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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Abilene Christian University Molten Salt Research Reactor Fuel Qualification Methodology Topical Report, Revision 0

Abilene Christian University received a construction permit for the MSRR from the Nuclear Regulatory Commission on September 16, 2024 (ML24243A040). As a pre-application activity leading to the submission of an operating license application, ACU has prepared the enclosed topical report on fuel qualification methodology. ACU requests that the Nuclear Regulatory Commission review and approve the proposed methodology as sufficient for establishing an acceptable fuel salt operating envelope for the MSRR.

The topical report does not contain any proprietary or commercially sensitive information and does not need to be withheld from public disclosure in accordance with 10 CFR 2.390. If you have questions or need additional information, please contact Vicente Rojas at v.rojas@acu.edu or Lester Towell at Lester.Towell@acu.edu.

Respectfully,

Rusty Towell

Rusty Towell, PhD
Director of NEXT Lab

Enclosure: Topical Report on Fuel Qualification Methodology, Revision 0

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Abilene Christian University Molten Salt Research Reactor

Topical Report on

Fuel Qualification Methodology

Revision 0

January 2025

Enclosure

Document Approval

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Version History

Revision	Description of Change	Date
0	Initial Issuance	January 2025

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List of Acronyms

ACU	Abilene Christian University
ASTM	American Society for Testing and Materials
BeF_2	Beryllium fluoride
CFR	Code of Federal Regulations
DOE	Department of Energy
DSC	Differential scanning calorimetry
EBR II	Experimental breeder reactor II
FHS	Fuel handling system
FLiBe	Lithium fluoride beryllium fluoride, LiF-BeF_2
FP	Fission product
FSF	Fundamental safety function
GMS	Gas management system
HALEU	High-assay low-enriched uranium
HF	High burnup fission
ICP-MS	Inductively coupled plasma mass spectrometry
INL	Idaho National Laboratory
LF	Low burnup fission
LiF	Lithium fluoride
LP	Liquidus point
MSR	Molten salt reactor
MSRE	Molten Salt Reactor Experiment
MSRR	Molten Salt Research Reactor
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PDC	Principal design criteria
PHX	Primary heat exchanger
RAV	Reactor access vessel
UF_3	Uranium trifluoride
UF_4	Uranium tetrafluoride
UO_2	Uranium oxide
USA	United States of America
ZrF_4	Zirconium tetrafluoride

Glossary

Term	Definition
Carrier salt	The salt mixture of LiF and BeF_2 in an approximate 2:1 ratio with enriched ^{7}Li to 99.99 mol % and without the UF_4 fuel.
Coolant salt	The salt mixture of LiF and BeF_2 in an approximate 2:1 ratio used in the MSRR's coolant salt system to remove heat from the reactor loop during normal operation.
Fuel	UF_4 salt enriched in ^{235}U.
Fuel salt	The combination of the carrier salt and fuel, which is the salt mixture of LiF, BeF_2, and UF_4 in a nominal molar ratio of 63:32:5 used in the reactor system.
HALEU	Uranium enriched to between 5% and 20% ^{235}U.
MSRE salt	The salt from the coolant, or secondary, loop system of the MSRE at ORNL.
Test salts	A variety of salt mixtures used in the laboratory testing program which have comparable compositions to salts used in the MSRR.

Executive Summary

This report describes Abilene Christian University's (ACU) methodology for qualifying liquid salt fuel to support the Molten Salt Research Reactor (MSRR). The fuel salt is composed of a lithium fluoride and beryllium fluoride (FLiBe) carrier salt fueled with high-assay low-enriched uranium (HALEU) in the form of uranium tetrafluoride (UF_4). The qualification process described in this report focuses on assessing the thermophysical properties for a range of fuel salt compositions that are representative of the MSRR's planned operational fuel salt. The test salts used for qualification address various chemical compositions, thermal effects, and the presence of fission products (FPs).

ACU's fuel design builds on the reported fuel performance from the Molten Salt Reactor Experiment (MSRE). Its qualification is augmented by testing programs intended to inform the MSRR's design and operation. The primary objective of the data generated through the laboratory testing programs is to provide validated inputs for the reactor physics and thermal-hydraulic models used in the MSRR's safety analyses. Testing the salt's thermophysical properties will confirm the design basis for MSRR's safety systems and verify that these systems achieve the fundamental safety functions (FSFs) of controlling reactivity, removing heat, and restricting radioactivity release.

This report discusses the fuel design rationale, performance requirements, qualification methods, fuel preparation, content adjustment techniques, and the proposed laboratory testing program ("the Program").

Overall, the ACU MSRR fuel qualification methodology provides reasonable assurance that the MSRR fuel will support safe operations within the operational boundaries and design basis accidents. After completing the Program, a report which describes the results will be provided to the Nuclear Regulatory Commission (NRC) in support of the MSRR operating license application.

ACU requests that the NRC review and approve the proposed methodology described in Section 4 as sufficient for establishing an acceptable fuel salt performance envelope. Fuel design (Section 3), fuel surveillance (Section 5), and fuel handling and storage (Section 6) are not investigated by the Program but are included in this topical report to provide background information about the MSRR.

1 Introduction

ACU's MSRR is a loop-type, liquid-fueled molten salt reactor (MSR) designed to operate at a thermal power output of up to 1 MW. The MSRR will be constructed and operated at ACU as a Class 104(c) utilization facility in accordance with 10 CFR 50.21(c) and licensed under 10 CFR Part 50.

ACU received a letter of support from the United States Department of Energy (DOE) Office of Nuclear Energy stating that the MSRR will be added to the Research Reactor Infrastructure program, and requests for fuel, fuel delivery, and fuel dispositions will be considered (A. Caponiti, personal communication, November 15, 2019). ACU has requested fuel from the Experimental Breeder Reactor II (EBR II) and enriched FLiBe salt used in the MSRE's secondary coolant loop. Although the fuel and MSRE salt have been requested, contracts with the DOE have not been executed at this time. ACU is working with DOE to define a path for the procurement of these materials, which includes the provisions of HALEU, conversion of the uranium to UF_4, and the provision of the MSRE salt (Smith, 2024). Characteristics of the EBR II fuel and the MSRE salt are provided in this report. If the DOE does not provide the fuel or MSRE salt, fuel or carrier salt of similar character will be procured by ACU.

1.1 Purpose

This topical report provides the methodology for qualification of the MSRR's molten salt fuel ("fuel salt"). This qualification methodology has been informed by the NRC's *NUREG/CR-7299 Fuel Qualification for Molten Salt Reactors* (2022) and *NUREG-2246 Fuel Qualification for Advanced Reactors* (2022). The results from the execution of the methodology will be presented in a future licensing submission that will demonstrate acceptable fuel behavior for the MSRR operating envelope through validated analysis. The MSRR's fuel qualification will provide assurance that the physical behavior of the fuel salt is sufficiently understood so that its neutronics and thermal hydraulics can be adequately modeled for both normal operation and accident conditions, thus reflecting the role of the fuel design in the overall safety of the facility.

The MSRR mission is to support the development of commercial MSRs by providing operational experience with the fuel salt. For commercial power reactor designs that differ significantly from existing light-water reactor designs, 10 CFR 50.43(e) requires that the performance of safety features of the new reactor design be demonstrated through 1) either analysis, appropriate test programs, experience, or a combination thereof, or 2) acceptable testing of a prototype plant over a sufficient range of conditions. The MSRR's operating experience, in combination with the laboratory testing described in this topical report, will help to fulfill the requirements of 10 CFR 50.43(e).

1.2 Scope

The ACU methodology for fuel qualification focuses on testing and characterizing the fuel salt's thermophysical properties that support the performance of fuel salt in the reactor system. The salt's interaction with Type 316H stainless steel, as affected by corrosion reaction kinetics and indicated by redox potential, is not within the scope of ACU's fuel qualification and will be addressed in the Degradation Management Program.

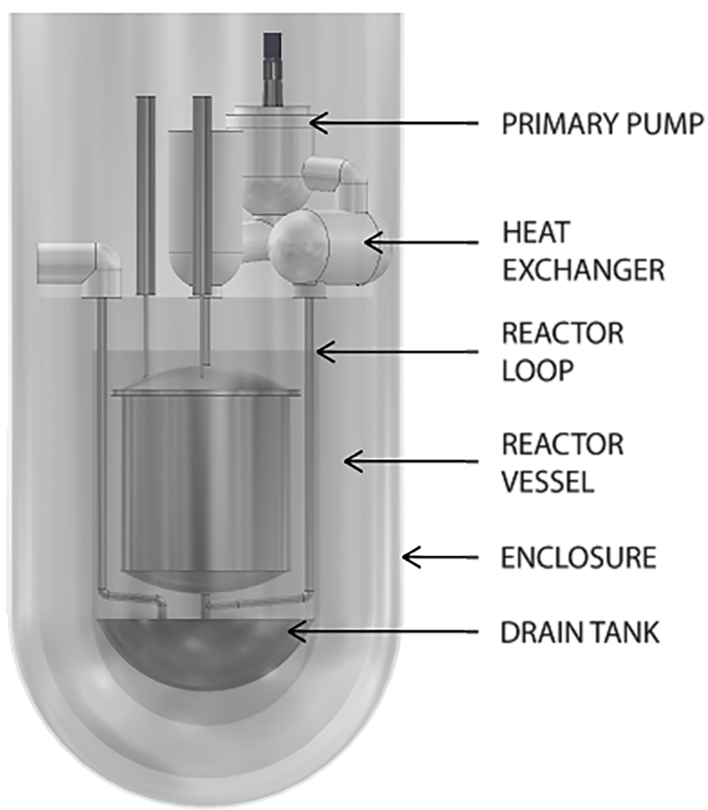
2 Reactor Design and Regulatory Background

2.1 MSRR Design Features

The MSRR is fueled with a uranium-bearing, fluoride-based salt containing minimal oxidative impurities. The fuel salt mixture is nominally composed of lithium fluoride, beryllium fluoride, and uranium tetrafluoride (LiF-BeF_2-UF_4) in a molar ratio of approximately 63:32:5, respectively. The fuel used is HALEU, which is ^{235}U enriched to at least 19.5% and is in the form of UF_4. Additionally, the lithium in the fuel salt is enriched to 99.99% ^{7}Li.

The reactor vessel, the reactor pump, the primary heat exchanger (PHX), the drain tank, the reactor enclosure, and the associated piping comprise the loop-type reactor system, as shown in Figure 1. The reactor drain tank is located below the reactor vessel at the lowest point in the reactor system and is connected to the reactor loop at the cold leg between the PHX and the reactor vessel.

Figure 1: Conceptual Diagram of MSRR



During normal reactor operation, the fuel salt is kept in the reactor loop by pneumatically pressurizing the drain tank's headspace. The fuel salt flows through the reactor system via forced circulation. The reactor vessel containing the graphite neutron moderator is designed to sustain a nuclear fission chain reaction. The fuel salt flows through the neutron moderator, which is composed of square lattice graphite blocks with flow channels. From the reactor vessel, the fuel salt moves into the Reactor Access Vessel (RAV) and the pump bowl. It is then pumped into the shell side of the PHX, transferring heat to the coolant salt before being directed back to the reactor vessel through the cold leg.

Reactor shutdown, whether under normal or accident conditions, is achieved by draining the fuel salt into the drain tank. This draining process involves equalizing the cover gas pressures between the RAV and the drain tank by opening reactor trip valves. As the pressures equalize, the fuel salt passively drains from the reactor system into the drain tank under the force of gravity. The drain tank is designed to hold the entire end-of-life fuel salt inventory in a highly subcritical configuration and to passively manage decay heat.

The FSFs of the MSRR under normal and accident conditions are to 1) limit radionuclide release, 2) enable adequate heat removal, and 3) control reactivity.

The fuel salt limits radionuclide releases by retaining non-gaseous radionuclides in the molten salt mixture. Although the fuel salt inherently retains FPs, it is not, by itself, a leak-tight barrier to the release of radioactive materials. Multiple leak-tight barriers, which together provide defense-in-depth, restrict radioactive material release.

The fuel salt supports the safety function of adequate heat removal by possessing thermophysical properties (i.e., heat capacity, thermal conductivity, and viscosity) that enable effective heat transfer during forced-flow circulation and passive decay heat removal conditions. Forced-flow circulation allows the fuel salt to flow through the reactor vessel, where fission reactions occur, and then through the PHX, where the heat generated from fission is transferred to the coolant salt. The reactor drain tank adequately transfers decay heat to surrounding structures during shutdown and accident conditions.

The fuel salt supports the safety function of suitable negative thermal reactivity feedback through its thermal expansion and temperature-dependent changes in nuclear properties. Control rods, which can compensate, maintain, and terminate the fission chain reaction during normal reactor operation and are not relied upon for safe shutdown, provide additional reactivity control.

2.2 Accident Analysis

The MSRR accident analysis presented in ACU's Preliminary Safety Analysis Report (2024) Chapter 13, "Accident Analysis," examines the potential consequences of adverse events and accidents, as well as the facility's ability to manage such disturbances while maintaining the health and safety of its workers and the public. Accident conditions and their impact on the fuel performance envelope are discussed further in Section 4.3.

During an accident, radiological material is contained by several barriers throughout the MSRR. The fuel salt is a barrier for non-gaseous radionuclides since they remain chemically or physically bound within the salt and salt-wetted surfaces. However, gaseous FPs are not always entrained in the fuel salt and could exit the reactor system and gas management system (GMS) if the reactor system boundary is compromised. The reactor thermal management system provides a barrier to the release of solid FPs, although it is not leak-tight because of the penetrations through the top plate. The leak-tight reactor enclosure provides another barrier to FP release.

2.3 Regulatory Review

On August 12, 2022, ACU submitted a construction permit application to the NRC for the MSRR (ML22227A201). The NRC staff issued the construction permit on September 16, 2024 (ML24243A040).

ACU's fuel qualification efforts have been informed by the NRC's *NUREG/CR-7299 Fuel Qualification for Molten Salt Reactors* (2022) and *NUREG-2246 Fuel Qualification for Advanced Reactors* (2022). The NRC regulations, guidance, and principal design criteria (PDC) relevant to the fuel design are described below.

2.3.1 Regulations Relevant to the MSRR Fuel Qualification

The regulations relevant to fuel qualification of liquid-fueled MSRs are contained in 10 CFR 50.34 "Content of applications; technical information."

10 CFR 50.34(b) requires a description and safety analysis of the MSRR's structures, systems, and components.

10 CFR 50.34(b)(2) "*A description and analysis of the structures, systems, and components of the facility, with emphasis upon performance requirements, the bases, with technical justification therefor, upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished.*"

10 CFR 50.34(b)(3) *"…the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20.*"

10 CFR50.34(b)(4) "*A final analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective*" of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

10 CFR 50.36(c)(4) requires technical specifications for design features that, if altered or modified, would have a significant effect on safety.

2.3.2 Principal Design Criteria Relevant to the MSRR Fuel Qualification

The following design criteria are relevant to the qualification of the ACU MSRR fuel:

PDC 10: Reactor design

The reactor system and associated heat removal, control, and protection systems shall be designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

PDC 11: Reactor inherent protection

The reactor core and associated systems that contribute to reactivity feedback shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

PDC 16: Containment design

A reactor functional containment shall be provided to control the release of radioactivity to the environment and to ensure that the safety-related functional containment design conditions are not exceeded for as long as postulated accident conditions require.

PDC 71: Fuel salt composition control

Systems shall be provided as necessary to maintain the composition of the fuel salt within specified limits. These limits shall be based on the ability of the fuel salt to perform its safety functions.

3 Fuel Design

3.1 Background of Fuel Design

Molten salts exhibit numerous properties that make them ideal candidates for use as a liquid fuel. The fuel being a liquid salt provides benefits of improved heat transfer characteristics, minimal stress on pressure vessels, core homogeneity, fuel structure stability at high burnup, online refueling, fission gas removal, and FP retention. The ability to operate to high burnup and potentially utilize the heat produced from transuranic fission and the transmutation of FPs are also advantages for liquid-fueled MSRs in terms of economics and waste minimization. Collectively, these factors—along with the reactor's strong inherent negative reactivity feedback—contribute to the high level of intrinsic safety of liquid-fueled MSRs.

The selection of the fuel salt requires careful consideration of the chemical and thermodynamic properties of the salt constituents. Mixing a fissile salt, such as UF_4, with a carrier salt like FLiBe leads to a fuel salt with a large liquid temperature range with very low vapor pressure, a very high boiling point, and a liquidus temperature several hundred degrees Celsius below the melting point of either salt constituent. Molten salts are outstanding in their heat transport properties, aiding in the liquid fuel's exchange of heat to a secondary system. For a thermal reactor, fluorine is preferred over other halogens due to its low thermal neutron capture cross-section. The strong ionic bonding of salts gives a high capacity for retaining FPs. A majority of FPs form stable fluorides that remain in the salt, which is expected to be in chemical equilibrium with structural materials (Compere et al., 1975).

The development and demonstration of MSR technology trace back to the 1960s, with the U.S. Atomic Energy Commission funding the MSRE at Oak Ridge National Laboratory (ORNL). The MSRE fuel loop was composed of LiF-BeF_2-ZrF_4-UF_4 (65-29-5-1 mol %) and was operated at 8 MW_{th}. The MSRE used two primary fuels: uranium-235 and uranium-233. Uranium-235 was the first fuel used in the MSRE. Operation with ^{235}U (33% enrichment) in the fuel salt began in June 1965, and by March 1968, nuclear operation amounted to 9,000 equivalent full power hours (Haubenreich & Engel, 1970).

By comparison, the MSRR will have a maximum power level of 1 MW_{th} and a nominal fuel composition of LiF-BeF_2-UF_4 (63-32-5 mol %). The MSRR carrier salt will be the salt from the secondary loop of the MSRE, which already meets the desired lithium enrichment. The MSRE secondary loop did not contain uranium or FPs but was radioactive due to tritium migration through the heat exchanger. Tritium is difficult to confine, especially at high temperatures. Since the MSRE operated fifty-five years ago, this tritium has had over four half-lives of decay.

Natural lithium contains 7.5 mol % ^6Li, which produces tritium through the reaction:

$$^6Li + n \rightarrow {}^3H + {}^4He.$$

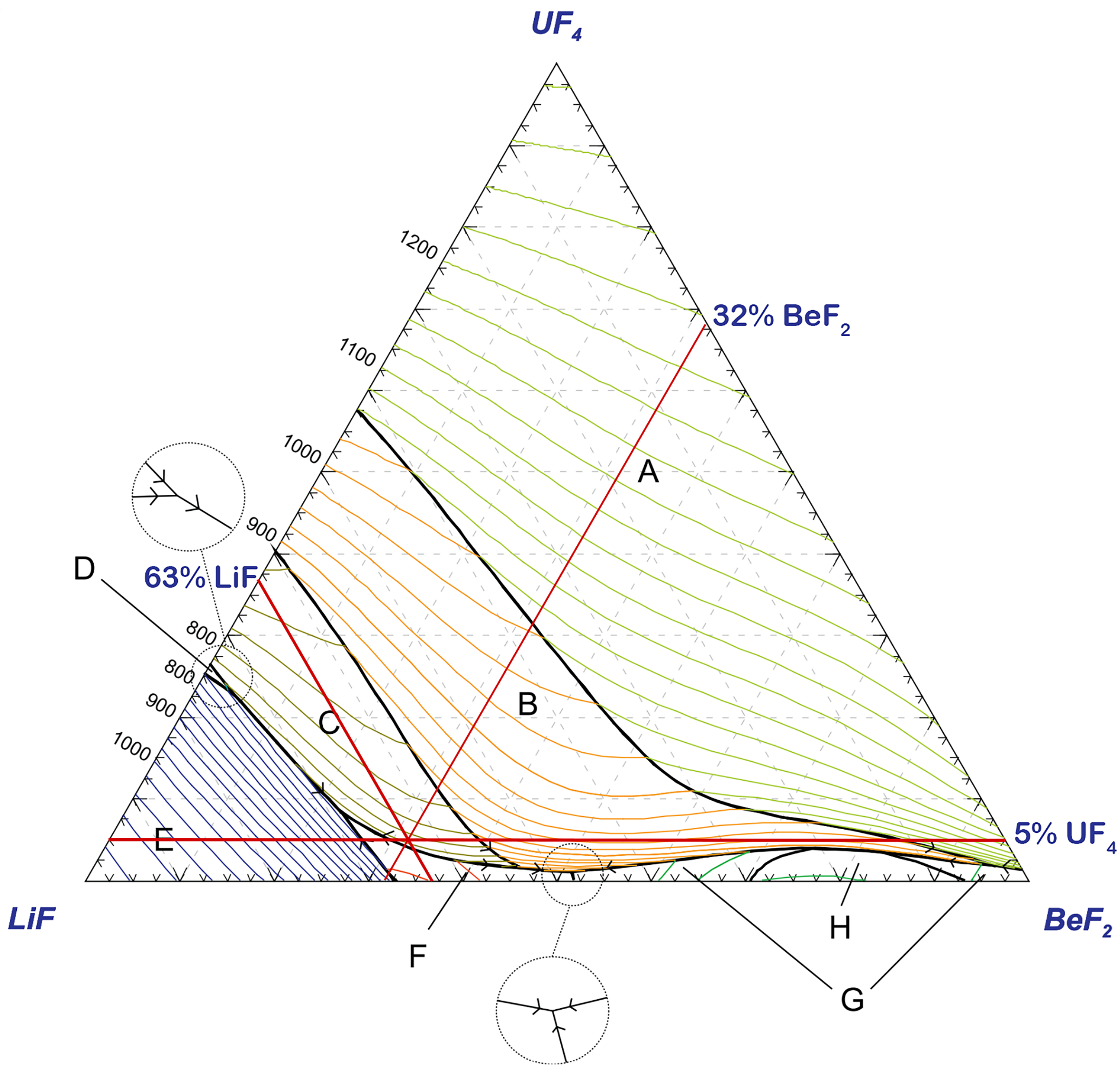
The same high thermal neutron absorption cross-section of ^6Li, which generates tritium, is also detrimental to the neutron economy in the MSRR. As such, the lithium in the MSRR carrier and coolant salts must be enriched to greater than 99.99 mol % ^7Li to support reactor physics and minimize tritium production.

The preferred source for UF_4 fuel is HALEU from the EBR II. The EBR II used metallic uranium of high enrichment. To be used by the MSRR, the metallic uranium would need to be down-blended and converted into the form of a salt as UF_4. Idaho National Laboratory (INL) would potentially perform this process.

Figure 2 provides a phase diagram for the approximate liquidus point (LP) of the fuel salt mixture. The red lines represent the nominal concentration of each salt in the MSRR mixture. The phase diagram shows how higher concentrations of UF_4 will increase the LP of the fuel salt. On one hand, a high LP may raise the reactor's operating temperature, making mechanical design and the selection of structural materials more challenging, or it could decrease the margin to the operating temperature. On the other hand, a lower LP provides thermal margin to prevent inadvertent freezing and requires less start-up energy to fully melt the salt with external heaters.

During operation, gaseous FPs accumulate in the reactor’s headspace and are managed by the GMS. Solid FPs remain suspended in the fuel salt even during accident conditions. However, a few solid FPs that are not soluble in the fuel salt may plate out throughout the reactor system. The addition of FPs is not expected to significantly change the properties of the fuel salt in the MSRR and is addressed by experiments described in Section 4.5.

Figure 2: Liquidus Projection of $LiF-BeF_2-UF_4$ (Capelli et al., 2014). Units in Kelvin.



3.2 Fuel Design Description

The $LiF-BeF_2$-based fuel salt has been chosen for the significant beneficial attributes identified in the previous section. Uranium enrichment will be at least 19.5 wt. % ^{235}U, and the lithium enrichment will be 99.99 mol % in ^{7}Li. The maximum concentration of UF_4 will be 7.0 mol %. The final specification for the fuel salt composition is to be developed by ACU with the involvement of the DOE and will be reported in the operating license application. The composition of the MSRR fuel salt constituents is summarized in Table 1 through Table 4.

The temperature range for the operation of the MSRR is well within the liquid range of the fuel salt. The liquidus of MSRR fuel salt will be around 460 °C and will be determined by the

Program described in Section 4.5. The reactor thermal management system will maintain the fuel salt temperature above the minimum operating temperature of 550 °C. The maximum normal operating temperature of the MSRR is 650 °C. The peak fuel salt temperature could reach 750 °C during postulated accident conditions, which is well below the boiling point of around 1430 °C. This postulated accident does not violate any MSRR safety limits, including the reactor vessel weld temperature limit of 704 °C.

The fuel salt will change in composition during operation because of

- FP buildup,
- the depletion of fissile material,
- the addition of UF_4 to maintain criticality,
- the addition of corrosion products from the reactor structure, and
- the addition of contaminants from cover gases.

These changes to fuel salt composition will be managed so that the fuel salt remains within the acceptable composition range in Table 1. Management of fuel composition in this range will ensure the MSRR will function as designed to limit the release of radioactive materials, ensure acceptable heat transfer, and control reactivity. The retention of radioactive materials and thermal conductivity results from the inherent physical properties of the salt. Even if the fuel salt composition were outside the performance envelope, it would remain a salt, retain FPs, and have good heat transfer properties. Reactivity will vary slightly with fuel salt composition if the concentration of UF_4 changes, with a higher concentration of UF_4 corresponding to a higher reactivity. The UF_4 concentration is managed to support reactivity control and maintain a LP well below the operating range.

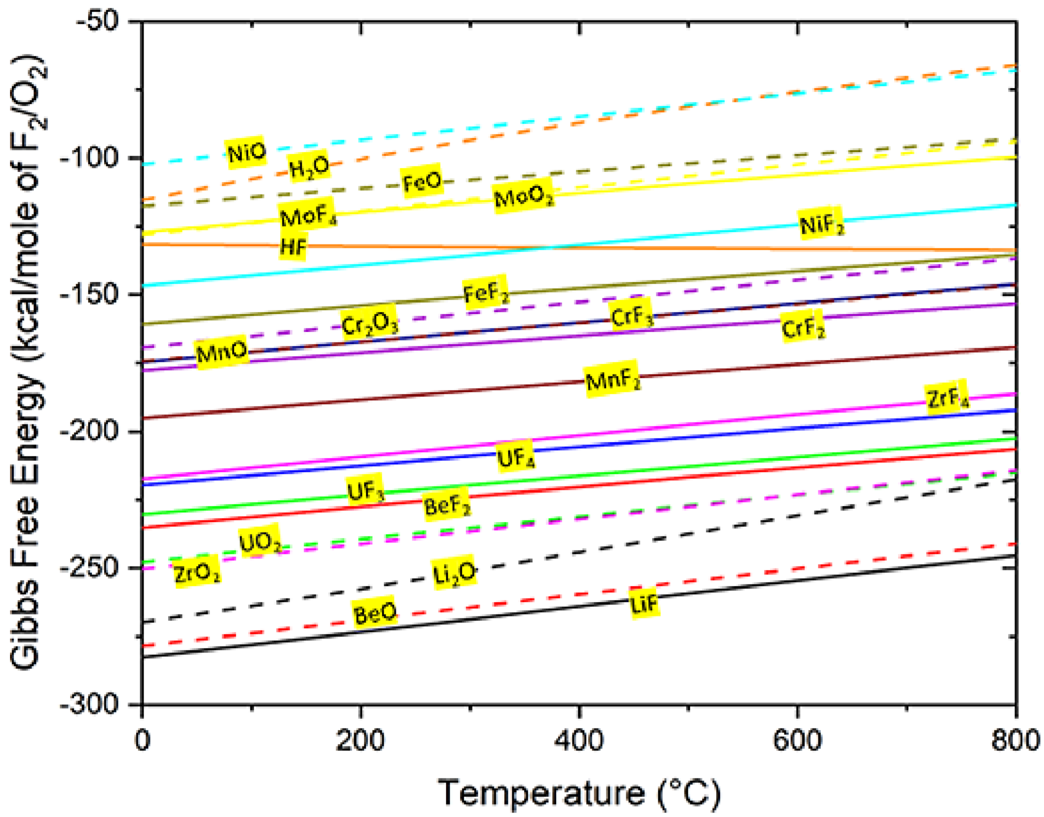
Fuel salt composition will be measured directly from sampling and analysis during operation. Specific limits will be defined for oxygen and may be defined for corrosion indicators such as chromium, iron, and redox potential. The salt will become progressively more oxidizing over the course of reactor operations. This effect will be counteracted by the periodic addition of metallic beryllium, as was done in the MSRE. Management of the redox potential will minimize corrosion in the reactor system.

Verification of the amount of impurities, the ratio of ^{6}Li to ^{7}Li, and ^{235}U enrichment will be gathered through spectrometric analysis techniques. Oxygen content will be measured with combustion analysis. Other impurities can be determined by analysis of the salt using inductively coupled plasma mass spectrometry (ICP-MS).

Reactor criticality is dependent in part on the fuel salt's density. This property creates strong negative reactivity feedback for temperature increases. Criticality is not possible outside of the reactor vessel due to the lack of an effective moderator. The fuel salt constituents and reactor system structural components scatter fast-spectrum neutrons. A moderator such as graphite is necessary to adequately thermalize the neutrons, shifting the neutron spectrum toward the thermal ^{235}U fission cross-section. The fuel salt can be drained from the reactor vessel into the drain tank to achieve subcriticality.

Corrosion of the reactor system structural material by the fuel salt is primarily driven by the amount of free fluorine and oxidizing impurities within the fuel salt. The fuel salt's redox potential and the presence of impurities will be controlled to minimize corrosion. The amount of UF_3 present in the salt will influence the redox potential of the salt but will have a negligible effect on thermophysical properties. If not controlled, oxygen within the fuel salt will cause corrosion and may lead to the precipitation of UO_2. Figure 3 shows the Gibbs Free Energy as a function of temperature for possible constituents in the fuel salt. Oxygen and other oxidizing impurities in the fuel salt are monitored by sampling throughout the reactor's operation. Beryllium will be periodically added to the fuel salt to control the redox potential of the fuel salt. Based on experiences with the MSRE molten fluoride salt systems and numerous scientific investigations into corrosion, corrosion is not expected to result in structural challenges to the reactor system boundary with controlled redox potential of the fuel salt. The Degradation Management Program will assess corrosion and other degradation mechanisms and implement mitigating measures during design, fabrication, and operation.

Figure 3: Gibbs Free Energy of Several Salt Constituents



The viscosity of the fuel salt will vary with temperature and composition. The viscosity of the fuel salt will need to be sufficiently understood to design the pump used in the reactor system and to calculate the drain time of the reactor vessel for shutdown. The Program described in Section 4.5 will measure viscosity over the operational temperature range of the reactor and possible temperatures during an accident scenario with salt compositions that encompass the expected operational ranges.

The heat capacity of the fuel salt affects the change of fuel salt temperature with heat input from the fission process and decay heat. The temperature, in turn, affects the fuel salt density, which affects the reactivity. A lower heat capacity will cause greater temperature fluctuations as the reactor power or cooling capability changes. Fuel salt heat capacity also impacts the reactor drain tank temperature immediately after the fuel salt is drained following full power operations. A higher heat capacity will have the effect of dampening the temperature changes. The Program described in Section 4.5 will measure the heat capacity for salt compositions and temperatures that encompass the expected operational ranges.

3.3 Fuel Performance Requirements

The MSRR fuel salt supports the FSFs, which are primarily achieved by the reactor system or engineered safety features. The primary contribution of the fuel salt to FSFs is the retention of FPs. The fuel salt is the first barrier in the functional containment design of the MSRR. The fuel salt achieves this safety function by its nature as a mixture with high intermolecular forces. It is an inherent physical function, not a performance requirement of the salt that can fail or degrade. The fuel salt will retain FPs, whether liquid or solid. The MSRE demonstrated this trait of FP retention (Compere et al., 1975). The MSRR will conduct sampling of both the fuel salt and the cover gas to confirm this retention behavior.

The properties of the fuel salt that support safety and operation are passive properties inherent to the fuel salt. The fuel salt density plays a role in operational reactivity control via the salt temperature. The salt's contribution to the safety function of reactor shutdown is simply by being in the liquid phase to allow draining from the reactor vessel. Apart from being a liquid, the fuel salt does not play a role in stopping the fission process. Similarly, the fuel salt's thermal conductivity supports the safety function of adequate cooling.

Fuel salt thermophysical properties are a function of the chemical composition of the salt to varying degrees. The properties related to MSRR safety and operation are heat capacity, thermal conductivity, liquidus temperature, density, and viscosity. The effect of composition on these properties will be assessed by the testing described in Section 4.5. The testing will demonstrate that the thermophysical properties are acceptable for the MSRR operating envelope. This operating envelope includes the fuel salt's chemical composition, reactor system pressure, and reactor system temperature. The range of acceptable values for the thermophysical properties is determined by the successful performance of the MSRR systems in the accident analyses. The heat capacity, thermal conductivity, liquidus temperature, density, and viscosity are all passive properties inherent to the fuel salt that contribute to the accident tolerance of the liquid fuel salt.

Because of the corrosive nature of the fuel salt in the presence of oxygen, the chemistry of the fuel salt is controlled to protect the integrity of the reactor system. Even without oxygen present, the fuel salt will become progressively more oxidizing during reactor operation. As mentioned in the fuel specification section, specific limits may be defined for impurities and redox potential.

The oxidizing nature will be counteracted by the periodic addition of metallic beryllium, which will affect the redox potential of the salt. Management of the redox potential will minimize corrosion in the reactor system.

The performance requirements for the MSRR fuel salt are to maintain its chemical composition, to remain a liquid to support reactor shutdown, to retain non-gaseous FPs, and maintain thermal conductivity for heat removal. As a highly stable salt mixture, the fuel salt will not decompose or deteriorate. The salt phase, FP retention, and thermal conductivity are inherent properties of the fuel salt.

4 Fuel Qualification Methodology

4.1 Overview

Although the objective of qualifying liquid and solid fuels is the same, the techniques to qualify liquid fuels differ from that of solid fuels. A central difference is that the composition of solid fuel is determined prior to its use by its fabrication process, whereas the composition of liquid fuel is determined prior to its use and can be adjusted over its entire operating life to maintain the desired properties.

The fuel salt's thermophysical properties, and thus its performance, are derived from its chemical composition and temperature. The qualification of the fuel salt will ensure an understanding of the fuel salt's behavior under normal operation and accident conditions, demonstrating how the fuel salt supports the various reactor systems in achieving the FSFs.

ACU's qualification methodology will utilize a laboratory testing program ("the Program") to measure the thermophysical properties of the fuel salt for both normal operation and accident conditions for the entire range of compositions anticipated during the MSRR's operating life. Property measurements will be performed using test salts loaded with varying amounts of UF_4 (up to and exceeding the planned fuel loading) and FP surrogates to mimic fuel burnup. The surrogate elements will be chosen to represent groups that exhibit similar chemical behavior. The concentrations of the simulated FPs will exceed the expected burnup of the MSRR. The Program's experimental data will validate the salt properties used in analyses, which demonstrate that the MSRR can achieve its FSFs in normal operation and accident conditions. Uncertainty tolerances for each thermophysical property will be calculated using safety analysis models. These tolerances will be used to inform statistical requirements for the Program. While not available at this time, this information will be reported alongside the results of the Program.

4.2 Fuel Specification

The fuel salt mixture is composed of lithium fluoride, beryllium fluoride, and uranium tetrafluoride (LiF-BeF_2-UF_4) in a nominal molar ratio of 63:32:5. The fuel used is HALEU as UF_4, enriched to at least 19.5%. Additionally, the lithium in the fuel salt is enriched to 99.99% ^7Li. Table 1 shows the fuel salt's design specification and its bases.

Table 1: Design Fuel Salt Base Composition

Component	Value	Basis
LiF	> 62 mol %	Maintain approximately a 2:1 ratio of LiF and BeF_2 to support the liquidus temperature.
BeF_2	> 31 mol %	Maintain approximately a 2:1 ratio of LiF and BeF_2 to support the liquidus temperature.
UF_4	< 7.0 mol %	Maintain the liquidus temperature of the fuel salt below 500 °C (Capelli et al., 2014).
^{7}Li enrichment	≥ 99.99 mol %	Reduce neutron adsorption and minimize tritium production (Riley et al., 2019).
^{235}U enrichment	≥ 19.5 wt. % < 20 wt. %	Reactor physics design and HALEU specification from EBR II (Vaden, 2021).

Unlike the fuel salt, the test salts used in the Program will not use enriched ^{7}Li or ^{235}U. This isotopic difference will not lead to any appreciable change in chemical or physical property results.

Table 2 and Table 3 show the isotopic characterization of HALEU from EBR II (Vaden, 2021). This isotopic characterization will be used in the reactor physics portion of the safety analyses that demonstrate the fuel salt supports the MSRR's FSFs.

Table 2: EBR II HALEU Uranium Assay (Vaden, 2021)

Analyte	Units	Average	Standard Deviation	Minimum	Maximum
Total U*	wt. %	99.572	0.980	98.043	100.371
^{234}U*	iso % U	0.169	0.009	0.158	0.179
^{235}U*	iso % U	19.655	0.194	19.346	19.892
^{236}U*	iso % U	0.518	0.016	0.493	0.538
^{238}U*	iso % U	79.658	0.209	79.390	79.983
^{232}U^	ng / gU	0.287	0.172	0.150	0.641
^{233}U^	ng / gU	75.771	12.170	53.175	84.067
^{237}U^	pg / gU	0.316	0.056	0.236	0.386
* Actinide isotopes measured analytically ^ Estimated using the method described in Appendix A of Vaden, 2021 iso. % U = isotope wt. % of total uranium ng / gU = g per billion grams of uranium pg / gU = g per trillion grams of uranium					

Table 3: Measured Contaminants of EBR II HALEU (Vaden, 2021)

Analyte	Units	Average	Standard Deviation	Minimum	Maximum
Cr	ppm	14.25	2.47	12.50	16.00
Ni	ND	ND	ND	ND	ND
Fe	ppm	181.10	184.32	38.80	785.00
ppm = parts per million ND = Not detected or below the minimum detection limit					

Before MSRE salt is placed into the reactor system, salt samples will be taken to assess its impurity content. For reference, Table 4 shows the MSRE's specifications for allowable concentrations of contaminants, the average concentration of impurities measured in salt samples from 24 batches of post-production MSRE coolant salt (Shaffer, 1971), and the measured concentration of impurities from an as-received batch of MSRE coolant salt used for recent purification experiments (Kelleher, 2015).

Table 4: MSRE Coolant Salt Allowable and Measured Impurity Concentrations (ppm)

Impurity	Allowable Concentration	Average Measured Concentration Post-Production	As-Received Measured Concentration
Cr	25	19	46 ± 1
Ni	25	26	0.38 ± 0.32
Fe	100	166	144 ± 13
S	250	< 5	Not reported

4.3 Fuel Performance Envelope

The fuel performance envelope is the bounding set of conditions the fuel salt is qualified to experience. The fuel salt must be maintained within the acceptable range of chemical compositions tested to ensure safe reactor operations. Compositions outside the tested range may still possess thermophysical properties that support FSFs. However, because a broader composition range has not been tested, operation outside the fuel performance envelope will not be allowed.

The following are the parameters that primarily affect fuel performance:

- Temperature
    - Thermophysical Properties: The fuel salt temperature directly affects its thermophysical properties, such as density, viscosity, thermal conductivity, and heat capacity.
    - Reactivity Control: Temperature affects reactivity due to changes in the fuel salt density.
- Chemical Composition

- Fuel Salt Stability: The fuel salt is a chemically stable, ionically bonded, eutectic mixture that retains FPs and does not react energetically with materials during normal operation and accident conditions.
- Uranium Concentration: Any significant changes in the uranium concentration will impact the reactor's reactivity and fuel salt thermophysical properties.
- FPs: Soluble FPs will remain within the salt mixture. Insoluble FPs may adhere to salt-wetted surfaces, such as Type 316H and graphite, or volatilize into the headspace gas.

The fuel salt performance envelope is provided in Table 5, alongside the Program's test envelope. The laboratory testing aims to provide additional evidence that the fuel salt will support the reactor's FSFs throughout all conditions.

Table 5: Fuel Salt Performance Envelope & Test Envelope for Thermophysical Properties

Parameter		Fuel Salt Performance Envelope	Test Envelope
Temperature Range (ºC)		25 – 750	25 – 750
Fuel Salt Constituents (mol %) $LiF:BeF_2:UF_4$		66:34:0 to 62:31:7 with approximately a 2:1 ratio of $LiF:BeF_2$	See Table 7
FP Concentration Ranges (mol %)	Transition metals	0.0 – 1.08	0.0, 0.2, 1.08
	Alkali metals	0.0 – 0.54	0.0, 0.1, 0.54
	Alkaline-earth metals	0.0 – 1.08	0.0, 0.2, 1.08
	Lanthanides	0.0 – 1.08	0.0, 0.2, 1.08
Note: Table 6 identifies FP surrogates for the laboratory testing.			

The fuel is expected to demonstrate inherent safety during all design basis events. In all design basis accidents the fuel salt is expected to:

- Maintain its chemical composition,
- Retain all non-gaseous FPs,
- Maintain adequate heat transfer properties for decay heat removal, and
- Support reactivity control.

4.4 Fuel Preparation and Fuel Content Adjustment

It is currently anticipated that the carrier salt and the UF_4 will arrive separately at ACU. Upon receipt of the salts, they will be combined and processed in the fuel handling system (FHS) to prepare the fuel salt. Samples of each salt constituent may be taken and analyzed prior to mixing to confirm their chemical composition and impurity levels. Once combined, the fuel salt may undergo further sampling and analysis to select the chemical and mechanical processes needed to achieve its allowable impurity levels and specifications. These processes include metallic beryllium additions, further UF_4 additions, hydrogen fluoride-hydrogen sparging treatment, and mechanical filtration. General descriptions of these processes and their effects are provided further below. Additional fuel salt samples may be taken and analyzed to assess the effects of the processes on its chemical composition. Once the chemical composition of the fuel salt sample satisfies its specification, the fuel salt can be transferred from the FHS into the reactor drain tank. The samples will be analyzed using various methods and instruments, including, but not limited to, ICP-MS, electrochemistry, and combustion analysis. This array of chemical analysis techniques can ensure that the salt's composition is appropriate for use in the reactor system.

The hydrogen fluoride-hydrogen sparging method (Shaffer, 1971; Kelleher et al., 2015; Mathews & Baes, 1968) utilizes hydrogen fluoride and hydrogen to purify the molten salt. This process has proven to be effective in removing oxidative impurities present in fluoride salts.

Salt that has been treated with hydrogen fluoride and hydrogen will contain dissolved hydrogen fluoride and metal fluorides, both leading to an undesirable redox potential (Shaffer, 1971; Mathews & Baes, 1968). Continued sparging with just hydrogen reduces dissolved metal fluorides and removes any excess hydrogen fluoride contained in the salt. Hydrogen sparging is not sufficient for removing chromium fluoride (Shaffer, 1971). Metallic beryllium, a strong reducing agent, must be added to purify chromium fluoride-containing molten salts (Shaffer, 1971). Beryllium fluoride is more thermodynamically stable than chromium fluoride and, therefore, beryllium will reduce chromium fluoride, (Kelleher et al., 2015). Metallic species may plate on the metal surface or become particulate, which can be removed from the fuel salt through mechanical filtration.

As the uranium fuel is consumed during reactor operations, new fuel in the form of UF_4 will be added to the fuel salt. The quantity and schedule of UF_4 additions will be determined by reactor physics needs as assessed by reactor modeling and fuel salt sampling. If the redox potential needs to be adjusted, metallic beryllium or UF_4 will be added to the fuel salt in the RAV.

The FHS is a safety-related, leak-tight system designed to receive, process, adjust, transfer, and store fuel salt. It ensures the fuel salt is enclosed in a manner such that radionuclides are contained during all processes. In the FHS, the fuel salt is maintained in geometries that prevent criticality in all conditions. The FHS ensures appropriate radiation shielding when storing irradiated fuel salt in the FHS.

During reactor operations, fuel salt samples will be collected periodically. The collection schedule will be informed by operational parameters and the results of prior samples. Depending on the impurity levels and composition of the salt samples, the necessary chemical purification processes described above will be employed to adjust the fuel salt chemistry to support continued operation.

4.5 Fuel Laboratory Testing

The Program will characterize a variety of test salts to qualify the MSRR fuel salt. The Program will measure thermophysical properties, including density, viscosity, thermal conductivity, heat capacity, enthalpy (heat) of fusion, and liquidus point, across a range of salt mixtures that vary in FP inventory and UF_4 loadings. The Program will investigate how changes to UF_4 and FP burnup concentrations affect thermophysical property values. Test plans for each thermophysical property measurement will be developed, specifying their acceptance criteria, number of data points, and test procedures.

FP surrogates will be used in the test salts to account for a variety of salt-seeking species expected to be found in the fuel salt throughout reactor operations. Table 6 shows the FP surrogate compositions to be used for the test salts. These compositions will be used to determine the influence of FPs on the thermophysical properties (McMurray et al., 2021). The results from the Program related to FP impurities in the test salts will be used to define the acceptable concentration for the fuel salt specification. The low burnup's total FP concentration of 0.7 mol % exceeds that of the expected end-of-life MSRR fuel salt. The high burnup's total FP concentration of 3.78 mol % is intended to approximate that of a commercial-scale MSR to support any future development that builds on these tests and the experience of the MSRR.

Table 6: Fission Product Surrogate Compositions

Compound	Low Burnup (mol %)	High Burnup (mol %)
ZrF_4	0.1	0.54
Mo		
NdF_3		
CeF_3		
CsF		
BaF_2		
SrF_2		
Total	**0.7**	**3.78**

Notes
1. ZrF_4 and Mo represent transition metals.
2. NdF_3 represents NdF_3, LaF_3, and YF_3.
3. CeF_3 represents CeF_3 and PrF_3.
4. CsF represents alkali metals.
5. BaF_2 and SrF_2 represent alkaline-earth metals.

Different UF_4 additions will be used in the test salts to account for the entire range of UF_4 loadings in the fuel salt throughout the lifetime of the MSRR. Table 7 shows the compositional matrix for the test salts with different UF_4 and FP burnup concentrations. The codes in the table represent the uranium concentration and either no FPs, low burnup fission products (LF), or high burnup fission products (HF). For example, the code "0U-LF" represents a test salt with 0 UF_4 mol % and low burnup fission products.

Table 7: Compositional Matrix for Test Salts

		UF_4 (mol %)					
		0	**2**	**4**	**5**	**6**	**7**
FP Burnup (mol %)	**0.0 (Fresh)**	0U	2U	4U	5U	6U	7U
	0.7 (Low)	0U-LF	2U-LF	4U-LF	5U-LF	6U-LF	7U-LF
	3.78 (High)	0U-HF	2U-HF	4U-HF	5U-HF	6U-HF	7U-HF

Table 8 shows the Program's test plan methods, temperature ranges, and the applicable standards for each thermophysical property. For each thermophysical property, the apparatus and method of measurement will be validated by comparing measured values of test salts with published thermophysical properties.

Table 8: Thermophysical Property Measurement Test Plan Overview

Property	**Test Plan**		**Standard (if applicable)**
	Method	**Temperature Range (°C)**	
Chemical composition	ICP-MS	n/a	ASTM C1287-18
	Combustion analysis	n/a	n/a
Liquid density	Archimedes bob	LP to 750	n/a
Solid density	Pycnometry	n/a	ASTM B923-22
Viscosity	Rotational rheometry	LP to 750	n/a
Thermal conductivity	Laser flash analysis	25 to 750	ASTM E1461-13
Heat capacity	Differential scanning calorimetry (DSC)	25 to 750	ASTM E1269-11
Enthalpy (heat) of fusion	DSC	25 to 750	ASTM E793-06
Liquidus point (LP)	DSC	n/a	ASTM E794-06

4.6 Fuel Performance Modeling

Fuel performance models evaluate solid fuel under normal operation and accident scenarios. These models assess factors such as fuel swelling, cracking, FP release, and the integrity of cladding of solid fuels. In solid-fueled reactors, the fuel experiences significant thermal and mechanical stresses, which can lead to degradation over time. Predicting such behaviors ensures reactor safety, optimizes solid fuel efficiency, and extends solid fuel life.

Liquid-fueled molten salt reactors, like the MSRR, operate fundamentally differently from traditional solid-fueled reactors. The fuel is dissolved in molten salt, creating a homogeneous distribution of fissile material throughout the liquid fuel. This configuration eliminates the need for solid fuel and cladding, which are subject to degradation. Liquid fuel allows for more even heat distribution, reducing associated stresses that solid fuels face. Thus, many of the degradation mechanisms in solid-fueled reactors—such as fuel cracking, thermal expansion, and cladding failure—are irrelevant to the liquid fuel in MSRs.

The MSRR container design includes the use of structural margin to preserve the primary FP boundary. This margin ensures the container does not degrade beyond what is safe. Additionally, the MSRR container degradation is carefully managed through salt chemistry control and the GMS.

For the MSRR's liquid fuel, an evaluation model that produces predictions of fuel behavior is not needed because the Program will provide experimental data on the fuel performance under normal operation and accident conditions. The data will support the evaluation of fuel and system performance in the accident analyses of the Final Safety Analysis Report.

NUREG-2246 provides guidance for fuel qualification in advanced reactors, including safety criteria that support regulatory findings related to fuel performance. It emphasizes the importance of evaluating fuel behavior under normal and accident conditions through evaluation models. Performance modeling provides a framework for assessing how a reactor's fuel can meet safety requirements and maintain integrity throughout its operational lifecycle.

Section 3.2.1 of NUREG-2246 discusses safety criterion G2.1, which is "Design Limits during Normal Operation and Anticipated Operational Occurrences." The guidance describes how fuel is expected to maintain its integrity under multiple degradation mechanisms and failure modes. The safety criterion identifies the need for a method to assess the fuel performance under normal operation and abnormal operational occurrences. Section 3.2.1.2 of NUREG-2246 identifies a goal that evaluation models be available to assess fuel performance against design limits to protect against fuel failure and degradation mechanisms. Section 3.3 of NUREG-2246 provides a framework for the assessment of evaluation models. An evaluation model is defined as an analytical tool, a computer code, or a combination of such tools. The NUREG indicates that a sophisticated tool such as a computer code may not be necessary to evaluate fuel performance. Simplified mathematical expressions or comparisons against data can serve as an evaluation model.

When the fuel is a liquid salt, the loss of structural integrity or mechanical failure of the fuel is not applicable. Maintaining a coolable geometry is a function of the reactor system rather than a function of the fuel because the liquid fuel's geometry is dependent on the characteristics of its container. Similarly, ensuring the ability to insert negative reactivity is a function of the system and cannot be affected by liquid fuel. Thus, an evaluation model for the assessment of molten salt fuel is not useful in the context of historical fuel performance models.

NUREG-2246 identifies three key capabilities that should be available in the evaluation model: geometry modeling, physical modeling, and material modeling. As a liquid, the MSRR fuel will take the shape of its container, whether in the reactor vessel or the drain tank. Similarly, physical phenomena such as strain, fatigue, or pressure do not affect fuel performance and cannot be modeled for the fuel. Many material properties are not relevant to liquid fuel in the same context as they are to solid fuels, such as cladding oxidation rate or Young's modulus. However, some properties of the liquid fuel are relevant to system behavior, such as the thermophysical properties being tested, and they should be well characterized.

The fuel performance and thermophysical properties depend on the fuel salt's chemical composition. The composition of the fuel salt may change during operation from the accumulation of FPs, the depletion of uranium, and beryllium and uranium additions. The effect on properties due to compositional changes will be assessed by the Program described in Section 4.5.

The density of the fuel salt will have a noticeable effect on reactivity. The density of the fuel salt will change with the depletion and addition of UF_4, as well as with temperature changes. Thus, the fuel salt temperature and the concentration of UF_4 will play a role in the safety function of controlling reactivity. Based on data available in the Molten Salt Thermal Properties Database–Thermophysical (Termini et al., 2023) and the data acquired during the Program, ACU will develop a table or similar tool documenting the change in density with the change in UF_4 concentration and change in temperature. This tool will support the evaluation of fuel and system performance in the accident analyses of the Final Safety Analysis Report.

The liquid fuel only plays a supporting role to MSRR systems in accomplishing the FSFs. Because the supporting role relies only on the inherent properties of the liquid fuel salt, an evaluation model of fuel performance cannot be developed in a way that is analogous to a solid-fueled reactor. The safety analysis will demonstrate the ability of the MSRR systems to achieve the FSFs with the thermophysical properties inherent in the liquid fuel salt.

5 Fuel Surveillance

As a research reactor, the MSRR will be monitored and studied while in operation. A non-safety-related scientific surveillance system will capture and store MSRR operational data. This system will provide knowledge on the critical parameters for future higher power, and potentially commercial liquid-fueled MSRs. The fuel salt and reactor off-gas will be sampled periodically.

5.1 Fuel Sampling

During reactor operations, periodic sampling will be performed to monitor the fuel salt's condition. Operational parameters and the results of prior samples will inform the collection schedule. While the FP concentration only depends on the power history, the fuel salt chemistry could change independently of the FP concentration.

Chemical analysis of fuel salt samples will provide the fuel salt's composition. Parameters to be monitored include the ratios of LiF-BeF_2-UF_4, key burnup indicators, and the presence of impurities. Other parameters that are not associated with fuel salt qualification that may be monitored include the plutonium concentration and the redox potential (e.g., UF_4:UF_3 ratio).

5.2 Cover Gas Sampling

Helium is the cover gas over the fuel salt in both the reactor system and the FHS. As an inert gas, the cover gas will not adversely affect the quality or condition of the fuel salt or its properties. Helium can be sparged through the fuel salt in the reactor system to remove the noble gas FPs as they are formed. Although cover gas sampling is not part of the fuel qualification, monitoring the gas will provide information about the health and performance of the reactor.

The GMS will control the cover gases in the reactor system and the FHS. Contaminated helium from the reactor system can be filtered to remove FPs in the off-gas clean-up system. The off-gas system uses a combination of decay tanks and adsorbent materials to remove xenon, krypton, and iodine compounds from the helium in the system.

Cover gas samples will be taken to determine the amounts and kinds of FPs found in the headspace. Activity measurements and spectroscopic measurements taken during reactor operations will be used to validate and improve simulation tools. The cover gas will be analyzed for radioactive material content using gamma spectrometers and monitored for oxygen levels with residual gas analyzers. The gas composition will be determined through gas chromatography-mass spectrometry of gas samples.

6 Fuel Handling and Storage

Fuel handling and storage processes will cover the receipt of carrier salt and fuel from DOE, storage of the salt and fuel, moving of fuel salt between systems, and ultimate shipping of fuel salt back to DOE. Fuel handling operations refer to the transfer of fuel or fuel salt between systems.

After receipt and initial storage, the carrier salt and fuel will be moved from shipping containers into the FHS for preparation and fuel salt content adjustment (i.e., UF_4 and beryllium metal addition and salt purification). When ready for operation, the fuel salt will be transferred from the FHS to the reactor system. When ACU is prepared for operations to cease permanently, the fuel salt will be moved out of the MSRR and packaged for transport back to DOE. All of these transfers between shipping containers, the FHS, and the reactor system are accomplished with pneumatic pressure using the helium cover gas.

The fuel salt may be stored in the reactor drain tank or the FHS. If extended storage is anticipated, ACU will develop maintenance procedures to monitor the fuel salt and assure its integrity.

7 Conclusions and Limitations

ACU will assess the thermophysical properties of the MSRR fuel salt for a range of compositions. Testing the thermophysical properties will confirm the design basis for MSRR's safety systems and verify that they achieve the FSFs. The results of the Program described in this report will be used to establish a fuel salt performance envelope that will be applied through operating limits specified in the MSRR Technical Specifications.

<u>Proposed Limitations</u>

Limitation 1: ACU expects to obtain fuel and MSRE salt from DOE. If DOE does not provide the fuel or MSRE salt, fuel or carrier salt of similar character will be procured by ACU. If fuel or carrier salt is not obtained from DOE, the fuel qualification activities and testing results will be evaluated for applicability to the salt obtained.

Limitation 2: The MSRR fuel salt will be qualified for its performance envelope as a result of the Program described in this report. Technical Specifications related to the safe operation of the MSRR will be established regarding the fuel salt performance envelope. If the fuel salt performance envelope is exceeded, an assessment of the safety significance and possible resolutions will be performed and reported in accordance with regulatory and Technical Specification requirements.

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