temperature</b>	860 °C [1580 °F] (882 °C [1620 °F] at peak)
<b>Fuel Flow Rate</b>	45 gal/min [2.8 L/s]
<b>Moderator</b>	BeO hexagonal blocks, 3.6 in. across × 6 in. high with 1.25 in. coolant passages
<b>Coolant</b>	Helium to water
<b>Reactor Vessel Material</b>	Inconel
<b>Critical</b>	November 3, 1954
<b>Shut Down</b>	November 12, 1954



**Figure 2-1 Schematic of ARE Core (Rosenthal, 2010).**



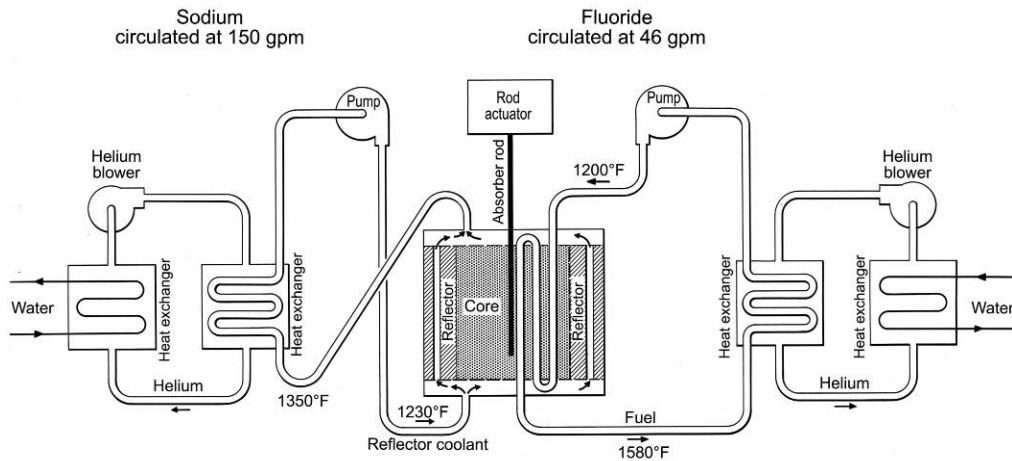


Figure 2-2 ARE Flow Diagram (Rosenthal, 2010).

## 2.2.2 Molten Salt Reactor Experiment

In 1956, following ARE operation, ORNL began to study the application of molten salt reactors for power station generation under the Civilian Nuclear Power Program. ORNL evaluated the technical feasibility and associated costs for both a single-fluid (fissile) system and a two-fluid (fissile/fertile) breeder system. ORNL pursued the design of a single-fluid fissile system to minimize design complexities. In 1960, the design for the Molten Salt Reactor Experiment (MSRE) began under the Molten Salt Reactor Program (MSRP). The MSRE was an eight megawatt-thermal (MWt) single-fluid test reactor, which operated successfully for over 17,000 hours between 1965 and 1969 (Rosenthal et al., 1970; Haubenreich, et al., 1970; Peretz, 1996). The reactor used a primary molten fluoride mixture of  $\text{UF}_4$  fuel in a carrier salt composed of lithium fluoride ( $\text{LiF}$ ), beryllium fluoride ( $\text{BeF}_2$ ), and zirconium tetrafluoride ( $\text{ZrF}_4$ ). The fuel-salt mixture circulated through the reactor vessel with a prismatic structure of unclad graphite moderator, a fuel-salt pump, and primary heat exchanger. The reactor vessel and piping components were made of Hastelloy N, a high strength nickel base alloy developed under the ANP Program (AEC, 1972). Heat was transferred from the fuel-salt mixture to a secondary coolant salt ( $\text{LiF}:\text{BeF}_2$ , 2:1 mole ratio) in the primary heat exchanger. The coolant salt passed from the primary heat exchanger to an air-cooled radiator, a coolant salt pump, and then returned to the primary heat exchanger.

The MSRE operated between June 1, 1965 and December 12, 1969 for a total of 105 GWh. The test demonstrated that fluoride salts in a liquid fuel are invulnerable to radiation-induced or thermal-induced degradation at the operating temperatures. The experiment also identified technical considerations related to the core performance of graphite and Hastelloy N components (AEC, 1972). Table 2-3 provides design and fuel parameters, as well as relevant dates for the MSRE.

**Table 2-3 Molten Salt Reactor Experiment – Design Parameters and Relevant Dates (AEC, 1972)**

<b>Maximum Power (Thermal)</b>	8 MWt
<b>Fuel Processing Method</b>	Batch, off-line
<b>Fuel/Carrier Salts</b>	<sup>7</sup> LiF (65 mole percent) - BeF <sub>2</sub> (29.1 mole percent) - ZrF <sub>4</sub> (5.0 mole percent) - UF <sub>4</sub> (0.9 mole percent) (nominal) 99.99 percent isotopically pure <sup>7</sup> LiF Fuel concentrate: <sup>7</sup> LiF (73 mole percent) - UF <sub>4</sub> (27 mole percent) Enrichment compositions: <sup>235</sup> U (20 percent) : <sup>238</sup> U (80 percent) <sup>233</sup> U (80 percent) : <sup>235</sup> U (20 percent) <sup>239</sup> Pu as PuF <sub>3</sub>
<b>Coolant/Flush Salts</b>	<sup>7</sup> LiF:BeF <sub>2</sub> (2:1 mole ratio)
<b>Fuel Outlet Temperature</b>	654°C [1210 °F]
<b>Fuel Flow Rate</b>	1200 gal/min [75.7 L/s]
<b>Moderator</b>	Graphite (unclad, unsealed)
<b>Secondary Coolant Salt</b>	<sup>7</sup> LiF-BeF <sub>2</sub> (2:1 mole ratio)
<b>Reactor Vessel Material</b>	Hastelloy N
<b>Design Initiated</b>	July 1960
<b>Critical with <sup>235</sup>U Fuel</b>	June 1, 1965
<b>Critical with <sup>233</sup>U Fuel</b>	October 2, 1968
<b>Hours Critical</b>	15,655
<b>Shut Down</b>	December 12, 1969

### **2.3 Prior Operating Experience on Enrichment and Production of Molten Salt Reactor Fuel-Salt Materials**

ORNL developed techniques and procedures for preparing and handling molten salts since 1953, which were implemented during support of the ANP project for the preparation of fluoride mixtures and their loading into the ARE. The molten salt production facility was operated by the Reactor Chemistry Division as an integral part of the MSRP to provide fused fluoride mixtures for its chemical and engineering tests (Shaffer, 1971). The facility also developed procedures for handling operations, including filling, sampling, and emptying engineering test loops.

The fluoride production facility had two processing units operating as a batch process, each with a capacity of about two cubic feet of fused salt per batch. Since commercial sources of fused fluoride mixtures that would meet MSRE specifications were not available, the production plant was adapted to support the repetitive preparation of fluoride mixtures of diverse chemical compositions to support engineering tests and the MSRE operation.

Fluoride-based feed materials for the MSRE primary fuel-salt, secondary coolant salt, and flush salts were purchased from commercial sources or otherwise obtained from the AEC, as summarized in Table 2-4 (Shaffer, 1971).

**Table 2-4 Summary of Fluoride Procured for the MSRE (Shaffer, 1971)**

Fluoride Salt	Estimate (lb)	Procured (lb)	Source
${}^7\text{LiF}$	12,476	22,000	AEC
$\text{BeF}_2$	11,055	12,000	Commercial
$\text{ZrF}_4$	2,283	2,300	Commercial
${}^{238}\text{UF}_4$	523	529	AEC
${}^{235}\text{UF}_4$	246	263	AEC

- *Lithium Fluoride (LiF)* – The material was sourced as a hydroxide from the AEC and was later converted to fluoride at the production facility in the Y-12 National Security Complex. To minimize neutron absorption cross section, all LiF produced had an isotopic assay of at least 99.99 percent  ${}^7\text{Li}$ .
- *Uranium tetrafluoride (UF<sub>4</sub>)* – Although the MSRE operated on a fuel mixture with UF<sub>4</sub> enriched to 32 percent  ${}^{235}\text{U}$ , the material provided by the AEC was highly enriched to 93 percent  ${}^{235}\text{U}$ . About 90 kilograms of  ${}^{235}\text{U}$  was procured directly as UF<sub>4</sub>. The balance of the uranium inventory added to the MSRE fuel charge was depleted  ${}^{235}\text{U}$ .
- *Zirconium tetrafluoride (ZrF<sub>4</sub>)* – Hafnium-free (<50 ppm) ZrF<sub>4</sub> was procured from commercial sources to minimize neutron absorption penalties.
- *Beryllium fluoride (BeF<sub>2</sub>)* – BeF<sub>2</sub> of best-commercial purity per MSRE chemical specifications was obtained from commercial sources.

The production method for fluoride mixtures was generally independent of fluoride mixture composition provided that the liquidus temperature was within the capability of the process equipment. The production of multicomponent mixtures, however, was facilitated by the preparation and subsequent combination of simpler binary or ternary mixtures. Aside from the physical mixing of salts to obtain lower liquidus temperatures, the production process allowed for further purification of the resultant molten fluoride mixture. The removal of a limited number of impurity species (e.g., oxides, sulfates) during the production operation was achieved by treatment of the fluoride melt with anhydrous hydrogen fluoride (HF), hydrogen (H<sub>2</sub>), and, in some instances, strong metallic reducing agents. Volatile impurities were removed in the process gas effluent stream, and insoluble particulates were removed by decantation and filtration.

The production facility for fluoride mixtures was constructed on a 40- by 40-ft area in the high bay of Building 9201-3 within the Y-12 National Security Complex. To meet industrial hygiene requirements for handling beryllium and other toxic fluoride salts, the facility was enclosed and equipped with an air ventilation and filtration system (air flow rate of 14,000 cubic feet per

minute). The air flow maintained the atmosphere of the production facility at a negative pressure with respect to the outside area to control airborne contamination. Specific work areas within the production plant were compartmented and provided with direct exhaust air ducts and filtered makeup air. The exhaust air was filtered prior to discharge outside the building.

Fused fluoride mixtures and the enriched fuel concentrate were produced in two batch processing units. Additional details on the operations are provided in Shaffer (1971).

#### **2.4 Prior Experience on Unirradiated Molten Salt Reactor Fuel-Salt Materials**

Very limited information is available on the transportation activities in support of the ARE and MSRE. MSRE's activities for production of enriched uranium concentrate and fused fluoride mixtures occurred in different buildings at the Y-12 National Security Complex (Shaffer, 1971). After processing, the fuel-salt mixtures were transferred to a storage container and placed in a nuclear-safe transport container and shipped to a security warehouse within the Y-12 national security complex. Information related to these activities has not been identified.

### 3 NEAR-TERM MOLTEN SALT REACTOR TECHNOLOGIES

The landscape for molten salt reactor (MSR) technologies is one of the most diverse for Generation IV nuclear power plants with a large number of vendors announcing plans or activities in support of their development. The U.S. Department of Energy (DOE) has also provided funding to support research and resolution to important technical issues for demonstrating adequate safety for operation of some of the designs. The U.S. Nuclear Regulatory Commission (NRC) is currently aware of five (5) vendors with the potential to pursue near-term licensing and certification activities for MSR technologies, including:

- the Integral Molten Salt Reactor under development by Terrestrial Energy USA,
- the Fluoride Salt-Cooled High Temperature Reactor under development by Kairos Power,
- the Lithium Fluoride Thorium Reactor under development by Flibe Energy,
- the Molten Chloride Fast Reactor under development by TerraPower and Southern Nuclear, and
- the Molten Salt Research Reactor under development by Abilene Christian University.

Table 3-1 compiles information on the characteristics and status of these MSR technologies. This section provides background on the designs and status of these near-term technologies, including the current state of knowledge for the enrichment, production, and transportation of their fuel materials.

The scope of this report excludes additional commercial vendors that have publicly communicated plans to develop other MSR technologies but have not yet engaged with the NRC on pre-application engagements. These include:

- the Stable Salt Reactor under development by Moltex Energy,
- the Molten Chloride Salt Fast Reactor under development by Elysium Industries,
- the Thorium Molten Salt Reactor 500, under development by ThorCon Power,
- the Mu\*STAR Accelerator-Driven Small Modular Reactor, under development by Muons, Inc., and
- the Micro Molten Salt Reactor (ARC Generator) under development by Alpha Tech Research Corp.

In addition, multiple international organizations and countries are developing MSR technologies, such as the Molten Salt Fast Reactor (SAMOFAR) project funded by European Atomic Energy Commission (Euratom), the Molten Salt Actinide Recycler and Transmuter (MOSART) concept explored by the Russian Federation, the Thorium-Fueled Liquid Fluoride MSR to be operated by the People's Republic of China, and the Thorium-Fueled Molten Salt Breeder Reactor designed by the Republic of India. The scope of this report excludes an assessment of these efforts and associated activities related to the enrichment, production, and transportation of fuel-salt mixtures.

**Table 3-1 Characteristics and Status of Molten Salt Reactor Technologies Under Commercial Development**

<b>Technology</b>	<b>Integral Molten Salt Reactor (IMSR) (Hill, 2020)</b>	<b>Molten Fluoride Salt-Cooled High Temperature Reactor (FHR) (Kairos, 2020b)</b>	<b>Lithium Fluoride Thorium Reactor (LFTR) (EPRI, 2015)</b>	<b>Molten Chloride Fast Reactor (MCFR) [Collaboration with Southern Company] (Latkowski, 2021)</b>	<b>Molten Salt Research Reactor (MSRR) (Schubert, 2020)</b>
<b>Commercial Vendor</b>	Terrestrial Energy USA	Kairos Power	Flibe Energy	TerraPower	Abilene Christian University
<b>Base Model Output (MWe, MWt)</b>	195 MWe; 442 MWth	140 MWe	250 MWe; 600 MWth	500-1200 MWe (grid-scale); 30-300 MWe (mid-scale)	1 MWth
<b>Neutron Spectrum</b>	Thermal	Thermal	Thermal	Fast	Thermal
<b>Moderator</b>	Graphite	Graphite <sup>1</sup>	Graphite	Multiple materials under consideration <sup>4</sup>	Graphite
<b>Heat Transfer Medium</b>	Molten fluoride salt	Molten fluoride salt	Molten fluoride salt	Molten chloride salt	Molten fluoride salt
<b>Outlet Temperature</b>	700 °C [1292 °F]	650 °C [1202 °F]	650 °C [1202 °F]	Unknown	Unknown
<b>Thermal Efficiency</b>	44 percent	45 percent (net)	42 percent (net)	Unknown	Unknown
<b>Fuel Enrichment (<sup>235</sup>U)</b>	<5 percent	19.75 percent	None	~12 percent	19.9 percent
<b>Fuel Composition</b>	Fluoride salt	TRISO (uranium oxycarbide)	Primary: UF <sub>4</sub> ( <sup>233</sup> U) Blanket: ThF <sub>4</sub> ( <sup>232</sup> Th)	Chloride salt	UF <sub>4</sub>
<b>Salt Coolant Composition</b>	Proprietary fluoride salt	Primary salt: FLiBe <sup>2</sup> (enriched with <sup>7</sup> Li) Secondary salt: Solar	Primary and secondary salt: FLiBe	Proprietary chloride salt	Primary salt: FLiBe (enriched with <sup>7</sup> Li) Secondary salt: Unknown
<b>Initial Reactor Demonstration / Siting</b>	Unknown	Low-power demonstration reactor at the East Tennessee Technology Park (ETTP) in Oak Ridge (Hermes Reactor) expected to be operational in 2026 <sup>3</sup> .	Unknown	The small-scale Molten Chloride Reactor Experiment will be located at the Idaho National Laboratory site and will inform the design of a small to mid-scale MCFR demonstration reactor	Test reactor expected to be sited at Abilene, Texas.
<b>DOE Support</b>	U.S. Industry Opportunities for Advanced Nuclear Technology Development	ADRP Risk-Reduction Awardee	ADRP Pathway Awardee; GAIN voucher	ADRP Risk-Reduction Awardee	RRIP Awardee
<b>DOE Laboratory Partner</b>	Argonne National Laboratory	Oak Ridge National Laboratory	Pacific Northwest National Laboratory	Idaho National Laboratory	Undecided

<b>Technology</b>	<b>Integral Molten Salt Reactor (IMSR) (Hill, 2020)</b>	<b>Molten Fluoride Salt-Cooled High Temperature Reactor (FHR) (Kairos, 2020b)</b>	<b>Lithium Fluoride Thorium Reactor (LFTR) (EPRI, 2015)</b>	<b>Molten Chloride Fast Reactor (MCFR) [Collaboration with Southern Company] (Latkowski, 2021)</b>	<b>Molten Salt Research Reactor (MSRR) (Schubert, 2020)</b>
<b>NRC Status</b>	Pre-application engagement. Regulatory engagement plan submitted 10/2021.	Construction Permit application for test reactor accepted for docketing and review in 11/2021. Pre-application engagement for power reactor project – several topical reports approved and more under review.	Pre-application engagement (limited).	Pre-application engagement (limited).	Pre-application engagement. Preparing a regulatory engagement plan following a public meeting held in 11/2021. Construction Permit application expected end of FY22.

<sup>1</sup> (Kairos, 2020a)

<sup>2</sup> (Hastings, 2020)

<sup>3</sup> (Haugh, 2021)

<sup>4</sup> (TerraPower, 2020)

### 3.1 Terrestrial Energy – Integral Molten Salt Reactor

#### 3.1.1 Technology Description and Status

The Integral Molten Salt Reactor (IMSR) is a thermal-spectrum liquid-fueled small MSR design with 195 megawatt-electric (MWe) base model output (scalable to 390 MWe). The IMSR is based on the Molten Salt Reactor Experiment (MSRE) operated by Oak Ridge National Laboratory (ORNL) with the use of similar core unit components and fuel-salt mixtures. The graphite-moderated molten fluoride salt reactor uses low-enriched uranium (LEU) in a fluoride-based homogeneous primary salt mixture (Hill, 2020).

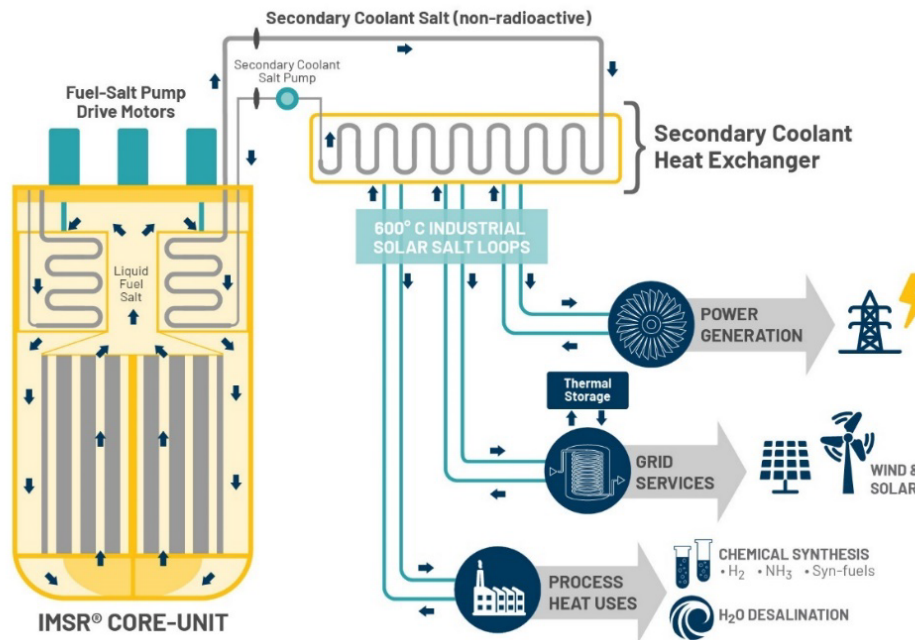


Figure 3-1 Conceptual Design of Terrestrial Energy's IMSR (Terrestrial, 2021a).

The reactor design features a completely sealed reactor vessel (core unit) with integrated pumps, heat exchangers and shutdown rods (Figure 3-1). The core unit is anticipated to be replaced completely as a single unit at the end of its useful service life (nominally 7 years) with a plant design-life of eight cycles (nominally 56 years). In the integral process within the core unit, the primary fissile fuel-salt mixture is pumped through a graphite moderator and then through the integral heat exchangers to transfer its heat to the secondary loop. The secondary loop contains a non-fissile coolant salt mixture, which transfers heat away from the primary heat exchangers to a tertiary salt loop that powers a steam generator. The core unit is designed to allow for continuous/periodic addition of fissile material that does not require opening or shut down for refueling.

The fissile fuel material for the IMSR is LEU, with enrichments between 2 – 3 percent <sup>235</sup>U at reactor startup and 4.95 percent <sup>235</sup>U for periodic fuel additions (Hill, 2020). The primary molten salt used for mixing with the fuel is a fluoride-based proprietary coolant salt.



Terrestrial Energy USA, Inc. (TEUSA) is currently engaged in pre-application activities with the NRC in support of a future application for a Standard Design Approval of the IMSR core unit under 10 CFR Part 52. TEUSA is an affiliate of Terrestrial Energy Inc. (TEI) and has entered into agreements with TEI to support TEUSA's development and deployment program for the IMSR in the U.S.

### **3.1.2 State of Knowledge on Fuel Enrichment, Fabrication, and Transportation Options**

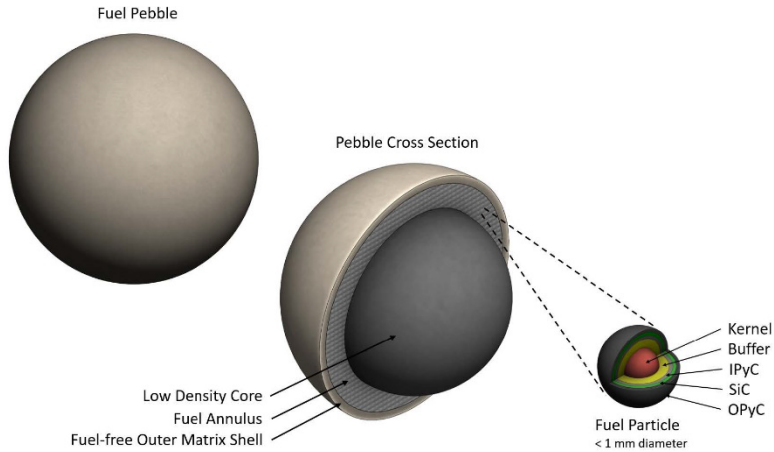
Terrestrial Energy recently announced an agreement with Westinghouse and the United Kingdom's National Nuclear Laboratory (NNL) to support the IMSR's fuel supply needs (Terrestrial, 2021b). NNL will support Westinghouse's Springfields Fuels production facility in Lancashire, England. A review of information submitted to the NRC has not identified specific details on plans for enrichment, fabrication, and transportation options under consideration for the IMSR, though docketed information indicates that TEUSA will use low-enriched fuel.

## **3.2 Kairos Power – Fluoride Salt-Cooled High Temperature Reactor**

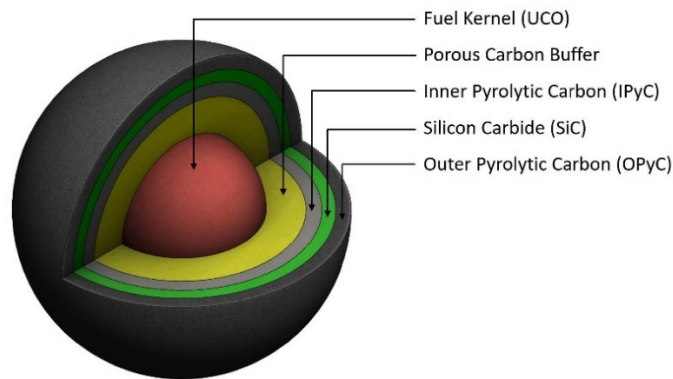
### **3.2.1 Technology Description and Status**

The Fluoride Salt-Cooled High Temperature Reactor (FHR) is a thermal-spectrum solid-fueled MSR design with 140 MWe base model output to operate at near atmospheric pressure using solid fuel within a molten fluoride salt coolant. The KP-FHR solid fuel design is based on spherical pebble fuel elements embedded with tristructural isotropic (TRISO) graphite-matrix coated particles (Kairos, 2020a). TRISO-based fuels were initially developed and successfully implemented for high temperature gas-cooled reactors.

The FHR pebble fuel element design contains a central sub-dense inner core surrounded by an annular layer of TRISO particles packed into a partially-graphitized matrix material (fuel pebble annulus) and covered by an outer shell of fuel-free matrix material (Figure 3-2) (Kairos, 2020a). Each TRISO particle is composed of a fissile kernel enriched with 19.75 percent <sup>235</sup>U in the form of a heterogeneous mixture of uranium oxide and uranium carbide (UCO) surrounded by successive coating layers of a porous carbon buffer layer, a dense inner pyrolytic carbon (IPyC) layer, a silicon carbide (SiC) layer, and a dense outer pyrolytic carbon (OPyC) layer (Figure 3-3). The TRISO particles are overcoated with resinated graphite powder, which are pressed and heat-treated to become the fuel matrix material in the fuel pebble annulus (Kairos, 2020b).



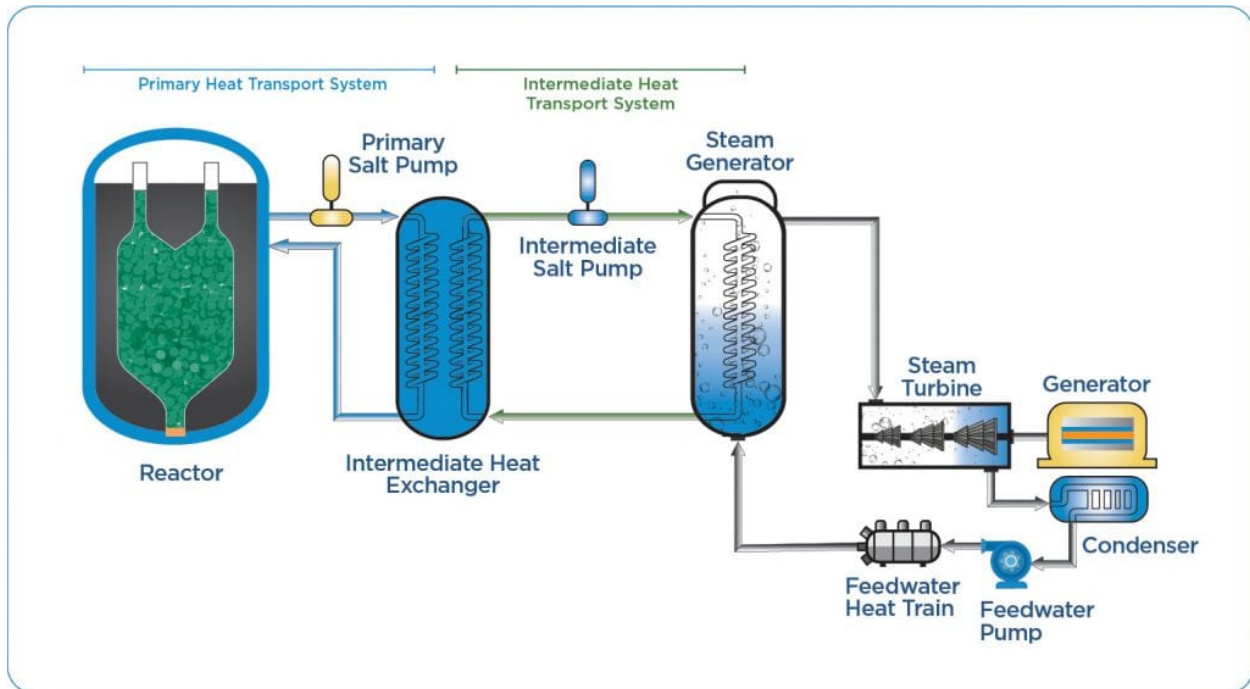
**Figure 3-2 Notional Design of FHR's TRISO-coated Particles Embedded in a Spherical Fuel Pebble (Kairos, 2020a).**



**Figure 3-3 Graphical Description of an FHR TRISO-coated Particle (Kairos, 2020a).**

The primary reactor coolant is a molten fluoride salt mixture commonly referred to as FLiBe ( ${}^7\text{LiF}:\text{BeF}_2$  at a nominal 2:1 mole ratio), which also provides retention of fission products that escape the TRISO fuel particle and pebble barriers. This is the same salt mixture used as the intermediate coolant in the MSRE.  $\text{LiF}_2$  generally requires a very high purity of  ${}^7\text{Li}$  since the other naturally occurring isotope  ${}^6\text{Li}$  has a high neutron absorption cross section.  $\text{BeF}_2$  is chemically toxic but is desired for its advantageous neutronic and physical properties.

Figure 3-4 provides a schematic of the conceptual design for the FHR. A primary coolant loop circulates the FLiBe salt mixture using pumps and transfers the heat to an intermediate coolant loop via a heat exchanger. The pumped flow intermediate coolant loop utilizes a “solar” nitrate salt compatible with reactor coolant, which transfers heat from the reactor coolant to the power conversion system through a steam generator. The design includes two decay heat removal systems. One normal decay heat removal system is used following normal shutdowns and anticipated operational occurrences. A separate passive decay heat removal system, which along with natural circulation in the reactor vessel, removes decay heat in response to a design-basis accident and does not rely on electrical power.



**Figure 3-4. Conceptual design of the FHR (Kairos, 2020b).**

Kairos Power has partnered with the Tennessee Valley Authority to eventually deploy a test reactor demonstration of the FHR (named the Hermes test reactor) at the East Tennessee Technology Park in Oak Ridge, Tennessee. The 35-MWth Hermes test reactor is being funded by the DOE Advanced Reactor Demonstration Program (ARDP) under the risk-reduction development pathway and is expected to be operational by 2026. Kairos Power has submitted a construction permit application for the Hermes test reactor and plans to submit an operating license application under 10 CFR Part 50. The Hermes test reactor is expected to be operational in 2026 (Haugh, 2021).

### **3.2.2 State of Knowledge on Fuel Enrichment, Fabrication, and Transportation Options**

A review of information submitted to the NRC has not identified specific details on plans for enrichment, fabrication, and transportation options under consideration for the FHR.

## **3.3 Flibe Energy – Lithium Fluoride Thorium Reactor**

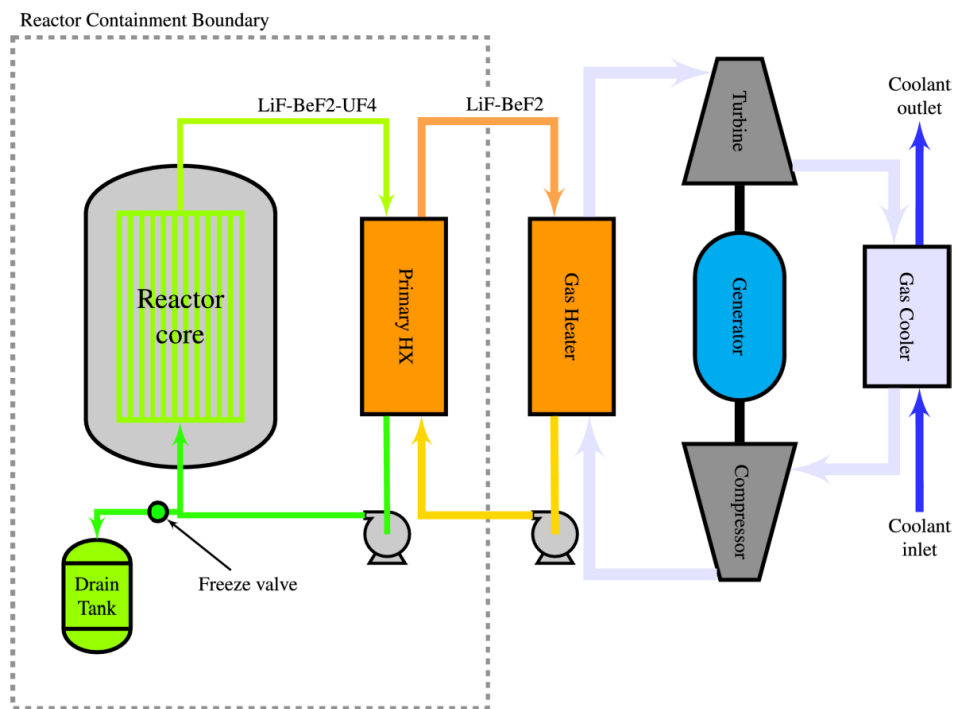
### **3.3.1 Technology Description and Status**

Flibe Energy's Lithium Fluoride Thorium Reactor (LFTR)<sup>1</sup>, designed for both power and isotope production, is a liquid-fueled, graphite-moderated thermal-spectrum breeder reactor optimized for operation on a thorium-supported <sup>233</sup>U fuel cycle (EPRI, 2015). The reference design has a

<sup>1</sup>The reactor was rebranded in 2021 as the "Lithium Fluoride Thorium Reactor (LFTR)", with "Lithium" replacing "Liquid".

nominal power output of 600 MWth and 250 MWe. The core, blanket, and primary cooling salt loops are all operated at or near atmospheric pressure and without steam for the blanket. The reactor is designed to “breed” fuel by converting  $^{232}\text{Th}$  in the blanket salt into fissile  $^{233}\text{U}$  through neutron absorption in an operational reactor fueled with existing fissile  $^{233}\text{U}$  material. Throughout the reactor lifecycle,  $^{233}\text{U}$  is extracted from the blanket salts and fed back into the reactor core to perpetuate the fuel production cycle (Harris, 2015).

Figure 3-5 below depicts the conceptual LFTR design and shows how the heat generated in the reactor core is transferred via heat exchangers to a clean secondary coolant salt loop that exits the containment boundary and is transferred to the working fluid of a gas turbine engine to generate power. The Flibe Energy LFTR design uses a supercritical carbon dioxide ( $\text{CO}_2$ ) Brayton cycle power conversion system to achieve greater thermal efficiency (EPRI, 2015).



**Figure 3-5 Conceptual Design of the LFTR (EPRI, 2015).**

Flibe Energy’s LFTR has the most unique fuel forms of near-term MSR designs since instead of the more traditional  $^{235}\text{U}$  fissile material,  $^{233}\text{U}$  is used in the primary fuel-salt and  $^{232}\text{Th}$  is used for the blanket salt that provides the subsequent fissile material. Flibe Energy has no needs for enrichment of  $^{233}\text{U}$ . The fissile and fertile materials for the fuel and blanket salts are in the form of  $\text{UF}_4$  and  $\text{ThF}_4$ , respectively. The fuel-salt and blanket salt use FLiBe and consist of  $2\text{LiF}_2\text{-BeF}_2\text{-XF}_4$ , where “X” is  $^{233}\text{U}$  for the fuel and  $^{232}\text{Th}$  for the blanket (EPRI, 2015). FLiBe is a chemically stable, ionically bonded salt that mitigates radiation damage and retains a majority of fission products by reacting to form stable fluoride salts that are retained in the mixture under normal operations and accident conditions (Flibe, 2021).

Flibe Energy and Pacific Northwest National Laboratory were funded under DOE's Industry Funding Opportunity Announcement (FOA) to study the use of nitrogen trifluoride to remove uranium from the molten salt-fuel mixtures as a preliminary step for fission product removal.

### 3.3.2 State of Knowledge on Fuel Enrichment, Fabrication, and Transportation Options

A review of information submitted to the NRC has not identified specific details on plans regarding fabrication or transportation options under consideration for the LFTR. Flibe Energy has no needs for enrichment of  $^{233}\text{U}$ .

## 3.4 TerraPower / Southern Nuclear – Molten Chloride Fast Reactor

### 3.4.1 Technology Description and Status

TerraPower's Molten Chloride Fast Reactor (MCFR) is a fast-spectrum molten salt reactor with a molten chloride salt serving as both the fuel and the coolant. The MCFR demonstration reactor is designed as a small to mid-scale reactor with an output of 30-300 MWe, and the final commercial grid-scale reactor will produce electricity in the range of 500-1200 MWe (Latkowski, 2021). The fast spectrum improves the neutron economy and largely mitigates the fission product poisoning that normally necessitates online reprocessing capabilities.

Figure 3-6 shows the conceptual design for the MCFR. The reflector in this design can potentially change throughout the fuel cycle, beginning with a heavier fast neutron reflector (e.g., lead, lead oxide, lead bismuth, tungsten) and shifting to include a lighter neutron reflector/moderator (e.g., graphite, beryllium, lithium, zirconium hydride, non-graphite carbon) (TerraPower, 2020).

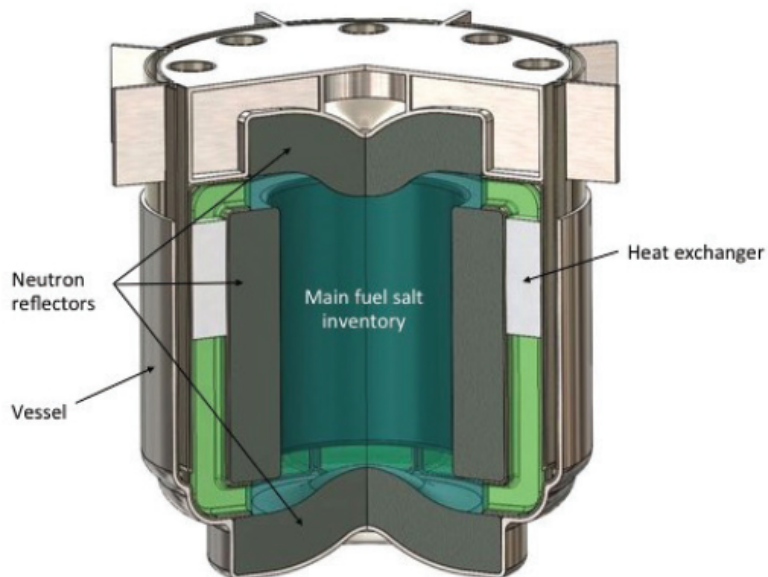


Figure 3-6 Conceptual Design of the MCFR (TerraPower, 2021).

In December 2020, the U.S. Department of Energy selected the Molten Chloride Reactor Experiment (MCRE) proposal, with Southern Company as the Prime, as a winner of the Advanced Reactor Demonstration Program risk-reduction pathway. This effort is relevant to TerraPower's MCFR design.

Southern Company and TerraPower are working on an Integrated Effects Test (IET) to learn how the MCFR technology will scale and behave at larger, commercially relevant sizes. The IET is expected to be commissioned and begin operating in TerraPower's Everett, Washington, facility in 2022.

### **3.4.2 State of Knowledge on Fuel Enrichment, Fabrication, and Transportation Options**

A review of information submitted to the NRC has not identified specific details on plans for enrichment, fabrication, and transportation options under consideration for the MCFR.

## **3.5 Abilene Christian University – Molten Salt Research Reactor**

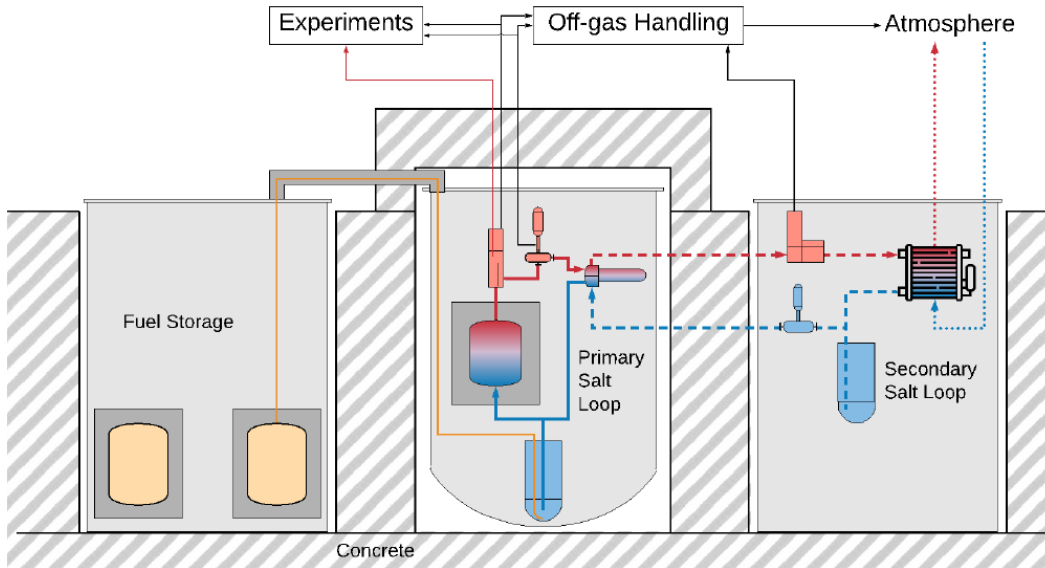
### **3.5.1 Technology Description and Status**

The Abilene Christian University (ACU) Molten Salt Research Reactor (MSRR) is a graphite-moderated fluoride salt test reactor designed for a thermal power of 1 MWth. The reactor primary system will be configured in a loop, with a reactor vessel, a heat exchanger, a fuel pump, and experimental access points. A secondary loop will be used to remove the reactor thermal power, and the final heat sink will be air (Towell, 2020).

The MSRR is essentially a simplified version of the MSRE operated by ORNL. The major differences of the MSRR relative to the MSRE is that the reactor will be fueled with HALEU instead of high enriched uranium (HEU), it will have a lower power density, and it will not need cooling water within the containment (Towell, 2020).

Figure 3-7 shows a conceptual design for the MSRR.

The reactor containment vessel will be approximately 20 feet tall and 10 feet in diameter. The reactor vessel inside the containment vessel will be approximately 6 feet tall and 4.5 feet in diameter and will be surrounded by a 6-inch internal concrete shield. The reactor containment vessel is located in a below-grade trench within the facility. The lower reactor containment vessel head is surrounded by a small pool of water to remove decay heat under normal and accident conditions including loss of normal electrical power (Schubert, 2020).



**Figure 3-7 Conceptual design of the MSRR (Towell, 2020).**

The baseline fuel-salt composition for the MSRR is a mixture of lithium fluoride (LiF), beryllium fluoride (BeF<sub>2</sub>), and HALEU fuel in the form of uranium tetrafluoride (UF<sub>4</sub>). The lithium will be enriched to greater than 99.99 percent <sup>7</sup>Li. The MSRR will use approximately 300 kg of nearly 20 percent (~19.9 percent) HALEU fuel (Schubert, 2020).

ACU has completed Phase 1 and is in Phase 2 of their project, which consists of designing and building a large, advanced molten salt system with advanced instrumentation and monitoring capabilities for the testing of the individual components that will be used in the MSRR (Towell, 2020).

The MSRR project is being carried out under a cooperative fuel agreement with the U.S. Department of Energy's Research Reactor Infrastructure Program (RRIP) and supported through a collaboration with major U.S. industry partners, national laboratories, and universities. Natura Resources, LLC, is partnering in a \$30.5 million effort to design and license the research reactor in collaboration with a consortium of universities that includes ACU, University of Texas, Texas A&M University, and Georgia Institute of Technology (Schubert, 2020).

### **3.5.2 State of Knowledge on Fuel Enrichment, Fabrication, and Transportation Options**

ACU plans on the fuel coming on-site premixed and ready to be loaded into the core (Schubert, 2020). However, details on their plans for sourcing or transportation to the site have not been provided.

## 4 TECHNICAL AND REGULATORY CONSIDERATIONS – FRONT-END ACTIVITIES FOR MOLTEN SALT REACTORS

The enrichment, production and transportation of fuel and salt materials to support operation of near-term molten salt reactors (MSRs) present distinct technical and regulatory challenges relative to uranium oxide based fuels used in current light-water reactor technologies. The fuel-salt mixtures will be a combination of fissile and/or fertile materials of LEU or other isotopic ( $^{233}\text{U}$ ,  $^{232}\text{Th}$ ) compositions with non-radioactive diluent or carrier salts. Fluoride salt-fueled reactors are expected to operate with uranium tetrafluoride ( $\text{UF}_4$ ) and thorium fluoride ( $\text{ThF}_4$ ), if fertile material is used, dissolved in molten fluoride coolant salts of various composition. Similarly, the chloride salt-fueled reactors could potentially operate on uranium trichloride ( $\text{UCl}_3$ ) in various chloride salts. Fluoride salt-cooled reactors are designed to operate with a fluoride salt-based coolant but with a solid pebble fuel based on tristructural isotropic (TRISO) particles.

This section provides a preliminary assessment of technical and regulatory considerations for the enrichment, production, and transportation activities necessary to support near-term MSR technologies discussed in Section 3 of this report. The discussion provides a description of potential methods and approaches for these activities, associated safety considerations, and an initial assessment of the regulatory frameworks to support approval of these approaches by the U.S. Nuclear Regulatory Commission (NRC).

### 4.1 Technical Considerations for Fuel Enrichment, Production, and Transportation Methods

#### 4.1.1 Uranium Tetrafluoride ( $\text{UF}_4$ )

##### 4.1.1.1 $^{235}\text{U}$ Enrichment

Near-term MSR technologies are based on LEU fuel compositions (i.e., uranium weight percent of  $^{235}\text{U}$  less than 20 percent, as defined in 10 CFR Part 50). Three of the technologies will need higher enrichments than current light-water reactor solid oxide fuels (i.e., uranium weight percent of  $^{235}\text{U}$  exceeding 5 percent), which is commonly referred to as high-assay low-enriched uranium (HALEU).

To support industry's needs, the U.S. DOE established the HALEU Demonstration Program in collaboration with Centrus Energy, to deploy a small cascade of AC-100M enrichment centrifuges to demonstrate the capability to produce HALEU. In June 2021, the NRC approved a license amendment request for their American Centrifuge Plant at Piketon, Ohio, to begin production of LEU in early 2022. The technical and regulatory considerations for the gas centrifuge enrichment process have been addressed in the safety assessment of that application and are not considered further in this report (NRC, 2021).

DOE and its national laboratories are also exploring various options for the production of HALEU fuel feed material, including two chemical processes (Electrochemical Processing,



Hybrid Zirconium Extraction Process). Both methods involve the recycling of spent nuclear fuel from DOE research reactors to recover highly enriched uranium (greater than 20 percent) that can then be down blended to make HALEU fuel feed material. A follow-on assessment may evaluate technical and regulatory considerations related to these DOE processes, if they were to be implemented by a commercial facility subject to NRC purview.

#### **4.1.1.2 <sup>233</sup>U Production**

<sup>233</sup>U is a fissile isotope of uranium that is produced either in a reactor, and subsequently isotopically purified, or with an accelerator by bombarding <sup>232</sup>Th with neutrons. Approximately two tons of <sup>233</sup>U was produced between 1954 and 1970 in various weapons material production reactors and commercial nuclear power plants. A majority of the U.S. inventory of <sup>233</sup>U is stored at Oak Ridge National Laboratory (ORNL) in the form of oxides (e.g., UO<sub>2</sub>, U<sub>3</sub>O<sub>8</sub>) and Idaho National Laboratory in the form of fuel elements. The approximately 500 kg of <sup>233</sup>U in storage at ORNL has been gradually down blended and mixed with a grout for transportation and disposal (Peretz, 1996).

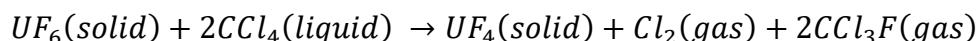
#### **4.1.1.3 UF<sub>4</sub> Production**

All commercial <sup>235</sup>U enrichment plants use uranium hexafluoride (UF<sub>6</sub>) as the feed material for reactor fuel productions, so that is likely the starting compound for producing both LEU and HALEU UF<sub>4</sub> for molten salt reactors. Chemical methods used for conversion of UF<sub>6</sub> to UF<sub>4</sub> are available (IAEA, 1994; McFarlane et al., 2019), including direct conversion reactions as well as indirect conversion through the initial production of UO<sub>2</sub> from UF<sub>6</sub> (as is currently done for the production of light-water reactor fuel forms). Currently it is not clear which of these processes would be chosen for large-scale production of enriched UF<sub>4</sub> from UF<sub>6</sub> for near-term MSR technologies. The conversion of UF<sub>6</sub> to UF<sub>4</sub> has been previously demonstrated to produce metallic fuels, and potential alternative approaches for the production of UF<sub>4</sub> from spent nuclear fuel have also been proposed.

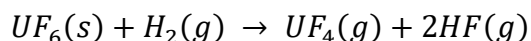
##### **4.1.1.3.1 Methods for direct conversion of UF<sub>6</sub> to UF<sub>4</sub>**

Two methods implemented on a commercial scale involve the conversion of UF<sub>6</sub> to UF<sub>4</sub> via carbon tetrachloride (CCl<sub>4</sub>) reduction or hydrogen (H<sub>2</sub>) reduction (McFarlane et al., 2019). Other processes have been tested on a small scale and are not discussed here. These methods involve reactants such as nitrogen oxides (NO<sub>x</sub>), hydrogen bromide (HBr), hydrogen chloride (HCl) or hydrogen iodide (HI).

The reduction of UF<sub>6</sub> with CCl<sub>4</sub> produces UF<sub>4</sub> at 120 – 150°C with gaseous effluent byproducts of chlorine (Cl<sub>2</sub>), freons (CCl<sub>x</sub>F<sub>(4-x)</sub>), and excess CCl<sub>4</sub> (McFarlane et al., 2019). All these byproducts are toxic and corrosive, and the occupational permissible exposure limits (PELs) per the Occupational Safety and Health Administration (OSHA) are provided in Table 4-2. The reaction is exothermic and a resultant pressure buildup must be managed (McFarlane et al., 2019).



The hydrogen reduction process involves the reaction of UF<sub>6</sub> with H<sub>2</sub> gas to form UF<sub>4</sub> and hydrofluoric acid (HF) at elevated temperatures (1,200-1,700°C) in a monel reactor (IAEA, 1994; Aleksandrov et al., 2016; McFarlane et al., 2019). The reaction is exothermic, and the HF gas product is toxic and highly corrosive. The generation of HF gas must be managed per OSHA PELs (Table 4-2) and should be evaluated for compatibility with the materials used in the process.



IAEA (1994) discusses process and equipment considerations for the hydrogen reduction process. UF<sub>6</sub> cylinders are heated in autoclaves that provide a controlled vaporized UF<sub>6</sub> feed to a surge vessel ahead of the reduction reactors. Nitrogen gas (N<sub>2</sub>) is used to purge the cylinder discharge lines and reactor feed lines. Hydrogen gas is fed to each cylindrical reactor and mixed with UF<sub>6</sub> in a conical section designed to minimize slag buildup on the reactor walls. The reactor heating profile (480 °C [896 °F] (bottom) – 650 °C [1202 °F] (top)) is maintained by clam-shell zone electrical heaters. Compressed air is supplied to each zone for rapid cool-down or shutdown of the reactors.

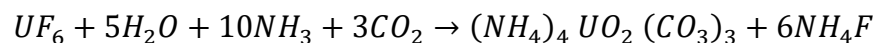
The UF<sub>4</sub> and gaseous reaction products pass from the bottom of the reactor to a screw-type cooling conveyor and later to a pulverizer. The process must be controlled to avoid the production and accumulation of UF<sub>x</sub>, where 4 < x < 5, which can deposit on the process equipment. Removal of these products may require mechanical removal, which would generate radioactive particulates and aerosols.

An off-gas and particulate management system is necessary to manage or allow reuse of process products, including HF, N<sub>2</sub>, H<sub>2</sub>, residual UF<sub>6</sub>, and fine particulates of UF<sub>4</sub>. UF<sub>4</sub> particulates are captured in sintered metal filters and residual UF<sub>6</sub> is captured in activated charcoal traps. The off-gas stream then runs through chemical traps for HF, N<sub>2</sub> and H<sub>2</sub>, and is passed through a circuit consisting of a pre-cooler, a partial condenser, and finally a total condenser to remove the HF. The HF is collected in storage tanks and may be recycled to a UF<sub>6</sub> conversion plant. Recovered H<sub>2</sub> and N<sub>2</sub> is discharged to atmosphere via a mist eliminator and a neutralization scrubber.

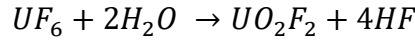
#### 4.1.1.3.2 Methods for indirect conversion of UF<sub>6</sub> to UF<sub>4</sub> (UO<sub>2</sub> intermediate)

Various approaches could be used for the indirect production of UF<sub>4</sub> from UF<sub>6</sub> via the intermediate UO<sub>2</sub>. Five different methods of converting UF<sub>6</sub> to UO<sub>2</sub> are discussed in detail in IAEA (1994). These UF<sub>6</sub> conversion approaches are as follows:

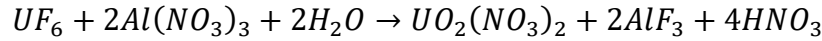
- 1) Precipitation of ammonium uranyl carbonate ((NH<sub>4</sub>)<sub>4</sub>UO<sub>2</sub>(CO<sub>3</sub>)<sub>3</sub>, AUC) where UF<sub>6</sub> is fed into a stirred aqueous system with gaseous CO<sub>2</sub> and NH<sub>3</sub>. The AUC precipitate is filtered out and later decomposed in a fluidized-bed reactor to produce UO<sub>2</sub>.



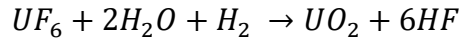
- 2) Production of the intermediate ammonium diuranate ((NH<sub>4</sub>)<sub>2</sub>U<sub>2</sub>O<sub>7</sub>, ADU), where ADU is produced per the following reactions. The ADU precipitate is subsequently filtered and washed with hot water and then converted to UO<sub>2</sub> using hydrogen or cracked ammonia.



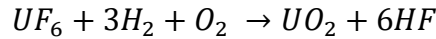
In a modified ADU conversion process, the first step is replaced by hydrolysis of UF<sub>6</sub> in aluminum nitrate solution (see below) and followed by total phenolic solvent extraction.



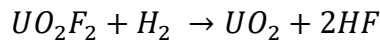
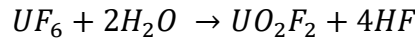
- 3) In the integrated dry route process method, UF<sub>6</sub> is converted directly to a ceramic grade UO<sub>2</sub> powder by reaction of UF<sub>6</sub> vapor with steam and hydrogen in the following overall reaction:



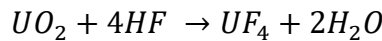
- 4) In this method developed by the General Electric Company, vaporized UF<sub>6</sub> is fed into a flame reactor to produce U<sub>3</sub>O<sub>8</sub> and the U<sub>3</sub>O<sub>8</sub> is reduced in a kiln to produce UO<sub>2</sub> in the summarized reaction below.



- 5) In the final method, known as the fluidized-bed process, UF<sub>6</sub> is converted to UO<sub>2</sub> by steam and hydrogen via uranyl fluoride (UO<sub>2</sub>F<sub>2</sub>) at multiple fluidized-bed reactors. Dry scrubbers use calcium hydroxide to recover HF from the effluent gas in the form of calcium fluoride. The intermediate stage and overall reactions are shown below.



Once enriched UO<sub>2</sub> powder is produced using any of the above methods (or any not listed), the UO<sub>2</sub> powder can be reacted with pure HF gas at elevated temperatures to produce enriched UF<sub>4</sub> per the following reaction (Souček et al., 2017):



Care must be exercised in this final conversion step to ensure an inert environment to avoid any undesired reactions with moisture and oxygen from the ambient atmosphere. Additionally, the uranium oxide powder should be prepared by reducing any U(VI) to U(IV) and facilitating the highest possible surface area to enhance the kinetics of the fluorination reaction.

#### 4.1.1.4 *UF<sub>4</sub> Transportation*

A preliminary review of transportation packages on the RAMPAC (Radioactive Materials Packages) database was performed to identify transportation packages approved by NRC, DOE, and U.S. Department of Transportation (DOT) for shipment of UF<sub>4</sub>.

The NRC has not yet approved transportation packages for transport of UF<sub>4</sub> under domestic use, i.e., no specific approvals per the regulatory requirements of 10 CFR Part 71. However, the NRC has approved two transportation packages for import and export from outside the United States under revalidations of approvals by foreign competent authorities. These revalidations are conducted under a Memorandum of Understanding with DOT, the Competent Authority for import and export transport of radioactive materials for the United States. The associated safety reviews are conducted per IAEA safety requirements, which are mostly consistent with 10 CFR Part 71. A brief summary of these revalidations follows:

- *DAHER Nuclear Technologies GmbH, Transportation Package Model No. BU-D*  
On May 3, 2019, the DOT issued a certificate of revalidation for German Certificate of Approval No. D/4305/AF-96, Revision 10 for Model No. BU-D (DOT, 2019a). The certificate of revalidation expires on January 31, 2024. The safety review was conducted per IAEA Safety Standards Regulations for the Safe Transport of Radioactive Material, Specific Safety Requirements No. SSR-6 (IAEA, 2012). The transportation package allows for transport of depleted, natural and/or enriched UF<sub>4</sub> with a maximum enrichment (<sup>235</sup>U) of 5 percent. The maximum allowed mass content of <sup>235</sup>U in the depleted, natural, and enriched uranium is 0.8 kg.
- *TN International, Transportation Package Model No. TN-BGC1*  
On June 13, 2019, the DOT issued a certificate of revalidation for French Certificate of Competent Authority F/313/B(U)F-96, Revision 20 for Model No. TN-BGC-1 (DOT, 2019b). The certificate of revalidation expires on March 1, 2024. The safety review was conducted per IAEA Safety Standards Regulations for the Safe Transport of Radioactive Material, Specific Safety Requirements No. SSR-6 (IAEA, 2012). The transportation package allows for transport of UF<sub>4</sub> in multiple configurations with a maximum enrichment (<sup>235</sup>U) of 30 percent. The maximum allowed mass content of <sup>235</sup>U is dependent on the containment diameter, the presence of hydrogen-bearing materials with higher hydrogen quantity than water, and the number of packages in an array.

Consistent with 49 CFR 173.7(d), the DOT may also grant authorization to the DOE for the use of transportation packages for radioactive materials made by or under the direction of DOE if evaluated against equivalent regulations to 10 CFR Part 71. Under this authorization, DOE's Packaging Certification Program (PCP) has approved one transportation package for UF<sub>4</sub>:

- *DOE, National Nuclear Security Administration, Y-12 National Security Complex, Transportation Package Model No. ES-3100*  
On April 22, 2021, DOE's PCP issued Revision 18 of Certificate of Compliance No. 9315 for the Model No. ES-3100 (DOE, 2021). The transportation package allows for transport of UF<sub>4</sub> with a maximum enrichment (<sup>235</sup>U) of 100 percent. The maximum allowed mass

content of  $^{235}\text{U}$  is approximately 18 kg depending on the Criticality Safety Index of the package.

The above approvals would provide limited flexibilities for the transportation of  $\text{UF}_4$  to a reactor site. The NRC revalidations of approvals by foreign competent authorities would require initiation of transport from a foreign facility. Further, approvals under the DOE PCP would require initiation of transport from a DOE facility and receipt at a reactor locality also located in a DOE facility. Therefore, the above approvals would not allow for the transportation of  $\text{UF}_4$  from a domestic commercial facility (licensed under 10 CFR Part 70) to a commercial reactor site. New NRC-approvals are likely necessary to support this scenario.

It is also important to note that the transport of  $\text{UF}_4$  enriched with  $^{233}\text{U}$  will likely require approval of a Type B transportation package due to the limited quantity acceptable for shipment in a Type A transportation package per Table A-1 in Appendix A of 10 CFR 71. Therefore, transportation packages for  $\text{UF}_4$  enriched with  $^{233}\text{U}$  will likely be subject to more rigorous test requirements than for  $\text{UF}_4$  enriched with  $^{235}\text{U}$ .

The safe transportation of  $\text{UF}_4$  is not expected to involve new hazards relative to the transportation of  $\text{UF}_6$ , which is regularly performed in standardized cylinders (ANSI, 2019).  $\text{UF}_4$  is not expected to undergo thermal decomposition during the thermal test for hypothetical accident conditions.  $\text{UF}_4$  will exothermically react with water, albeit slower than  $\text{UF}_6$ , for the production of HF and uranyl fluoride ( $\text{UO}_2\text{F}_2$ ). The primary considerations for the safety hazards will involve ensuring compatibility with materials used for primary containment of  $\text{UF}_4$  in a transportation package and maintaining an adequate subcriticality margin for hypothetical accident conditions. Additionally, loading operations will require ensuring that an inert atmosphere is maintained in the primary containment and transport operations that preclude the ingress and interaction of water and air with the  $\text{UF}_4$  contents.

#### **4.1.2 Uranium Trichloride ( $\text{UCl}_3$ )**

##### **4.1.2.1 Chlorine Isotopic Enrichment**

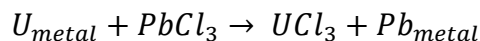
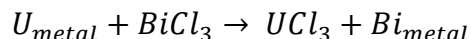
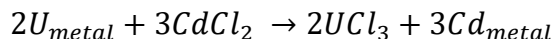
Molten salt reactor vendors planning to use chlorine-based fuel materials (e.g.,  $\text{UCl}_3$ ) would need to consider purifying the chlorine to increase the  $^{37}\text{Cl}$ -to- $^{35}\text{Cl}$  ratio to decrease the production of  $^{36}\text{Cl}$  (long half-life beta emitter) and  $^{35}\text{S}$  that increases corrosion.  $^{37}\text{Cl}$  has a natural abundance of 24.23 percent compared to 75.77 percent for  $^{35}\text{Cl}$ . Although no commercial scale chlorine purification method has been established, several laboratory-scale methods have been demonstrated that could be scaled. Among these methods are gas-phase isotopic separation, liquid-phase thermal diffusion, room-temperature anion exchange chromatography, and NaCl crystallization (McFarlane et al., 2019). These purification processes are expected to take place in a commercial facility, outside the purview of the NRC.

##### **4.1.2.2 $\text{UCl}_3$ Production**

Uranium forms four chlorides (i.e.,  $\text{UCl}_6$ ,  $\text{UCl}_5$ ,  $\text{UCl}_4$ ,  $\text{UCl}_3$ ) depending on its oxidation state. It is feasible that  $\text{UCl}_3$  could be employed in a molten salt fast reactor.  $\text{UCl}_3$  has been most

commonly produced from the precursor  $UCl_4$ , which is the most stable of the uranium chlorides.  $UCl_4$  can be produced by chlorinating uranium oxides or fluorides, such as  $UF_6$  (most common since this is the general form after enrichment) or  $UF_4$ . Effective chlorinating agents include  $CCl_4$ , hexachloropropene ( $C_3Cl_6$ ),  $Cl_2$ , phosgene ( $COCl_2$ ), or mixtures of carbon monoxide ( $CO$ ) with  $Cl_2$ . Each of these chlorinating agents are hazardous materials of varying levels of toxicity that must be handled and stored under appropriate controls. Once  $UCl_4$  is formed, it readily reduces to  $UCl_3$  when reacted with hydrogen ( $H_2$ ) at high temperature ( $550^\circ C$ ). McFarlane et al. (2019) provide additional details on the production methods for the precursor  $UCl_4$ .

$UCl_3$  has also been prepared in kilogram quantity by a two-step process involving the conversion of pure uranium metal (cleaned in nitric acid) with  $H_2$  to produce uranium hydride ( $UH_3$ ) at  $250^\circ C$ , followed by reaction with  $HCl$  gas (Johnson et al., 1944). Alternatively,  $UCl_3$  can be formed directly through reaction of pure uranium metal with a metal chloride such as  $CdCl_2$  (Yoo et al., 2018),  $BiCl_3$  (Lambert et al., 2018), or  $PbCl_3$  (Bradley, 1957)) as shown by the below representative reactions. These methods are much less likely to be used since enriched uranium is generally in the form of uranium oxides or fluorides.



#### 4.1.2.3 $UCl_3$ Transportation

A preliminary review of transportation packages on the RAMPAC site certified by the NRC, DOE, and DOT did not identify transportation packages approved for shipment of  $UCl_3$ .

The safe transportation of  $UCl_3$  will require controls for loading and transport operations to maintain the contents in the same oxidation state and preclude ingress and interaction of water or air with the contents.  $UCl_3$  has a melting point of  $842 \pm 5^\circ C$  and decomposes at temperatures above  $850^\circ C$  to produce  $UCl_4$ . Since transportation of  $UCl_3$  is likely to occur in Type A fissile packages, these temperatures are unlikely to occur during normal conditions of transport.  $UCl_3$  will react with water to produce hydrogen, uranium dioxide, and  $HCl$ . The material has also been reported to exothermically react with oxygen at  $150-200^\circ C$  to produce uranyl chloride ( $UO_2Cl_2$ ),  $Cl_2$  and  $UO_2$  (Johnson et al., 1944). The primary considerations for safety hazards will involve ensuring compatibility with materials used for primary containment of  $UCl_3$  in a transportation package and implementing loading operations that ensure an inert atmosphere is maintained for the  $UCl_3$  contents.

### 4.1.3 Thorium Fluoride ( $ThF_4$ )

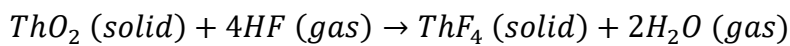
#### 4.1.3.1 $ThF_4$ Production

$^{232}Th$  has been proposed for use as a fertile material for Flibe Energy's Lithium Fluoride Thorium Reactor (LTFR), in which  $^{232}Th$  would transmute to  $^{233}U$  to sustain the fission reaction.  $^{232}Th$  is a

natural-occurring weakly radioactive isotope with a natural isotopic abundance of 99.98 percent  $^{232}\text{Th}$  and a half-life of approximately 14 billion years. Thorium is estimated to be over three times as abundant as uranium in the earth's crust and is currently primarily recovered from the mineral monazite (a rare-earth phosphate mineral) as a byproduct of processing heavy mineral sand deposits for titanium-, zirconium-, or tin-bearing minerals (IAEA, 2005). It is unclear whether the LTFR blanket fuel will require higher purity of  $^{232}\text{Th}$  relative to its natural isotopic abundance.

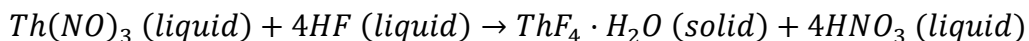
Significant experience exists on the production of thorium oxide or thoria ( $\text{ThO}_2$ ) used in nuclear fuels for operating water-cooled reactors (NEA & IAEA, 2016; AEC, 1969; IAEA, 2005). However,  $\text{ThO}_2$  is highly insoluble in lithium-beryllium fluoride mixtures and therefore not suitable for use in MSR. Instead,  $\text{ThF}_4$  has been proposed for use as the fertile blanket fuel in the Flibe Energy LFTR design (EPRI, 2015).

$\text{ThF}_4$  is currently produced and used in optical coatings for antireflective, bandpass and dichroic filters, and was previously used in carbon arc lamps.  $\text{ThF}_4$  can be synthesized using various techniques involving the reaction of  $\text{ThO}_2$  with hydrogen fluoride gas, aqueous hydrofluoric acid, ammonium fluoride ( $\text{NH}_4\text{F}$ ), or ammonium bifluoride ( $\text{NH}_4\text{HF}_2$ ) (Bahri et al., 2019). The most convenient of these methods is the heating of  $\text{ThO}_2$  in excess anhydrous hydrogen fluoride gas at temperatures of 400 – 550 °C and near atmospheric pressure, per the following chemical reaction (Souček et al., 2017; Briggs et al., 1971).



This method requires the elimination of moisture, which may result in the formation of thorium oxyhydrates or oxyfluorides.

Another aqueous-based method involves the precipitation of  $\text{ThF}_4$  from the reaction of thorium nitrate ( $\text{Th}(\text{NO}_3)_3$ ) and hydrofluoric acid in a semi-continuous process in polymer-based equipment, per the following chemical reaction (Briggs et al., 1971).

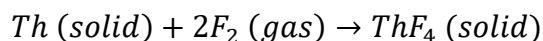


After recovery of the hydrated  $\text{ThF}_4$  by filtration, the material is dried at 350°F in air to a moisture content of 1 or 2 percent.  $\text{ThF}_4$  is then thoroughly dehydrated in batch tray retorts of Inconel construction by heating in the range 800° to 1000°F under very small flows of nitrogen and anhydrous HF to prevent oxidation or pyrohydrolysis. The dehydration process is known to be limited by the formation of thorium hydroxide ( $\text{Th}(\text{OH})_4$ ) or thorium oxyfluoride ( $\text{ThOF}_2$ ) (Bahri et al., 2019).

Fluorination of  $\text{ThO}_2$  with  $\text{NH}_4\text{F}$  and  $\text{NH}_4\text{HF}_2$  has also been studied. The reaction with  $\text{NH}_4\text{F}$  yields a tetravalent fluoro-complex  $\text{NH}_4\text{ThF}_6$ , which decomposes to  $\text{ThF}_4$  at temperatures above 350 °C (Bahri et al., 2019). The reaction with  $\text{NH}_4\text{HF}_2$  yields, in part,  $(\text{NH}_4)_3\text{ThF}_7$ , which decomposes to  $\text{ThF}_4$  at temperatures above 450 °C. Both methods require high ratios of the fluorinating agents, i.e. eight-fold excess or higher. While these fluorinating agents are easier to

handle than HF, the need for high amounts of fluorinating agents may hinder deployment of these methods in large-scale.

Direct conversion to ThF<sub>4</sub> by the reaction of thorium metal with F<sub>2</sub> gas, per the following equation, is also possible. A follow-on assessment to this report may include a thorough review of the viability and process conditions for this process.



It is important to note that the use of ThF<sub>4</sub> in an MSR design was previously explored but not implemented in an operating reactor. In the late 1960s, the Molten Salt Breeder Reactor (MSBR) program, a follow-on activity to the Molten Salt Reactor Experiment at Oak Ridge National Laboratory (ORNL), pursued the design of a thermal-spectrum breeder MSR with online continuous monitoring, which was to operate on a <sup>233</sup>U-<sup>232</sup>Th fuel cycle (AEC, 1972). The system was designed to operate with a fuel-salt mixture of ThF<sub>4</sub> (12 mole percent) and UF<sub>4</sub> (0.3 mole percent) in <sup>7</sup>LiF (72 mole percent) and BeF<sub>2</sub> (16 mole percent). The reactor was never constructed and operated. A preliminary review of the literature has not provided details on the methods that were proposed for the synthesis of ThF<sub>4</sub> for the MSBR program. The fluorination of thorium was also studied under the Manhattan Project (Sorenson, 2014) but information on those activities is not readily available.

It is also important to note that the People's Republic of China will begin operations in the near-term of a liquid fluoride thorium reactor (Mallapaty, 2021). The reactor will produce two megawatts of thermal energy and will be operated by the Shanghai Institute of Applied Physics at Wuwei, on the outskirts of the Gobi Desert (Song, 2012). A preliminary review of open literature has not identified details related to the production or transportation approaches used for supplying ThF<sub>4</sub> to that reactor.

#### **4.1.3.2 ThF<sub>4</sub> Transportation**

A preliminary review of transportation packages on the RAMPAC site certified by the DOE, NRC, and DOT did not identify transportation packages approved for shipment of ThF<sub>4</sub>. Transportation is likely to occur in a Type A package.

The safe transportation of ThF<sub>4</sub> is expected to involve similar hazards to the transportation of UF<sub>4</sub>. ThF<sub>4</sub> is not expected to undergo thermal decomposition during temperatures corresponding to normal conditions of transport for the expected shipment in Type A packages.

ThF<sub>4</sub> is reported to be significantly more stable than ThO<sub>2</sub> upon comparison of their relative heats of formation (ΔH<sub>f</sub>) (Baes, 1974). Therefore, reaction of ThF<sub>4</sub> in the presence of oxygen and moisture (e.g., production of ThOF<sub>2</sub>) during transport conditions is expected to be limited. The primary considerations for safety hazards will involve ensuring compatibility with materials used for primary containment of ThF<sub>4</sub> in a transportation package and implementing loading operations that ensure an inert atmosphere is maintained to maintain ThF<sub>4</sub> purity (IBI, 2014).

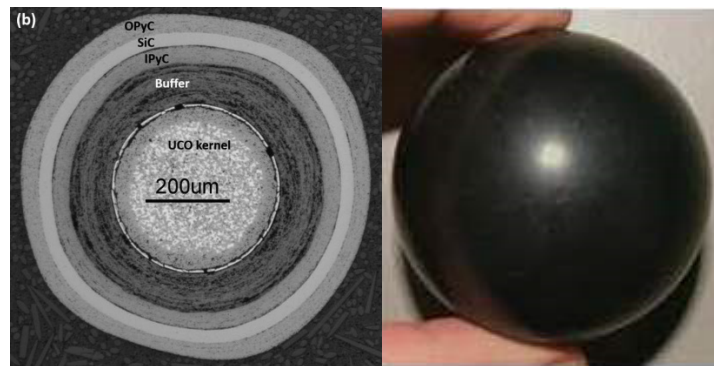


#### 4.1.4 Tristructural Isotropic (TRISO) Solid Fuel

##### 4.1.4.1 TRISO Production

TRISO-coated fuel particles and pebbles have been produced and used in high temperature gas reactors (HTGRs) operating under pressurized helium with coolant outlet temperatures in the range of 700 to 950 °C (Demkowicz et al., 2019; IAEA, 2010; Richards, 2002). Historically, a broad range of TRISO fuel particles have been fabricated and tested around the world, including compositions based on uranium oxide ( $\text{UO}_2$ ), uranium-thorium oxide ((U,Th) $\text{O}_2$ ), uranium dicarbide ( $\text{UC}_2$ ), uranium-thorium carbide ((U,Th)C), plutonium oxide ( $\text{PuO}_2$ ) and a heterogeneous mixture of uranium oxide and uranium carbide (UCO). TRISO fuel pebbles are being considered for the FHR under development by Kairos Power.

TRISO-coated fuel pebbles consist of fuel particles with fissile or fertile material (UCO being proposed for the KP-FHR) with diameters in the range of 350–600  $\mu\text{m}$  (Figure 4-1) (Demkowicz et al., 2019). Sol-gel synthesis processes are the most common for the fabrication of the fuel kernels. In this method, sol-gel spheres are made using internal or external gelation methods. For example, uranium is dissolved in nitric acid ( $\text{HNO}_3$ ) and ammonia ( $\text{NH}_3$ ) is used to precipitate a metal oxide from the sol (e.g., uranium trioxide ( $\text{UO}_3$ )) and form a gel sphere as the droplets pass through a heated liquid medium (Demkowicz et al., 2019). Alternatively, uranyl nitrate ( $\text{UO}_2(\text{NO}_3)_2$ ) can be reacted in a high-pH (basic) solution of hexamethylenetetramine ( $\text{C}_6\text{H}_{12}\text{N}_4$ ) and urea ( $\text{CH}_4\text{N}_2\text{O}$ ) to produce uranyl hydroxide ( $\text{UO}(\text{OH})_2$ ) gel spheres (EPRI, 2020). The gel spheres are washed, dried, and heated in appropriate atmospheres to drive off byproducts of the sol-gel conversion and form a final sintered  $\text{UO}_2$  kernel. Carbon can be dispersed in the sol to produce uranium carbide within the kernel.



**Figure 4-1 UCO TRISO Particle (left) (Demkowicz et al., 2019) and Fuel Pebble (right) (Mitchell et al., 2020).**

Each  $\text{UO}_2$  or UCO kernel is generally coated by four different layers applied using fluidized-bed chemical vapor deposition (Figure 3-3). In this process, kernels are fluidized with argon and reactant gases that decompose and deposit on the kernels. An initial porous (approximately 50 percent of theoretical density) carbon buffer layer on the kernel is produced using acetylene ( $\text{C}_2\text{H}_2$ ). This layer provides void space for fission gases released from the kernel and accommodates fission recoils. A subsequent layer of dense, highly isotropic inner pyrocarbon

(IPyC) layer is then deposited by mixing propylene ( $C_3H_6$ ) with acetylene and argon. This layer contributes to the retention of fission gases in the particle, as well as acts as a seal layer to protect the kernel from the HCl byproduct resulting from subsequent silicon carbide (SiC) layer deposition. A third SiC layer is then applied by mixing methyltrichlorosilane ( $CH_3SiCl_3$ ) and hydrogen with or without the presence of argon. This layer provides the main structural strength of the particle and is the primary barrier to the release of non-gaseous fission products not sufficiently retained by the IPyC layer or within the kernel itself. An additional outer pyrocarbon (OPyC) layer is then deposited using a similar approach to the IPyC layer. This final layer protects the SiC layer during handling, acts as a surface for bonding to the graphitic fuel matrix and provides an additional barrier to fission product release (especially fission gases). The finished coated particle is a small ( $\leq 1$  mm outside diameter) carbon and ceramic sphere that is stable to temperatures well beyond  $1600^\circ C$  (Mitchell, et al., 2020).

TRISO particles are then consolidated in a matrix on around a low-density core of the fuel pebble (Figure 3-2 and Figure 4-1). The matrix is achieved by pressurizing graphite flakes and a thermoplastic or thermosetting resin with the fuel particles and heating to carbonize the resin. TRISO particles are overcoated with the graphite/resin blend prior to pressing in order to prevent particle-to-particle contact during the molding process, which may damage the particles.

In the 1960s and 1970s, a large-scale coated particle fuel fabrication facility was established by General Atomics to support the production of prismatic fuel elements containing TRISO-coated  $(Th,U)C_2$  fissile fuel particles and TRISO-coated  $ThC_2$  fertile fuel particles (EPRI, 2020). The facility has since been decommissioned and dismantled, and therefore no facility is currently licensed for the fabrication of TRISO fuels in the United States. In 2002, the U.S. DOE established the Advanced Gas Reactor Fuel Development and Qualification Program, which one of its objectives is to establish domestic TRISO particle fuel manufacturing capabilities in support of prismatic and pebble-bed HTGR fuel designs (Mitchell, et al., 2020). Under this program, TRISO particles with UCO kernels enriched to 19.7 percent  $^{235}U$  have been fabricated. The target quality level for TRISO-coated particle fuel is based on the quality level achieved for a fuel particle design (EUO 2358–2365) in a German program during the late 1980s. A follow-on assessment may include an evaluation of the chemical processes and hazards for the fabrication of TRISO-coated pebbles under this ongoing DOE program.

#### **4.1.4.2 TRISO Transportation**

A review of operating experience and potential challenges associated with the transportation of TRISO fuels has been documented elsewhere and is not discussed further in this report. (Hall et al., 2019a; Hall et al., 2019b).

#### **4.1.5 Non-Radioactive Salt Carriers and Coolants**

The following discussions provides a description and technical considerations related to the hazards for non-radioactive salts planned for use in the five MSR designs.

#### 4.1.5.1 FLiBe Salts

The most common salt being considered for the fuel carrier salt in MSR designs is a salt referred to as FLiBe due to its composition of fluoride, lithium, and beryllium. FLiBe salts are compounds with a stoichiometry of  $\text{Li}_2\text{BeF}_4$  and composition of lithium fluoride (LiF) and beryllium fluoride ( $\text{BeF}_2$ ). FLiBe salts are highly toxic and must be handled in closed systems with personal protective equipment to minimize the chance of exposure. FLiBe salts of a nominal 2 LiF: 1  $\text{BeF}_2$  mole ratio were first used as the intermediate coolant in the MSRE at ORNL. However, a preliminary review of publicly available information on the MSRE has not identified the specific processes used for FLiBe production. Table 4-1 provides the physical properties of the FLiBe salts used in the MSRE program.

**Table 4-1 Physical Properties of MSRE Coolant/Flush FLiBe Salts (Cantor et al., 1968)**

<b>Composition</b>	LiF:BeF <sub>2</sub> (2:1 mole ratio)
<b>Liquidus Temperature (°C) [Peritectic]</b>	458 ± 1
<b>Solidus Temperature (°C)</b>	360 ± 3
<b>Density of Liquid (g/cm<sup>3</sup>)</b>	2.214 - 4.2×10 <sup>-4</sup> · T ± 2 percent
<b>Thermal Conductivity (W/m/°C)</b>	0.010 ± 0.001
<b>Solid Heat Capacity, C<sub>p</sub> (cal/g/°C)</b>	0.317 + 3.61×10 <sup>-4</sup> · T ± 3 percent
<b>Liquid Heat Capacity, C<sub>p</sub> (cal/g/°C)</b>	0.57 ± 3 percent
<b>Heat of Fusion, ΔH<sub>f</sub> (cal/g)</b>	107 ± 3 percent
<b>Viscosity, η (cP)</b>	0.116 · exp(3755/T(°K)) ± 15 percent
<b>Volumetric Coefficient of Thermal Expansion (ppm/°C)</b>	2.1 ± 20 percent
<b>Vapor Pressure, P<sub>v</sub> (Torr)</b>	log P <sub>v</sub> (Torr) = 9.04 - (10500/T(°K)) ± 10 percent
<b>Electrical Conductivity (ohm/cm)</b>	1.54 + 6.0×10 <sup>-3</sup> (T - 500) ± 10 percent

Materion Corporation provided FLiBe salts for use in the MSRE (Vidal, 2013). A preliminary review of information available on the LFTR has not identified the specific composition of FLiBe salts to be used in that reactor design.

The production of FLiBe salts will generally require isotopic enrichment with <sup>7</sup>Li to be usable as an MSR carrier and coolant salt, due to the high neutron absorption cross section of <sup>6</sup>Li. It is estimated that similar purities to the MSRE flush and coolant salts will be required, which had a <sup>7</sup>Li abundance ranging from 99.991 to 99.994 percent. One method for <sup>7</sup>Li enrichment method that was used in the 1950 to 1960s was a lithium hydroxide-mercury amalgam column exchange-based (Colex) separation process; however, industrial-scale mercury usage presents large environmental contamination and health risks (McFarlane et al., 2019). Other lithium separation methods include ion exchange, chemical exchange, electrolysis, electro-migration, and selective capture of lithium isotopes with a crown ether (Martoyan et al., 2016).

FLiBe salts are corrosive and contain highly hazardous beryllium, which may impact handling and processing operations at the plant. Management of FLiBe salts requires protective clothing and respirators, or the use of glove boxes, as well as specialized training, policies, and procedures.

#### **4.1.5.2 Other Non-Radioactive Salts**

A unique aspect of MSR designs is that they feature a broad spectrum of different coolant compositions. In addition to the most common salt discussed above, FLiBe, near-term MSR technologies use a variety of commercial non-radioactive salts, including fluorides, chlorides, and nitrate-based solar salts. Many of these industrial salts are strong oxidizers and highly sensitive to moisture. Due to their hygroscopic properties, handling and storage methods will need to ensure an inert atmosphere is maintained during fuel-salt mixing and transportation operations.

Another salt considered for use in MSRs is referred to as FLiNaK and composed of fluoride (F), lithium (Li), sodium (Na), and potassium (K). FLiNaK is a ternary eutectic alkaline metal fluoride salt mixture of lithium fluoride (46.5 mole percent), sodium fluoride (11.5 mole percent), and potassium fluoride (42 mole percent). Technical considerations related to FLiNaK are not provided in this report since it is not used in near-term MSR technologies.

## **4.2 Regulatory Approaches and Safety Considerations**

Front-end operations for near-term MSR fuel technologies will involve the management of materials without regulatory precedent, which will require the safety review of different hazards relative to current solid-based light-water reactor fuels. Continued engagement to understand industry's plans for these activities and the regulatory approaches to be pursued will assist in assessing the risk significance of these hazards.

A review of available information submitted to the NRC and communicated through public workshops (e.g., annual MSR workshops hosted by ORNL) has not provided certainty about the approaches to be followed by each MSR vendor for front-end operations (ORNL, 2021). Different safety-related regulatory frameworks may be impacted depending on the chosen approach or necessary activity, including:

- 10 CFR Part 40, "Domestic Licensing of Source Material"
- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"
- 10 CFR Part 52, "Licenses, Certifications, and Approvals of Nuclear Power Plants"
- 10 CFR Part 53, "Risk Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors"
- 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"
- 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"

In addition, compliance with other regulatory frameworks related to environmental and physical protection, and materials control and accounting will need to be evaluated, including:

- 10 CFR Part 51, “Environmental Protection Regulations for Domestic Licensing and Regulatory Functions”
- 10 CFR Part 73, “Physical Protection of Plants and Materials”
- 10 CFR Part 74, “Material Control and Accounting of Special Nuclear Material”

The applicability of these regulatory frameworks will likely vary between molten salt-fueled and solid-fueled (molten salt-cooled) MSR designs as well as the need for fertile material sourcing to support reactor operation. Figure 4-2 provides a schematic of a preliminary assessment of potential regulatory approaches to be pursued.

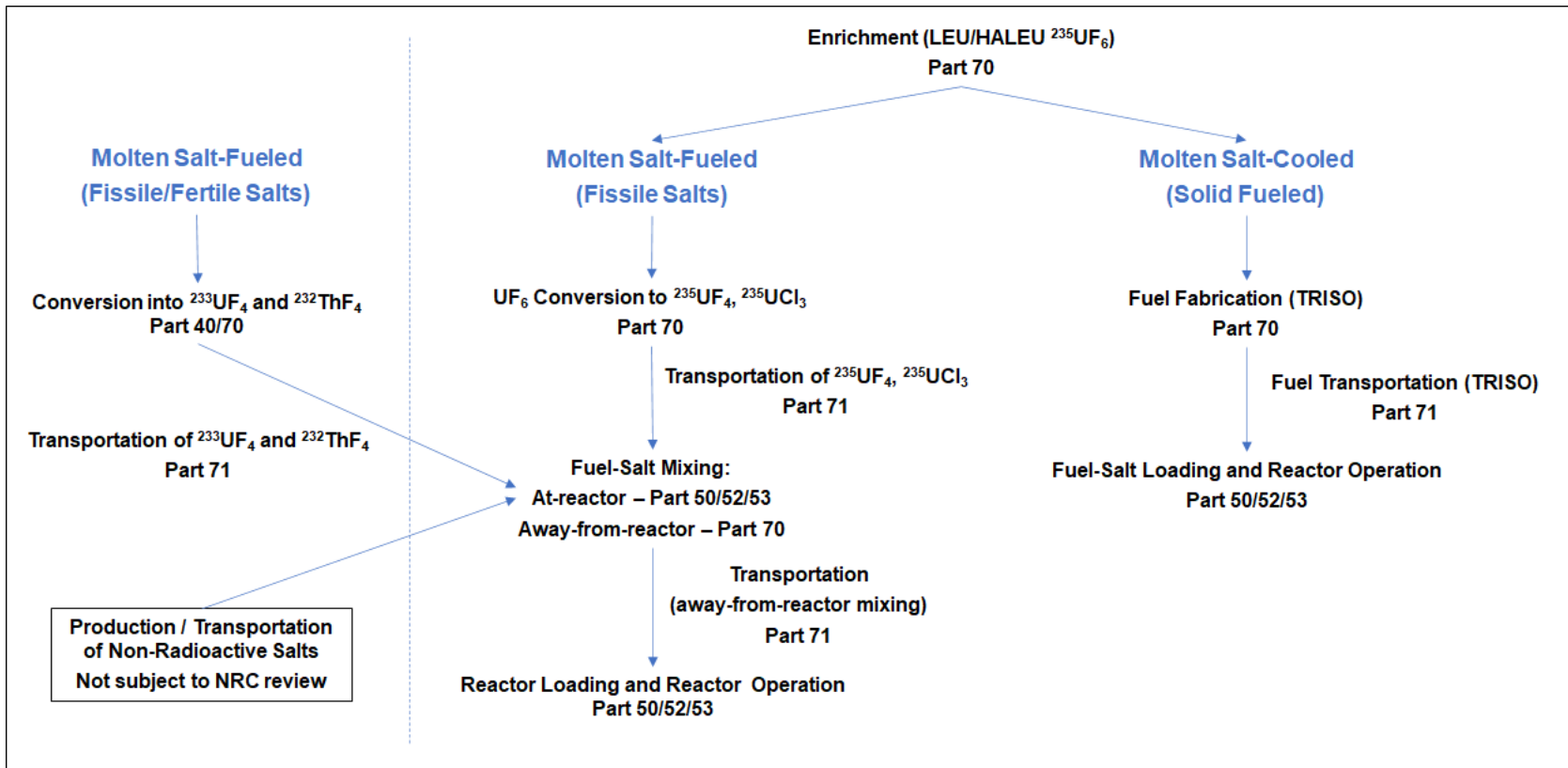


Figure 4-2 Applicability of Regulatory Frameworks to Front-End Activities for Molten Salt Reactors.

The need for new facilities for uranium conversion is not anticipated since this process step is not impacted by reactor-specific technologies. It is likely that the currently licensed uranium conversion facility operated by Honeywell near Metropolis, Illinois, which is licensed under 10 CFR Part 40, will be able to support UF<sub>6</sub> needs for MSR fuels. Approaches for the sourcing and mining operations for natural thorium to produce ThF<sub>4</sub> in support of the LFTR has not been discussed by Fluor Energy. Conventional mining of thorium ores is expected to be regulated by the Office of Surface Mining, the Department of the Interior, and the individual states where the mines are located. The activities may also be regulated by NRC under the regulatory framework for 10 CFR Part 40 if the ore is processed or chemically altered or exceeds thorium concentration thresholds, as defined in 10 CFR 40.13.

Operations for the enrichment and production of LEU and HALEU fuel materials (e.g., UF<sub>6</sub>, UF<sub>4</sub>, UCl<sub>3</sub>, ThF<sub>4</sub>, or TRISO enriched to less than 20 percent <sup>235</sup>U) are expected to be conducted under the regulatory framework for 10 CFR Part 70 and subject to safety reviews per NUREG-1520, Revision 2 (NRC, 2015). The Integrated Safety Analyses (ISA) for a production facility managing HALEU fuel materials will need to demonstrate compliance with the performance requirements of 10 CFR 70.61. NUREG-1520, Revision 2, defines a graded approach for acceptable level of risk, which will likely require different means and level of effort to meet the criteria for likelihood of events in 10 CFR 70.61.

The production of halide-based fuel materials being considered for MSRs will involve highly toxic, corrosive, and reactive compounds, which will need to be evaluated in the ISA. Consistent with the safety review guidance in NUREG-1520, Revision 2, the ISA should provide a systematic evaluation of the safety hazards credible accident sequences, items relied on for safety, and management measures that satisfy the performance requirements of 10 CFR 70.61. Further, the ISA should include quantitative standards for acute chemical exposures per the requirement in 10 CFR 70.65(b). The NRC has developed additional guidance on chemical safety practices acceptable for compliance with the regulations, as provided in NUREG-1601. Table 4-2 provides permissible exposure limits (PELs) defined by the Occupational Safety and Health Administration for the various potential chemicals to be used in the production of MSR fuels.

Although not denoted in Figure 4-2, the licensing of enrichment and fuel production facilities will also need to demonstrate compliance with the requirements in 10 CFR Parts 74 and Part 73. Consistent with NUREG-1520, Revision 2, an application for a 10 CFR Part 70 license will need to provide an adequate material control and accounting (MC&A) program to protect against the loss, theft, or diversion of special nuclear material (SNM) to be possessed, stored, and utilized at the facility. The review will assess compliance with the regulatory requirements in 10 CFR Part 74, which defines different category facilities to distinguish possession of different types and quantities of SNM (10 CFR 74.4). The NRC has previously licensed Category III facilities that possess SNM of low strategic significance facilities, which include those facilities used for LEU enrichment and fuel fabrication. However, the higher enrichment needs for MSR fuels will require licensing of Category II facilities that are authorized to possess SNM of moderate strategic significance. These facilities would manage HALEU fuel materials enriched with 10 percent but less than 20 percent <sup>235</sup>U or managing between 0.5 to 2.0 kilograms of <sup>233</sup>U.

NUREG-1520, Revision 2, provides high-level guidance for the necessary commitments for an MC&A program of Category II facilities. The NRC is developing additional guidance to support applicants in their development of MC&A programs. In addition, consistent with NUREG-1520, Revision 2, Category II facilities will need to establish and maintain a physical protection system consistent with the requirements in 10 CFR 73.67.

Different approaches may be implemented for the transportation of fuel-salt mixtures to MSR sites. The most likely approach assumed in Table 4-2 involves the independent transportation of fissile/fertile fuel material and the non-radioactive commercial salts to the reactor site facility, followed by on-site preparation of the initial reactor loading fuel-salt mixture. This approach would require NRC approval of transportation packages for individual fuel materials ( $^{233}\text{UF}_4$ ,  $^{235}\text{UF}_4$ ,  $^{235}\text{UCl}_3$ ,  $\text{ThF}_4$ , TRISO) under 10 CFR Part 71. If this approach is implemented, fuel-salt mixing would be conducted under the reactor operating license and subject to safety reviews under 10 CFR Parts 50, 52, or 53.

An alternate approach would involve the preparation of fuel-salt mixtures for reactor loading and refueling at a separate facility licensed under 10 CFR Part 70. This approach would reduce chemical processing risks at the reactor site, particularly for the preparation of the initial core loading which will require significantly more material than refueling quantities. A separate fuel-salt preparation facility may provide efficiencies when supplying multiple reactor sites. This approach would add complexities related to the performance of fuel-salt mixtures during transport.

NUREG-2216 provides guidance for the safety review of transportation packages, which is broadly applicable to MSR fuels and fuel-salt mixtures. The performance-based regulations in 10 CFR Part 71 include assessing the effects of corrosion, chemical reactions, and radiation on the contents (10 CFR 71.43(d)) and demonstrating that the package satisfies the standards specified in subparts E and F (10 CFR 71.35(a)). The guidance also reinforces the need to assess the environment inside the packaging to which the contents are exposed, including requirements for dryness or use of inert gases, which will likely be necessary for transport of MSR fuels and fuel-salt mixtures. The primary areas of the safety review for transportation packages that are expected to be impacted include criticality, radiation dose, containment, and materials.



**Table 4-2 Safety Considerations for Production and Transportation of Molten Salt Reactor Fuels**

Activity	Production Method	Hazardous Material	Safety Considerations	OSHA Permissible Exposure Limit TWA* (OSHA, 2021)	Facility or Transportation Package Design/Performance Considerations
<b>All uranium (U)-based fuel production methods</b>	-	Uranium (different forms)	Toxic; alpha emitter	25 ppb of U	
	Direct and indirect conversion	Hydrogen fluoride (HF)	Toxic; highly corrosive; reacts with metals to produce H <sub>2</sub>	3 ppm	
<b>Uranium tetrafluoride (UF<sub>4</sub>) production</b>	Direct conversion – hydrogen reduction	Hydrogen (H <sub>2</sub> )	Flammable	N/A	
	Direct conversion - CCl <sub>4</sub> reduction	Carbon tetrachloride (CCl <sub>4</sub> )	Toxic; possible carcinogen	10 ppm	
		Chlorine (Cl <sub>2</sub> )	Toxic; corrosive	1 ppm (ceiling)	
		Freons (CCl <sub>x</sub> F <sub>(4-x)</sub> )	May cause frostbite; may displace oxygen	1000 ppm	
	Indirect conversion - AUC & ADU	Ammonia (NH <sub>3</sub> )	Toxic; corrosive; may form explosive mixture with air at high temperatures	50 ppm	
<b>Uranium trichloride (UCl<sub>3</sub>) production</b>	Chlorination	Phosgene (COCl <sub>2</sub> )	Toxic; corrosive	0.1 ppm	10 CFR Part 70, NUREG-1520 Integrated Safety Analysis: Quantitative Standards for Chemical Consequences
		Hexachloropropene (C <sub>3</sub> Cl <sub>6</sub> )	Toxic	Not available	
		Carbon m(CO):	Toxic	50 ppm	
		Others - CCl <sub>4</sub> , Cl <sub>2</sub>	See above	See above	
<b>Thorium fluoride (ThF<sub>4</sub>) production</b>	Fluorination	HF	Toxic; extremely corrosive; reacts with metals to produce H <sub>2</sub>	3 ppm	
		Fluorine (F <sub>2</sub> )	Toxic; strong oxidizer/ corrosive	0.1 ppm	
		Ammonium fluoride (NH <sub>4</sub> F)	Toxic; corrosive; reacts with moisture to form HF	1.65 ppm	
		Ammonium bifluoride (NH <sub>4</sub> HF <sub>2</sub> )	Toxic; corrosive; reacts with moisture to form HF	1 ppm	
<b>TRISO fuel production</b>	Sol-gel synthesis	Nitric Acid (HNO <sub>3</sub> )	Toxic; corrosive	2 ppm	
		Ammonia (NH <sub>3</sub> )	Toxic; corrosive; may form explosive mixture with air at high temperatures	50 ppm	

Activity	Production Method	Hazardous Material	Safety Considerations	OSHA Permissible Exposure Limit TWA* (OSHA, 2021)	Facility or Transportation Package Design/Performance Considerations
<b>TRISO fuel production</b>	Sol-gel synthesis	Hexamethylenetetramine (C <sub>6</sub> H <sub>12</sub> N <sub>4</sub> )	Toxic; Combustible; Finely dispersed particles form explosive mixtures in air	174 ppb	
		Urea (CH <sub>4</sub> N <sub>2</sub> O)	Toxic	6 ppm	
	Chemical vapor deposition	Acetylene (C <sub>2</sub> H <sub>2</sub> )	Extremely flammable	Not available	
		Propylene (C <sub>3</sub> H <sub>6</sub> )	Extremely flammable	500 ppm	
		Methyltrichlorosilane (CH <sub>3</sub> SiCl <sub>3</sub> )	Flammable, toxic	5 ppm (ceiling)	
		Hydrochloric acid (HCl)	Toxic, Corrosive	5 ppm (ceiling)	
<b>Transportation of UF<sub>4</sub>, UCl<sub>3</sub>, and ThF<sub>4</sub></b>	-	Decomposition Byproduct: Cl <sub>2</sub> Moisture Reaction Byproduct: HF, hydrochloric acid (HCl), uranyl fluoride (UO <sub>2</sub> F <sub>2</sub> )	Moisture control, thermal stability	See above	10 CFR Part 71, NUREG-2216 Safety Analysis Report: Criticality, Radiation Dose, Containment, Materials

\* 8-hour time weighted average (TWA), unless specified as ceiling. TWA is defined by the Occupational Safety and Health Administration (OSHA) as an employee's average airborne exposure in any 8-hour work shift of a 40-hour work week which shall not be exceeded.

## 5 SUMMARY AND RECOMMENDATIONS

The U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Material Safety and Safeguards (NMSS) has been monitoring both the U.S. Department of Energy (DOE) and industry's activities related to the development of advanced non-light-water reactor technologies, including those related to the potential licensing and certification of molten salt reactors (MSRs). To date, the NRC is aware of five vendors that have expressed interest or have been actively engaged in the development and potential near-term licensing and certification of MSR technologies, namely:

- the Integral Molten Salt Reactor under development by Terrestrial Energy USA,
- the Fluoride Salt-Cooled High Temperature Reactor under development by Kairos Power,
- the Lithium Fluoride Thorium Reactor under development by Flibe Energy,
- the Molten Chloride Fast Reactor under development by TerraPower and Southern Nuclear, and
- the Molten Salt Research Reactor under development by Abilene Christian University.

This report provides a preliminary assessment of technical and regulatory considerations related to the management of unirradiated (fresh) fissile and fertile fuel materials (i.e., front-end operations) for near-term MSR technologies, including activities related to their enrichment, production, and transportation to a reactor site. These activities present distinct technical and regulatory challenges relative to uranium oxide based fuels used in current light-water reactor technologies and will involve licensing and certification actions without regulatory precedent.

Oak Ridge National Laboratory has historically led efforts to support the development of MSR technologies with two designs being built and operated, namely the Aircraft Reactor Experiment and the Molten Salt Reactor Experiment. The front-end activities for these prior operating reactors were conducted at the Y-12 National Security Complex with fuel-salt materials being provided by the Atomic Energy Commission and commercial sources. Production operations required specialized facilities for managing the highly corrosive and toxic materials and reducing potential impurities. Transportation activities were not conducted using commercial transportation packages. These operations provide perspective on the potential technical issues for front-end activities for future MSR technologies; however, the activities were not regulated under current NRC requirements.

A review of near-term MSR technologies based on publicly available information was conducted to identify preliminary fuel and salt compositions, as well as the current state of knowledge on plans for the enrichment, production, and transportation of unirradiated materials. The fuel-salt mixtures will be a combination of fissile and/or fertile materials of LEU or other isotopic ( $^{233}\text{U}$ ,  $^{232}\text{Th}$ ) compositions with non-radioactive diluent or carrier salts. Fluoride salt-fueled reactors are expected to operate with uranium tetrafluoride ( $\text{UF}_4$ ) and thorium fluoride ( $\text{ThF}_4$ ), if fertile

material is used, dissolved in molten salts including FLiBe. It is feasible that chloride salt-fueled reactors will operate using uranium trichloride ( $\text{UCl}_3$ ) in salts including sodium chloride ( $\text{NaCl}$ ), calcium chloride ( $\text{CaCl}_2$ ) and magnesium chloride ( $\text{MgCl}_2$ ). Fluoride salt-cooled reactors are designed to operate with a solid pebble fuel based on tristructural isotropic (TRISO) particles. A review of potential methods and technical considerations for the enrichment and production of these materials is provided. Commercial vendors have provided very limited information on approaches for transportation to the reactor sites.

Different safety-related regulatory frameworks may be impacted depending on the chosen approach or necessary front-end activity, including 10 CFR Parts 40, 52, 53, 70, and 71. In addition, compliance with other regulatory frameworks related to environmental and physical protection, and materials control and accounting will need to be evaluated, including 10 CFR Parts 51, 73, and 74. The applicability of these regulatory frameworks will likely vary between molten salt-fueled and solid-fueled (molten salt-cooled) MSR designs as well as the need for fertile material sourcing to support reactor operation.

The need for new facilities for uranium conversion is not anticipated since this process step is not impacted by reactor-specific technologies. Operations for the enrichment and production of LEU and HALEU fuel materials (e.g.,  $\text{UF}_6$ ,  $\text{UF}_4$ ,  $\text{UCl}_3$ ,  $\text{ThF}_4$ , or TRISO enriched to less than 20 percent  $^{235}\text{U}$ ) are expected to be conducted under the regulatory framework for 10 CFR Part 70 and subject to safety reviews per NUREG-1520, Revision 2 (NRC, 2015). The Integrated Safety Analyses (ISA) for a production facility managing HALEU fuel materials will need to demonstrate compliance with the performance requirements of 10 CFR 70.61. The production of halide-based fuels will involve highly toxic, corrosive, and reactive compounds, which will need to be evaluated in the ISA, including a systematic review of the safety hazards and credible accident sequences, items relied on for safety, and management measures that satisfy the performance requirements of 10 CFR 70.61.

Different approaches may be implemented for the transportation of fuel-salt mixtures to MSR sites. One approach would involve the independent transportation of fissile/fertile fuel material and the non-radioactive commercial salts to the reactor site facility, followed by on-site preparation of the initial reactor loading fuel-salt mixture. This approach would require NRC approval of transportation packages for individual fuel materials ( $^{233}\text{UF}_4$ ,  $^{235}\text{UF}_4$ ,  $^{235}\text{UCl}_3$ ,  $\text{ThF}_4$ , TRISO) under 10 CFR Part 71. If this approach is implemented, fuel-salt mixing would be conducted under the reactor operating license and subject to safety reviews under 10 CFR Parts 50, 52, or 53. Per this approach, the production and transportation of industrial non-radioactive salts would remain outside NRC's regulatory purview.

An alternate approach would involve the preparation of fuel-salt mixtures for reactor loading and refueling at a separate facility licensed under 10 CFR Part 70. This approach would reduce chemical processing risks at the reactor site, particularly for the preparation of the initial core loading which will require significantly more material than refueling quantities. A separate fuel-salt preparation facility may provide efficiencies when supplying multiple reactor sites. This approach would add complexities related to the performance of fuel-salt mixtures during transport.

The current safety review guidance for transportation packages is broadly applicable to MSR fuels and fuel-salt mixtures. The performance-based regulations in 10 CFR Part 71 include assessing the effects of corrosion, chemical reactions, and radiation on the contents (10 CFR 71.43(d)) and demonstrating that the package satisfies the standards specified in subparts E and F (10 CFR 71.35(a)). The guidance also reinforces the need to assess the environment inside the packaging to which the contents are exposed, including requirements for dryness or use of inert gases, which will likely be necessary for transport of MSR fuels and fuel-salt mixtures. The primary areas of the safety review for transportation packages that are expected to be impacted include criticality, radiation dose, containment, and materials.

The following recommendations are made for follow-on activities to support NMSS mission to license MSR fuel cycle operations, including front-end operations:

- **Expanded assessment of impacted regulatory frameworks.** There is some degree of uncertainty related to the approaches used for mixing (outside facility or at-reactor) and transportation of fuel-salt mixtures. The preliminary assessment in this report assumes that mixing operations will occur at the reactor site; however, some developers (e.g., Abilene Christian University) expect that the pre-mixed fuel will be shipped directly to the reactor site. The first approach would involve regulatory oversight for mixing operations under 10 CFR Part 50, 52, or 53. The latter approach would involve regulatory oversight under 10 CFR Part 70. An extensive review of the applicable safety review guidance for these approaches was not conducted in this report.
- **Hazards associated with FLiBe and FLiNaK salts.** The preliminary assessment in this report did not identify specific methods for the production of FLiBe and FLiNaK salts (although the latter are not proposed in near-term MSR technologies). Further, an evaluation of hazards associated with mixing operations of fuel materials with FLiBe and FLiNaK salts was not provided. A more extensive evaluation of these hazards and complexities for the design of transportation packages could be conducted in a follow-on assessment.
- **Production of HALEU feed materials for MSRs.** DOE and its national laboratories are exploring various options for the production of HALEU fuel feed material, including two chemical processes (Electrochemical Processing, Hybrid Zirconium Extraction Process). Both methods involve the recycling of spent nuclear fuel from DOE research reactors to recover highly enriched uranium (greater than 20 percent) that can then be down blended to make HALEU fuel feed material. A follow-on assessment could assess technical and regulatory considerations related to these DOE processes, if they were to be implemented by a commercial facility subject to NRC purview.
- **Fabrication of TRISO pebble fuels.** In the 1960s and 1970s, a large-scale coated particle fuel fabrication facility was established by General Atomics to support the production of prismatic fuel elements containing TRISO-coated (Th,U)C<sub>2</sub> fissile fuel particles and TRISO-coated ThC<sub>2</sub> fertile fuel particles (EPRI, 2020). The facility has since been decommissioned and dismantled, and therefore no facility is currently licensed for the fabrication of TRISO fuels in the United States. In 2002, the DOE

established the Advanced Gas Reactor Fuel Development and Qualification Program, which one of its objectives is to establish domestic TRISO particle fuel manufacturing capabilities in support of prismatic and pebble-bed HTGR fuel designs (Mitchell, et al., 2020). Under this program, TRISO particles with UCO kernels enriched to 19.7 percent  $^{235}\text{U}$  have been fabricated. The target quality level for TRISO-coated particle fuel is based on the quality level achieved for a fuel particle design (EUO 2358–2365) in a German program during the late 1980s. A follow-on assessment may include an evaluation of the chemical processes and hazards for the fabrication of TRISO-coated pebbles under this ongoing DOE program.

- **Thorium-232 Production.** A preliminary review of the literature has not identified the need or particular method for the enrichment of  $^{232}\text{Th}$  to remove traces amounts of other thorium isotopes (e.g.,  $^{230}\text{Th}$ ,  $^{229}\text{Th}$ ) in support of MSR breeders. Various reactors incorporating thorium-based fuels have been deployed internationally, including three in the United States (NEA & IAEA, 2016). A follow-on assessment may include a review of documents pertinent to front-end activities for those reactors, which could provide more details on any implemented processes for the purification of  $^{232}\text{Th}$ . The need for this assessment should be informed by vendor needs.
- **Graphite Performance.** A key technical concern related to MSR performance continues to be the limited lifetime of graphite materials in the reactor core. Although this does not present a technical issue for front-end activities in new MSR fuel cycles, consideration should be provided to the generation and management of irradiated graphite waste and implications to future safety reviews for its storage, transportation, and eventual disposal.
- **Materials Compatibility.** The production of halide-based fuels and mixing with commercial non-radioactive salts will require adequate materials to support high temperature operations and limit the generation of impurities in these processes. The need for new materials for transportation package designs may need to be reviewed per the requirements for normal conditions of transport and hypothetical accident conditions. The review could include an assessment of compatibility of previously used materials such as Inconel and Hastelloy N, as well as other mitigation strategies for degradation that could be implemented.
- **DOE Engagement.** Continued engagement with DOE will support NRC's understanding of activities in support of MSRs, including the Advanced Reactor Demonstration Program (ARDP), the Research Reactor Infrastructure Program (RRIP), the Gateway for Accelerated Innovation in Nuclear (GAIN) Program and the Advanced Gas Reactor Fuel Development and Qualification Program.
- **Assessment of other domestic and international MSR technologies under development.** A follow-on assessment may evaluate technical considerations for MSR technologies under development by other commercial vendors that have not yet engaged with the NRC on pre-application engagements. These include the Stable Salt Reactor (under development by Moltex Energy), the Molten Chloride Salt Fast Reactor (under development by Elysium Industries), the Thorium Molten Salt Reactor 500 (under

development by ThorCon Power), the Mu\*STAR accelerator-driven small modular reactor (under development by Muons), and the Micro Molten Salt Reactor (ARC Generator) under development by Alpha Tech Research Corp. In addition, the review could assess MSR technologies under development by international organizations and countries, such as the Molten Salt Fast Reactor (SAMOFAR) project funded by European Atomic Energy Commission (Euratom), the Molten Salt Actinide Recycler and Transmuter (MOSART) concept explored by the Russian Federation, the thorium-fueled liquid fluoride MSR to be operated by the People's Republic of China, and the thorium-fueled molten salt breeder reactor designed by the Republic of India.

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