INL/RPT-24-80651

Liquid Fuel Qualification for Molten Salt Reactors: Identification of Gaps and Challenges

Advanced Reactor Technology Program

Format and Content of a Liquid Fuel Qualification Report and the Identified Gaps and Challenges to Meet Regulatory Requirements and Expectations

Milestone: M3RD-24IN0702032

SEPTEMBER 2024

David E. Holcomb
Christopher P. Chwasz
James A. King

Idaho National Laboratory

INL/RPT-24-80651

Liquid Fuel Qualification for Molten Salt Reactors: Identification of Gaps and Challenges

Liquid Fuel Qualification Report Format and Content and the Identified Gaps and Challenges to Meet Regulatory Requirements and Expectations

David E. Holcomb
Christopher P. Chwasz
James A. King
Idaho National Laboratory

September 2024

Idaho National Laboratory
Advanced Reactor Technologies Program
Idaho Falls, Idaho 83415

http://www.inl.gov

Prepared for the
U.S. Department of Energy
Office of Nuclear Energy
Under DOE Idaho Operations Office
Contract DE-AC07-05ID14517

Page intentionally left blank

INL ART Program

Liquid Fuel Qualification for Molten Salt Reactors Identification of Gaps and Challenges

INL/RPT-24-80651

September 2024

Technical Reviewer: (Confirmation of mathematical accuracy, and correctness of data and appropriateness of assumptions.)

	9/19/2024
Efe Kurt Applied Mechanics Engineer	Date

Approved by:

	9/19/2024
Michael E. Davenport ART Project Manager	Date
	9/19/2024
Michelle T. Sharp INL Quality Assurance	Date

Page intentionally left blank

SUMMARY

This report identifies and describes challenges in implementing the molten salt reactor fuel qualification process described in NUREG/CR-7299, *Fuel Qualification for Molten Salt Reactors*. This report provides additional information including examples and additional use case descriptions to assist applicants in understanding the content and structure of the information necessary to develop a fuel qualification report. In particular, the current report emphasizes the relationship of fuel salt qualification to the contents of the plant safety analysis report. This report is limited to the safety functions relevant to the reactor aspects of the nuclear power plant. Molten salt reactors that include significant fuel salt processing would also need to address the safety aspects of the processing facility.

No significant gaps were found in NUREG/CR-7299. However, NUREG/CR-7299 focuses on generic descriptions of how the fuel salt supports achievement of plant fundamental safety functions. A fuel salt qualification report will need to explicitly address the role of the fuel salt in meeting regulatory requirements for a particular plant. NUREG/CR-7299 also provides a regulatory mapping between NRC rules and the safety functions. However, by remaining at a fundamental level, NUREG/CR-7299 does not address plant-specific safety analysis. Some aspects of the safety analysis may require additional emphasis for designs that incorporate particular features (such as additional confirmation of decay heat rejection performance for higher power reactors). This report provides additional structure to enable an applicant to develop their plant-specific safety analysis.

CONTENTS

Page intentionally left blank

ACRONYMS

BDBE	Beyond Design Basis Event
CFR	Code of Federal Regulations
FSF	Fundamental Safety Function
MSBR	Molten-salt Breeder Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PIRT	Phenomenon Identification and Ranking Table
SSCs	Structures, Systems, and Components

Page intentionally left blank

Liquid Fuel Qualification for Molten Salt Reactors: Identification of Gaps and Challenges

1. PURPOSE

Fuel qualification develops the necessary evidence that fuel with properties within certain boundaries will meet regulatory requirements. Fuel qualification, thus, provides a means to incorporate safety adequacy information (that has been obtained ahead of time) by reference into a license application. Fuel qualification also provides a means for multiple, similar reactors to take advantage of common fuel performance information and thereby to increase regulatory efficiency and predictability. Fuel qualification, also, supports regulatory efficiency and consistency by pre-evaluating an acceptable method to demonstrate that nuclear fuel will fulfill its safety objectives. Without qualification, each instance in which fuel is used would have to be evaluated against regulatory requirements substantially increasing the resources required for licensing as well as the potential for subjectivity in the evaluation.

The Nuclear Regulatory Commission's (NRC's) regulations and guidance provide the means to demonstrate achievement of the fundamental safety functions (FSFs) for the qualification of reactor fuel. While the FSFs are the same for any nuclear power plant (NPP), liquid salt fuel supports achieving the FSFs via different methods than does solid fuel. Consequently, NPPs that employ liquid salt fuel need a tailored fuel qualification process reflective of the safety role that fuel plays for them.

NUREG/CR-7299, *Fuel Qualification for Molten Salt Reactors* [1] both mechanistically describes how liquid salt fuel supports achieving FSFs and provides a mapping between the regulatory requirements and measurable fuel salt characteristics. The requirement fulfillment methodology employed in NUREG/CR-7299 aligns with the objective hierarchy approach employed in NUREG-2246, *Fuel Qualification for Advanced Reactors* [2]. The objective of this work is to identify challenges with implementing the molten salt reactor (MSR) fuel qualification process that may be barriers to regulatory approval. This report is limited to liquid-fueled MSRs that employ actinide halide salts as their nuclear fuel.

2. LIQUID FUEL QUALIFICATION REPORT

2.1 Format

The following sections describe an example format for a liquid fuel qualification for an MSR (fuel salt qualification). This format is preliminary and based on NUREG-2246 as interpreted by NUREG/CR-7299 for MSRs. The intent is to provide a logical format for the information required to support qualification for a fuel salt. A fuel qualification report is a technical report provided to the regulator to support licensing under 10 Code of Federal Regulations (CFR) part 50 or 52 or the proposed part 53. The activities under a part 70 license should be contained within reports that support the corresponding license. Fuel qualification reports would consolidate data and analyses pertaining to the fuel and would support conclusions and commitments made in an applicant's safety analysis report.

2.2 Liquid Fuel Qualification Report Content

The NRC staff has defined fuel qualification in terms of its physical and chemical behavior as:

> *Fuel qualification is a process which provides high confidence that physical and chemical behavior of fuel is sufficiently understood so that it can be adequately modeled for both normal and accident conditions, reflecting the role of the fuel design in the overall safety of the facility. Uncertainties are defined so that calculated fission product releases include the appropriate margins to ensure conservative calculation of radiological dose consequences [3].*

The intent of the above was to address fuel qualification from all technologies (ceramic, metallic, and liquid). However, the intent of the fuel qualification format discussed in this report is to describe the necessary components to justify using liquid fuel in the safe operation of the nuclear power reactor. This format is based upon the bases established in NUREG-2246 and interpreted by NUREG/CR-7299. The information associated with the fuel should follow these steps:

1. A scope of the qualification is defined
2. The pertinent safety functions within the scope identified in Step (1) associated with the fuel salt are identified using a structured approach
3. Design criteria (fuel salt properties) are defined to maintain the safety functions identified in Step (2), supported by experimental data and modeling
4. Safety analysis is conducted with conservating bounding parameter values identified in Step (3) to justify fuel performance under normal and off-normal conditions identified in Step (2).

The result will be a transparent, traceable, and repeatable methodology of demonstrating compliance with the regulations and fuel qualification guidance.

2.2.1 Scope

An applicant should include a thorough discussion on all applicable aspects of the fuel that is desired to be included within the fuel qualification. Applicants may choose a scope that pertains only to the fuel salt or may include additional materials and systems that interact with the fuel salt. If an applicant chooses a smaller scope for a fuel qualification, the additional information that addresses interfacing materials and systems will still need to be provided to the regulator on the docket to support a regulatory decision for the issuance of a certification, permit, or license.

The goal of the fuel qualification report scope section is to define the applicable details of the fuel salt, fuel system, and materials in contact with the fuel salt, and operational characteristics that influence the use and storage of fresh and used fuel during normal and off-normal conditions. Information that may be within the scope of a fuel qualification are:

- Neutronics: fuel composition, coolant, moderators, reflectors, uses of excess neutrons (breeder/burner/converter, activation, commercial applications, etc.), reactivity control mechanisms
- Core description: materials, geometry, interfacing materials, radionuclide retention barriers (liquid and gaseous)
- Heat removal/addition mechanisms: systems and capabilities
- Operational lifetime of the fuel salt (timeline of fuel use, additions, additives, extractions, fuel operational end-of-life, and interim onsite storage)
- Onsite fuel preparation (under a part 50, 52, or 53 license)
- Onsite fuel processing (under a part 50, 52, or 53 license).

The defined scope sets the bounds for the structured approach to identify safety functions.

2.2.2 Safety Functions Supported by the Fuel Salt

The safety-requirements language in the Atomic Energy Act of 1954, as amended, relative to obtaining a commercial nuclear power license is “agree to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish.” The FSFs are intended to protect health and minimize danger to life and property but are not focused on complying with NRC rules. Thus, a primary challenge for employing NUREG/CR-7299 for fuel qualification is aligning the safety functions with the rules intended to provide reasonable assurance of achieving adequate safety.

NUREG/CR-7299 does, however, provide a regulatory mapping to facilitate translating the achievement of the FSFs to comply with NRC regulations.

Fuel salt partially meets the FSFs on its own. However, the fuel salt supports the other plant structures, systems, and components (SSCs) in performing their safety functions, in order for the plant as a whole to achieve the FSFs. For example, while the fuel salt is a heat transfer medium, if the heat exchangers become fouled or plugged, the fuel salt will not be able to adequately remove heat. The fuel salt retains the salt soluble nuclides that are formed within it. Fuel salt, however, has both a low solubility for the noble gases and tritium and low laminar diffusion coefficients for dissolved gases. The gaseous fission products may escape from the fuel salt. For example, ^{137}Xe has a 3.8-minute half-life and decays to ^{137}Cs. The fraction of the ^{137}Cs that is created within the fuel salt is chemically retained by the fuel salt, whereas the fraction of the ^{137}Cs that is formed in the reactor headspace will deposit locally onto surfaces above the liquid salt. Insoluble solid fission product atoms are also formed within the fuel salt during use. The insoluble fission products lack a surface tension inhibition to escape from the liquid. The insoluble, solid fission products either deposit onto salt-wetted surfaces or escape into the reactor headspace where they deposit onto surfaces.

Corrosion products may also be incorporated into the fuel salt. Corrosion products tend to have a temperature dependent solubility. Materials that are dissolved at high temperature may deposit at lower temperature, potentially plugging flow paths. Thus, fuel salt that supports adequate heat transfer for a particular container material across a particular temperature span may not provide adequate heat transfer for a different container material or across a different temperature span. Thermally driven material transport in most instances would be a slow process, so fuel salt may support adequate heat transfer for a limited period.

Fuel salt meeting the FSFs will also directly support FSFs. For example, serving as nuclear fuel is also an aspect of the fuel salt. Consequently, the fuel salt properties strongly influence the core reactivity. However, the fuel salt fissile material geometry is largely determined by its container shape, which also has a direct influence on the core reactivity. The relative homogeneity of the fuel salt also supports reactivity control, but the concentration of fissile material could challenge the stability of reactivity in loops and core regions. The fuel salt also transfers heat to nearby moderating/reflecting materials, and the moderator temperature also directly impacts the core reactivity. The fuel salt is at low chemical potential energy so it does not chemically react vigorously with any environmental material. Thus, all of the applicable plant-specific fuel-salt system performance parameters need to be included in the safety analysis report or fuel qualification report.

This section of a fuel qualification report should detail the safety functions met by the fuel salt and supported by the fuel salt, as defined in the fuel qualification report scope section. Determining the safety functions should be accomplished through a structured methodology. A design-specific phenomenon identification and ranking table (PIRT) focused on identifying failure modes would support accident progression modeling in a structured framework such as a fault tree. The identification of event sequences necessary to release radionuclides would support generation of a mechanistic source term and the characterization of SSCs needed to achieve the safety functions. Specific characterization of the fuel salt chemical and physical limiting conditions for the safe operation and accident response will be an element of developing the bounds for overall plant safety analysis.

The sections listed below map the FSFs as well as control of chemical and physical reactions for normal operations and off-normal (transient and accident) conditions met and supported by the fuel salt. The contribution of the fuel salt to the safety of the reactor and safe storage of nuclear material should be expanded upon in the safety analysis report and supporting technical documentation.

2.2.2.1 Retention of Radionuclides

The FSF of retention of radionuclides will be supported by the fuel salt during normal and off-normal conditions. In addition to serving as a partial radionuclide retention mechanism itself, the fuel salt must not inhibit any safety-related structure from adequately performing its function(s) under normal and off-normal conditions. Moreover, fuel salt must continue to mitigate the release of radionuclides even in beyond design basis event (BDBE) conditions.

For fuel qualification, describing the role fuel salt properties play in meeting the retention of radionuclides FSF should be fully described by an applicant as fresh fuel, during normal and off-normal operation, and as used fuel in interim storage. Potential adverse interactions with other plant SSCs should be described in sufficient detail to evaluate the potential for fuel-salt-induced degradation in their proper function The role of fuel salt in meeting this safety function should also be addressed for any other onsite fuel preparation or processing under a 10 CFR Part 50, 52, or 53 license. Further discussion can be found in NUREG/CR-7299, Section 2.1.

2.2.2.2 Control of Heat

The FSF of control of heat removal including decay heat from radionuclide sources would rely on the properties of the salt in the fuel-salt mixture. Used fuel salt may be a solid that has been cast into a container that is externally cooled by a non-fuel salt. The fuel salt is the primary reactor coolant and enables heat transfer via the power cycle to the ultimate heat sink. The fuel salt also has a substantial role in enabling the NPP to continue to provide adequate heat rejection under accident conditions.

The fission gases, aerosols, and vapors are also part of the fuel salt and embody a few percent of the reactor power for accidents that occur while at (or shortly following) full-power operation. However, the heat content in the off-gas decreases rapidly once the reactor becomes subcritical. Consequently, cooling the off-gas stream is a significant issue for the first couple of hours following accident initiation. Under loss-of-forced-flow conditions, the fuel salt transfers heat via buoyancy driven convection, conduction, and radiation. Consequently, fuel salt properties that impact the fuel salt flow and heat transfer (e.g., viscosity, heat capacity, and thermal conductivity) have significant impact on achieving the safety function.

The loss of an FSF often affects other FSFs. The fuel salt contains the fissile materials, so loss of heat control can also directly impact the reactivity. Under accident conditions, such as an overcooling transient, the fissile material solubility limit may be exceeded potentially resulting in a reactivity transient.

The role of the fuel salt in supporting heat rejection depends on the specific reactor configuration and the accident scenario. For accidents in which the fuel salt has leaked from its intended container, the fuel salt must not prevent the decay heat removal systems from functioning. For example, fuel salt components can evaporate from spilled salt pools and deposit onto the cool surface of heat exchangers forming in insulative snowlike layer. If fuel salt leaves the core region, the normal reactivity control mechanisms (rods, drums, etc.) will no longer be effective. The fuel salt will be thermally hot at the initiation of full-power accidents. Leaked fuel salt interaction with materials in containment could challenge the containment function via pressurization (e.g., via interaction with water) or starting a fire.

The temperature of the fuel salt can change the geometry of the core and moderator changing the overall reactivity. Radiation damage to graphite also can result in cracking and fuel salt ingress changing the amount of fissile material in core. Higher temperatures will also promote evaporation of fuel salt components. While general concepts underlying avoiding fuel-salt-induced damage to other plant SSCs can be developed from first principles, specific issues are plant and accident scenario dependent.

Similarly, loss of temperature control may impact retention of radionuclides. Tritium that is retained in-core graphite during normal operations would be released if the graphite temperature becomes sufficiently high.

Applicants should discuss the heat control mechanisms (heat removal via conduction, convection, radiation, etc., or retention of heat to maintain salt fluidity if necessary) that fuel salt properties play a vital role in, to include fresh fuel, during normal and off-normal operation, and as used fuel in interim storage. The role of fuel in meeting this safety function should also be addressed for any other fuel preparation or processing under a 10 CFR Part 50, 52, or 53 license. Further discussion can be found in NUREG/CR-7299 section 2.2

2.2.2.3 Control of Reactivity

The fundamental safety function of control of reactivity is integral to the safe use of special nuclear material. Reactivity control may directly rely on fuel behavior (i.e., inherent reactivity feedback) to meet this safety function. The fuel salt is held into a critical geometry by the core structures. Fuel salt expansion out of the core region (which provides negative reactivity feedback) is also controlled by the core structures. Radiation damage to graphite will eventually result in cracking. Fuel salt would penetrate cracks in graphite, thus changing the amount of fissile material in core and thereby the reactivity. ^{135}Xe and its ^{135}I precursor will penetrate cracks and pores in graphite introducing a time and reactor power history dependent reactivity change [4]. Thus, the reactivity control FSF will be an element of determining the in-core graphite lifetime. Transmutation may also impact the location of the special nuclear material within the fuel salt. For example, ^{35}Cl has a significant cross section to transmute to ^{35}S. Sulfur has a strong affinity to chemically bond with uranium and oxygen in the fuel salt forming an insoluble uranium sulfide or oxysulfide, which would be anticipated to either plate out onto surfaces or form a solid slurry within the fuel salt, impacting fissile material homogeneity in the fuel salt.

For fuel qualification, describing the role fuel salt properties play in maintaining this FSF should be fully described by an applicant as fresh fuel, during normal and off-normal operation, and as used fuel in interim storage. The roles of the fuel salt in meeting this safety function should also be addressed for any other fuel preparation or processing under a part 50, 52, or 53 license. Further discussion can be found in NUREG/CR-7299 section 2.3.

2.2.2.4 Control of Chemical Interactions

Though not identified as an FSF, the safety function of control of chemical interactions to maintain the physical parameters of the fuel salt should be considered a cornerstone of safety in conjunction with the FSFs. Just as the loss of an FSF can impact other safety functions, the loss of control of chemical interactions can lead to the loss of one or more safety functions. Chemical interactions would include changes to the fuel salt that could affect the parameters for the FSFs, such as chemical changes in the fuel that would change the thermophysical properties, moderation, fuel concentration in the salt, or induce plating out fuel salt components. Shifting the chemical redox state of the fuel salt will change whether tritium is chemically bound. Chemically bound tritium will not readily diffuse through containers. Shifting the fuel salt redox will also shift the corrosivity of the fuel salt potentially impacting the functionality of the fuel-salt-wetted containment layer. Also, the solubility of elements within the fuel salt (notably the actinides) is temperature dependent, for concentrations near the solubility limit, materials may solidify out of the fuel salt at lower temperatures (e.g., at the cold end of the heat exchanger). Plating out actinides could significantly adversely impact reactivity control. Similarly, fission and corrosion products may have significant temperature dependent solubility. Corrosion product accumulation at the cold end of the heat exchanger would eventually result in plugging preventing achievement of the heat removal FSF. Understanding the chemical interactions would support developing chemical concentration parameters to ensure that FSFs would not be negatively affected by normal operational changes in the fuel salt and support maintaining acceptable conditions for systems to respond to events.

For fuel qualification, describing the roles fuel salt properties play in controlling or maintaining control of chemical interactions should be fully described by an applicant as fresh fuel, during normal and off-normal operation, and as used fuel in interim storage. The role of fuel in meeting this safety function should also be addressed for any other onsite fuel preparation or processing under a 10 CFR Part 50, 52, or 53 license.

2.2.3 Design Criteria for the Fuel Salt

The basic design criteria for the fuel salt is to adequately perform its role in the plant achieving the safety functions identified above and to not sufficiently decrease the functionality of the other safety-related SSCs such that they no longer perform adequately.

As with other Newtonian fluids, the fuel salt thermal and hydraulic characteristics can be described by properties (e.g., density, viscosity, specific heat, thermal conductivity, and surface tension). The limiting fuel salt safety characteristic tends to be the ability to adequately reject decay heat under loss-of-forced-flow conditions. Fuel salt interacts with its container material via both chemical and physical means. Changes in fuel salt properties alter how the fuel salt interacts with its container. For example, changing the fuel salt redox state to an oxidative condition changes whether tellurium remains in solution in the fuel salt or deposits onto the container wall [5]. Tellurium promotes grain boundary corrosion of engineering alloys. Thus, shifting the oxidation state of the fuel salt changes not only the rate of generalized corrosion via dissolution of the most easily oxidizable alloy component (generally chromium) but switches the dominant corrosion mechanism from generalized to a much faster grain boundary. Also, while near core reactivity control mechanisms would not be credited for reactivity control following a loss-of-fuel-salt accident, containment elements (e.g., concrete) could impact the achievement of reactivity control for fast spectrum fuel salt by providing moderation.

Properties of the fuel salt will change with composition and state (primarily temperature). The range of acceptable fuel salt properties is defined by a limiting condition envelope. Combinations of fuel salt properties within the envelope result in achieving the FSFs under both normal and off-normal conditions. The properties of fuel salt are not precisely known as a function of composition and state. Consequently, the acceptable composition envelope needs to be conservatively defined to include thermophysical property value uncertainties. All safety adequacy evaluations need to be performed with conservative values of the fuel salt properties. The required values of the fuel salt properties also need to reflect the plant state. For example, if the decay heat removal heat exchanger has begun to foul, then the acceptable range of fuel salt heat transfer related properties will be narrowed to accomplish the safety function. Fuel parameters selected need to be justified through adequate experimental evidence as well as accident progression modeling.

For fuel qualification, describing the bounding parameters of the fuel salt to fulfill its role in meeting the safety functions as fresh fuel, during normal and off-normal operation, and as used fuel in interim storage. Details on how fuel parameters are maintained during site operation should be covered as part of an operational application and not within the fuel qualification.

2.2.4 Safety Analysis Bases

To support the fuel behavior and parameters needed to support safety functions, the safety analyses bases should be detailed and include data and models that support SSCs and fuel salt meeting their safety functions. The basic methodology to understand the fuel salt parameters needed to support the plant achievement of the FSFs begins with understanding the fuel salt performance requirements under normal operations and accident conditions. The fuel salt thermophysical property requirements are plant design dependent. Consequently, to determine the fuel salt performance requirements, the potential accident progression sequences for the plant under consideration need to be determined. For example, a loss-of-forced-cooling accident would initially result in a fuel salt temperature increase. The peak fuel salt temperature reached will depend on the design-specific feedback mechanisms, whereas the impact of elevated fuel salt temperature on the salt-wetted components will depend on the structural material, the power history of the fuel salt, and the extent to which short-lived fission products have been stripped from the fuel salt.

Most external events (with the notable exception of seismic events) do not directly involve the fuel salt. Seismic events will stress the fuel salt container due to the sloshing of the fuel salt. External events can trigger accident sequences. For example, lightning strikes might increase the potential for loss-of-power events.

Fuel salt qualification requires having an adequate understanding of the properties of the fuel salt to be able to model its role in the plant achieving the FSFs. While the general heat transfer and nuclear properties of liquid fuel salt are known (albeit with undesirably high uncertainties) fuel salt properties under some conditions have sufficient degrees of remaining unknowns to substantially increase accident progression modeling uncertainty.

The majority of the labile radionuclides will be in the cover gas. Hence, phenomena that increases the potential for cover gas release from its designed locations has safety implications. For example, if the vent line between the fuel salt and the first stage holdup mechanism was to become plugged and fission gas continued to be produced, pressure would build up stressing and potentially rupturing the piping. Deposits in the cover gas line can be caused by contamination buildup, aerosol generation, vapor deposition, and the plate-out of suspended solids. Off-gas lines would be anticipated to include a scraper to remove solid deposits.

Irradiated capsule tests [6], in-pile loop tests [7], and operation of the Molten Salt Reactor Experiment (MSRE) [8][9] have been employed to evaluate the quantities of radionuclides likely to be released from fuel salt. Overall, fission gases with half-lives above a few seconds along with volatiles and aerosols release from the fuel salt. Sufficient quantities of radioactive materials are included in the off-gas that their early release into wind-driven dispersion would be extremely undesirable. The non-gaseous radioactive species, however, plate out locally without transport via a carrier.

Crust formation and snowlike salt deposition during salt spill accidents have potentially significant remaining uncertainties. Accident progression in MSR designs that rely on convective heat transfer from the containment atmosphere to cool salt spills would be impacted by vaporization from the fuel salt followed by condensation on cooler heat exchanger surfaces. Salt condensate tends to form snowlike insulative deposits [10].

Release of higher temperature material into a cooler containment volume will cause the pressure to rise due to heating. The historic Molten Salt Breeder Reactor (MSBR) program included both larger [11] and laboratory scale [12] experiments to evaluate fuel salt interactions with water. Modeling the amount and rate of pressure rise (and the consequent stress on the next container) requires having a thermal model of the mixing of the released material into the cooler containment. Notably, heating gaseous environments (air or nitrogen) from ambient pressure and temperature (300 K) to 900 K would triple the pressure, so employing a container rated to above 3 atmospheres would avoid the need to develop a heating model for the fuel salt boundary.

Other mechanisms for generating pressure within containment are design dependent. Fuel salt interaction with water or other phase change media would result in pressurization. Combustion also yields gaseous products, which can substantially increase the pressure and temperature within containment. However, models for the containment resultant pressurization are largely independent of the salt composition.

MSRE safety analysis featured a maximum credible accident that was based upon a fuel salt line rupture that rapidly released all the fuel salt into containment along with sufficient water to maximize the pressure spike [13]. Note the predicted iodine releases were estimated prior to the understanding of the limited release of iodine from water developed after the Three Mile Island-2 accident [14].

MSRs have no known cliff-edge-type accident progression sequences. Adequate accident performance can be achieved under degraded conditions. For example, if the heat transfer performance of the decay heat rejection heat exchangers is less than anticipated, the fuel salt will reach a higher temperature during a loss-of-forced-flow accident, and the accident will take longer to terminate. However, fuel salt has hundreds of Kelvin of margin to boiling, and the container materials would not degrade rapidly (for hundreds of hours) at somewhat (several tens of Kelvin) higher temperature.

This section of the fuel salt qualification report should flow from the previous sections: a scope of the qualification is defined, the pertinent safety functions associated with the fuel salt are identified using a structured approach, design criteria (properties) are defined to maintain the safety functions, and the safety analysis is conducted with conservative bounding parameter values to justify fuel performance under normal and off-normal conditions.

3. LIQUID FUEL QUALIFICATION GAPS AND CHALLENGES

3.1 Safety Analysis

Applying the safety adequacy methodology described in NUREG/CR-7299 requires having sufficient fuel salt property data for the specific fuel salt used over the full range of its potential use and accident conditions to be able to evaluate its role in the NPP achieving the FSFs. Any salt property data uncertainties and/or gaps need to be presumed to exhibit the least favorable properties for conservative estimation of accident consequences. The fuel salt property database currently being developed by DOE-NE [15] is currently incomplete and lacks well-bounded uncertainties for some fuel salts. The information gaps and data uncertainties result in substantial conservatism in modeling accident progression for some fuel salts.

The fuel salt property database also does not cover the full potential range of fuel salt situations, so it does not permit the calculation of fuel salt potential contributions to plant safety in all situations. For example, used fuel salt storage may result in the buildup of radioactive gases due to radiolysis.

The fuel salt is an element that contributes to the plant mitigating BDBEs. The fuel salt properties depend on the composition. If the fuel salt containment ruptures, then the fuel salt will be able to interact with local environmental materials. The specific environmental materials will depend on the plant design. The alkali halides are in a low chemical energy state, so they will not react vigorously with any environmental materials. However, the properties of fuel salt containing (dissolved or in a slurry) substantial burdens of some potential environmental materials such as oxide insulators remain largely unknown. For example, if the leaked fuel salt were to become much more viscous, it would no longer readily flow to drain tanks. Plant designers would need to consider which materials the fuel salt might interact within BDBEs to be able to acquire relevant fuel salt property data and perform accident progression modeling.

Fuel salt will have different safety functions depending on its current role in plant operation. Property measurements that provide needed performance information during use may not be adequate to represent fuel in other states. For example, used fuel salt that is entering long-term storage may no longer be liquid, so it would have similar cooling issues as other used nuclear fuels. While liquid fuel salts are largely immune from radiolysis due to the characteristics of ionic liquids, once the fuel is sufficiently deeply frozen, radiolysis can become important. Both chloride and fluoride salts have potential species that may evolve out and pressurize containers (notably Cl_2, F_2, and UF_6) and/or leak radionuclides.

Fuel salt must continue to support achievement of the FSFs while the plant is undergoing maintenance. The fuel salt may be processed to maintain its composition (and resultant properties) within acceptable bounds. The fuel salt will generate substantial decay heat (that decreases following a used fuel curve) following shutdown. Separated byproduct materials will generate significant quantities of decay heat. The byproduct materials would be considered as part of the fuel until they have been adequately separated from the actinide materials. Consequently, consideration of adequate cooling of the fuel salt during processing will be an element of fuel salt qualification. Providing adequate reactivity control will also be an element of fuel salt processing for both thermal spectrum fuel salts, as neutron poisons are removed, and fast spectrum fuel salts, due to the potential for increased neutron moderation.

Fissile and/or fertile materials may need to be added or removed from the fuel salt over time to maintain reactivity within control limits. Fresh uranium- or thorium-based fuel concentrate (before being incorporated into fuel salt) has limited safety implications when located outside of the fuel salt system. Fuel derived from previously used fuel would generally contain sufficiently radioactive materials to have potential localized safety issues but would not be anticipated to impact achievement of the NPP FSFs. Bulk storage of fuel salt concentrate will, however, have the potential for inadvertent criticality, so it will need to be addressed in the plant safety analysis report. The composition of the fuel salt concentrate needs to be adequately well known so that the overall fuel salt composition remains within its acceptance envelope.

Used fuel salt no longer being used to produce power will require cooling. The amount of cooling will depend on its use history and the extent to which byproduct materials have been stripped from the fuel. Once the fuel salt has solidified, its safety issues will be similar to those of solid-fuel reactors. The NPP fuel salt safety analysis will extend until the fuel salt is transferred to an independent spent fuel storage facility.

While the properties of the fuel salt will impact the performance of nearby materials (e.g., the amount of power being produced will determine the radiation flux on nearby materials), the performance of SSCs not in contact with the fuel salt would not be part of fuel salt qualification but addressed separately in the plant safety analysis.

Means to detect fuel-salt-related changes that could impact achievement of the FSFs would also be part of plant safety analysis. Fuel salt heat transfer and reactivity properties would ordinarily be anticipated to be monitored online, while in use. The overall fuel salt heat transfer performance would be determined by performing a plant heat balance which, along with fuel salt pump performance, provides evidence of the overall fuel salt heat transfer properties. Similarly, changes in the fuel salt reactivity during normal operation would be determined by a combination of neutron flux monitoring and temperature monitoring along with control element position monitoring.

Fuel salt property measurements to account for the above conditions, operational schemes, and events are needed for modeling capable of verification and validation to support operational and accident analyses.

3.2 Measurement

3.2.1 Fuel Salt Property Measurement

Fuel salt qualification requires employing conservative values of fuel salt properties in accident progression models to establish conservative values for potential radionuclide releases. Techniques for molten salt property measurements remain immature, lacking well-bounded uncertainties. While work is in progress to develop fuel salt property measurement techniques with bounded measurement errors, consensus standards with quantified uncertainties are not yet available for significant properties [16]. Applicants will either need to accommodate the parameter uncertainty in their design and operations by incorporating larger design margins in other plant SSCs or reducing the fuel salt property measurement uncertainties prior to submitting their license application.

3.2.2 Fuel Salt Property Data Availability

The U.S. DOE has been sponsoring development of a database on halide salt properties for the past few years that includes fuel salts. The database contains fundamental physical and chemical property data and is publicly available. The database incorporates a quality ranking system intended to provide an indication of the confidence in particular values [17]. Database development is ongoing. Fuel salts within the LiF-BeF_2-UF_4 family have substantially more performance evidence available due to their use during the historic MSBR program. Fuel salt property data availability and acceptability will need to be evaluated at the time of licensing.

3.2.3 Timing

The fuel salt properties will change over time due to salt composition evolution due to the buildup of fission products with use as well as contaminants, activation products, and corrosion products. Fissile and fertile material will also be consumed and bred within the fuel salt. The liquid fuel salt will also release gases, vapors, and aerosols over time. Insoluble fission products will plate out onto salt-wetted surfaces and subsequently diffuse into them. Consequently, the salt composition and therefore its properties will continuously change.

The acceptable fuel salt property envelope will be determined based on acceptable performance under normal and accident conditions. Changes in fuel salt heat transfer properties may not be apparent in normal operations for designs that include a fuel salt pump as changes in the pumping power could produce equivalent changes in heat transfer performance.

The acceptable fuel salt parameter envelope depends on the performance requirements. Fuel salt directly has a role in achieving the FSFs and cannot damage safety-related SSCs such that they cannot adequately perform their safety functions. Thus, the damage rates and mechanisms for fuel salt interacting with the safety-related SSCs partially determine acceptable fuel salt parameters. Sufficient neutron flux tends to embrittle metals while high temperatures soften metals increasing their creep rates. Thus, the acceptable fuel salt property envelope depends on the details of the plant design and its operating history.

Some reactor parameters are sensitive to small changes in fuel salt composition (e.g., reactivity), while other parameters (e.g., heat transfer performance) require more substantial changes in fuel salt composition to have a significant impact. The fuel salt parameter measurement frequency depends on both the plant sensitivity to the parameter and potential rate of change in the parameter. For example, the fuel salt fissile concentration might be measured each time that fissile material is added or removed by observing the reactor response to a time varying reactivity input (e.g., transient change in regulation rod position).

The fuel salt thermophysical properties combine to provide adequate heat transfer and reactivity control. Consequently, the combined set of thermophysical properties is important to provide adequate safety, whereas specific parameter values are of lesser importance. For example, a disadvantageous change in the fuel salt viscosity could be balanced by an advantageous change in its thermal conductivity to result in overall acceptable fuel salt heat transport. Changes in the overall fuel salt heat transfer characteristics may be semi-continuously monitored by observing the combination of pump power, reactor neutron power, fuel salt temperature, and heat transfer to the secondary coolant.

The importance of some fuel salt properties shifts with changes to salt contacting material properties. For example, fuel salt surface tension becomes more important to graphite penetration (reactivity control) with increasing radiation damage (crack growth).

3.3 Fuel Salt Interactions with Materials

3.3.1 Chemical

The fuel salt will chemically interact with the materials it contacts. Corrosion and insoluble material (i.e., insoluble fission products) deposition are primary chemical interaction mechanisms. Halide salts can dissolve protective oxide layers on the surface of alloys. Consequently, fuel salt container alloys need to rely on thermodynamic stability to avoid unacceptable corrosion. Halide salts can exhibit multiple different corrosion mechanisms. The corrosion may be small (microns per year scale) or severe depending on the details of the chemical interactions.

Oxidative corrosion occurs when a metal atom (in it most reduced state) on the surface of the salt container alloy is oxidized into a soluble salt molecule. For example, metallic chromium can interact with fluorine atoms within the salt melt to form dissolved CrF_3. The rate at which oxidative corrosion occurs depends on the composition of the alloy (with iron and chromium being more easily oxidizable than nickel), the temperature, and the oxidation state (redox) of the fuel salt. The oxidation state of the fuel salt depends on its composition and temperature with electronegative impurities increasing its oxidation potential.

Uranium in the fuel salt has two common oxidation states, 3+ and 4+, with the 4+ state being more oxidizing. Fuel salt that is in contact with graphite cannot be allowed to become too reducing (i.e., to have a high fraction of UF_3). The UF_3 would chemically bind with graphite to form uranium carbide ($4UF_3+2C \rightleftharpoons 3UF_4+UC_2$) [18]. Similarly for chlorides, UCl_4 is highly oxidizing while UCl_3 is vulnerable to disproportionation [19], which yields dissolved metallic uranium ($4UCl_3 \rightleftharpoons 3UCl_4+U^0$) at high temperatures [20]. Alloying dissolved uranium with the structural alloy is energetically favorable [21].

Galvanic corrosion (also referred to as dissimilar metal corrosion) is an oxidative electrochemical process in which one metal corrodes preferentially when it is in electrical contact with another in the presence of an electrolyte (in this case molten salt). The more reactive metal acts as an anode and is corroded more rapidly while the less reactive metal acts as a cathode at which corrosion is inhibited. While all molten halide salts are electrolytes and so participate in galvanic corrosion, galvanic corrosion is not significantly changed by fuel salt composition changes apart from redox state so is not directly an element of fuel salt qualification. Galvanic corrosion is ordinarily suppressed in fuel salts by maintaining the fuel salt redox condition to be more reducing than the potential difference between the two metals.

Some fission products have redox dependent solubility in the fuel salt. Notably, tellurium deposited onto the surface of container alloys can result in intergranular surface cracking [22]. However, tellurium under sufficiently reducing conditions forms an innocuous, soluble chrome telluride ($UF_4/UF_3 < 60$) [23]. Overall, fuel salt must be maintained within a redox window to minimize oxidative corrosion.

Fuel salt can also participate in non-oxidative corrosion of salt-wetted materials. The surface atoms of the fuel salt container have a non-zero probably of dissolving into the fuel salt. Refractory atoms (e.g., W or Mo) are not significantly vulnerable to dissolution into fuel salt and have been considered as corrosion resistant coatings for structural alloys in fuel salt [24]. The atomic solubility depends both on the salt composition and its temperature. Materials that dissolve out of the alloy at high temperature may deposit in the lower temperature portion of the fuel salt loop. For some salt compositions (e.g., LiF-BeF_2-UF_4), the temperature dependence of the dissolution reactions from the structural alloy is so small to result in negligible thermally driven non-oxidative deposition. Other fuel salt compositions (e.g., NaF-LiF-UF_4) have significantly larger temperature dependent corrosion product solubility and can result formation of structural material deposits in colder regions [25].

The acceptable quantity of impurities in fuel salt are based upon their impact on the fuel salt performance. Some impurities have the potential to adversely impact multiple aspects of achieving the FSFs. For example, sulfur is an electronegative impurity, which increases the fuel salt redox and thereby its corrosivity and thereby challenges the containment FSF. Sulfur also chemically bonds with uranium to form insoluble uranium sulfides, potentially impacting both the reactivity and heat removal FSFs. Other elements such as cesium have only small impact on the fuel salt performance until reaching much higher concentrations. The fuel salt qualification data will need to include sufficient information on likely impurities to be able to evaluate their impact on plant achievement of the FSFs.

3.3.2 Thermal

The fuel salt will transfer thermal energy to and from materials it touches. The thermal energy may degrade the properties of the contacted materials. For example, the normal operation container material will decrease in strength at higher temperatures. Decreased strength will facilitate increased material creep. The high fuel salt temperature may also impact the effects of radiation damage. For example, helium atoms generated by neutron interactions within structural alloys can become mobilized at higher temperatures. Helium atoms that migrate to grain boundaries and coalesce into bubbles can embrittle the structural alloy. However, fuel salt composition only has a secondary impact on the temperature of its container and thus the embrittlement and increased creep would not be elements of the fuel salt qualification.

Fuel salt may be at a high enough temperature to cause ignition if it contacts both combustible materials and oxygen. Accidents in which the fuel salt leaves its intended container will be elements of the plant safety evaluation. The fuel salt cannot become so hot that it prevents other safety-related SSCs from performing their functions. Evaluating the set of materials that the fuel salt might contact will also be an element of plant safety analysis. However, the fuel salt properties that might result in damage to safety-related SSCs are not significantly salt composition dependent.

Higher temperatures will result in enhanced volatilization from the fuel salt. Gases and vapors are more mobile than the liquid fuel salt. Fission gases, aerosols, and vapors may contain up to ~1% of the reactor full power with the heat load reducing rapidly for the first several hours [26]. Providing adequate cooling for the fission gases will be a part of the reactor safety evaluation. However, higher temperature has minimal impact on the fission gas heat load, so the increased temperature would not be an element of fuel salt qualification.

The fuel salt heat transfer safety function is generally improved by higher temperature—decreased fluid viscosity along with increased radiative and conductive heat transfer. Conservative values for heat transfer performance modeling can be obtained by evaluating the fuel salt properties at accident initiation from full-power operation.

3.3.3 Physical

High velocity fluid fuel salt flow can cause damage to its container through momentum transfer (stress) and surface erosion. Significant erosion of structural alloys requires materials harder than the alloy surface within the moving fuel salt. Hard materials may be introduced into the salt through gross failures of components (sensor sheath breaking off or pump bearing failure). However, these would be separate accidents evaluated independently of the fuel salt properties. If the fuel salt were to become sufficiently reducing for its UF_3 to form uranium carbide on the surface of the core graphite, the uranium carbide surface layer could exfoliate from the graphite to form suspended hard pieces in the fuel salt. Maintaining the fuel salt redox condition within acceptable bounds to not damage either the graphite (too reducing), plate out uranium onto surfaces (too reducing for systems without graphite), or oxidizing the container alloy (too oxidizing), will be a safety-related aspect of reactor control.

3.3.4 Graphite

Graphite is vulnerable to radiation damage. MSRs will require an appropriate grade of graphite such that fuel salt does not significantly penetrate its pores. Thermal spectrum breeder reactors will also need to restrict the penetration of ^{135}Xe into the graphite pores. The wetting characteristics of the fuel salt on graphite will be temperature dependent—wetting increases at higher temperatures. Radiation damage increases the size of the cracks within graphite. The wetting characteristics of fuel salt are dependent on its composition with moisture contamination increasing the graphite wetting [27]. The combined impact of fuel salt wetting/penetration into graphite pores along with radiation damage of the graphite increasing the size and depth of cracks within the graphite can impact the FSF or reactivity control by altering the amount of fissile material in the core. The fuel salt acceptable property envelope will (for designs in which the fuel salt contacts graphite) depend on the amount of radiation damage the graphite has undergone. A graphite damage specification would be an element of the plant's technical specifications unless sufficient data is available to show that the plant continues to achieve the FSFs independent of the graphite radiation damage state.

3.3.5 Connected Systems

Fuel salt can damage connected systems, and connected systems can alter the fuel salt properties sufficiently to result in unacceptable damage. For example, deposits from the gases, vapors, and aerosols released from fuel salt will produce heat due to radioactive material decay. The decay heat can damage SSCs. Also, connected systems may add materials to the fuel salt that deleteriously impacts its properties. For example, impulse lines are a common means to separate instruments from high-temperature, high-radiation environments. Liquid metals (e.g., NaK) are a typical impulse line fill material. If the diaphragm separating the liquid metal from the fuel salt were to fail, the liquid metal would cause the fuel salt to become much more reducing—potentially allowing the uranium within the fuel salt to attack the graphite or deposit onto the container alloy. However, the accident sequences, while involving the fuel salt, are not strongly impacted by the status of the fuel salt (following initial start-up). Diaphragm rupture or fission gas cooling failure on the second day of operation would have similar impacts as that years later. Consequently, the potential for connected system failure would be part of the overall plant safety evaluation but not an element of fuel salt qualification.

4. CONCLUSION

Fuel qualification reports consolidate data and analyses pertaining to the fuel support conclusions and commitments made in an applicant's safety analysis report. Liquid fuel systems, like those presented in liquid fueled MSRs present higher level of difficulty to justify performance and fulfilment of facility safety functions. The work presented here demonstrated a structured and traceable approach to justify fuel salt contributions to meeting safety functions though experimental data and modeling, through defined operational parameters.

Additionally, challenges to a fuel salt qualification report were identified. Some challenges are currently being addressed through the DOE Advanced Reactor Technologies Molten Salt Reactor Program, such as the fuel salt thermophysical database and thermochemical database development work. However, fuel salt interactions with materials (fuel salt boundary chemical, thermal, and physical effects, interactions with graphite) are not as well understood and will need to be supported through experimentation and modeling, as well as fuel salt behavior throughout the facility lifetime. Finally, consistent fuel salt qualification would be best supported through a thorough tailored methodology to identify the necessary safety functions fulfilled or supported by the fuel salt.

5. REFERENCES

1 Holcomb, D., W. Poore, and G. Flanagan. 2022. *Fuel Qualification for Molten Salt Reactors*. NUREG/CR-7299. U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2233/ML22339A161.pdf.

2 Drzewiecki, T., J. Schmidt, C. VanWert, and P. Clifford. 2022. *Fuel Qualification for Advanced Reactors*. NUREG-2246. U.S. Nuclear Regulatory Commission. https://www.nrc.gov/docs/ML2206/ML22063A131.pdf.

3 U.S. Nuclear Regulatory Commission. 2017. Public Meeting on Improvements for Advanced Reactors, August 3, 2017, Washington, DC, https://www.nrc.gov/docs/ML1722/ML17220A315.pdf.

4 Scott, D. and W. P. Eatherly.1970. "Graphite and Xenon Behavior and their Influence on Molten-Salt Reactor Design, Nuclear Applications and Technology." 8:2, 179-189, DOI: 10.13182/NT70-A28624.

5 Surenkov, A., et al. "Effect of the [U (IV)]/[U (III)] ratio on selective chromium corrosion and tellurium intergranular cracking of Hastelloy N alloy in the fuel LiF-BeF2-UF4 salt." *EPJ Nuclear Sciences & Technologies* 6 (2020): 4.

6 Blankenship, F. F., S. S. Kirslis, and H. A. Friedman. *In-Pile Irradiation Experiment 47-6 in Reactor Chemistry Division Annual Progress Report for Period Ending January 31, 1965*. ORNL-3789 p. 36-45.

7 Compere, E. L, E. G. Bohlmann, H. C. Savage, and J. M. Baker. *Molten-Salt Irradiation Experiments in Reactor Chemistry Division Annual Progress Report for Period Ending December 31, 1967*. ORNL-4170, p. 22-32.

8 Kirslis, S. S. and F. F. Blankenship. *Hot-Cell Tests on Fission Product Volatilization from Molten MSRE Fuel in Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 29, 1968*. ORNL-4254, p. 100-113.

9 Kirslis, S. S. and F. F. Blankenship. *Fission Product Volatilization Tests in Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 28, 1969*. ORNL-4396, p. 145-153.

10 Williams, D. F., L. M. Toth, and K. T. Clarno. *Assessment of Candidate Molten Salt Coolants for the Advanced High-Temperature Reactor (AHTR)*. March 2006, ORNL/TM-2006/12, DOI: 10.2172/885975.

11 Mann, L. A. 1955. *ART Reactor Accident Hazards Tests*. ORNL-CF-55-2-100.

12 Briggs, R. B. *Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 28, 1962*, Section 6.5.1 Physical Effects of Mixing Molten Fuel and Water, ORNL-3282, p. 124.

13 Beall, S. E., P. N. Haubenreich, R. B. Lindauer, and J. R. Tallackson. *MSRE Design and Operations Report*, Part V, Reactor Safety Analysis Report, Section 8.7.2 Release of Activity after Maximum Credible Accident, ORNL-TM-732, August 1964.

14 Pelletier, C A, Voilleque, P G, Thomas, C D, Daniel, J A, Schlomer, E A, & Noyce, J R. Preliminary radioiodine source-term and inventory assessment for TMI-2, GEND-028, March 1983, https://doi.org/10.2172/6460274.

15 "Molecular Science Data Hub." Oak Ridge National Laboratory. Accessed September 9, 2024. https://mstdb.ornl.gov.

16 Ebert, W. L., and Rose, M. A. 2021, "Data Quality of Salt Property Measurements", ANL/CFCT-2118, https://doi.org/10.2172/1880777.

17 Rose, M. A. 2023. *Quality Ranking of Unary Fluoride Salt Property Data in MSTDB-TP*. ANL/CFCT-23/48. DOI:10.2172/2278997.

18 Toth, L. M. and L. O. Gilpatrick. 1972. *The Equilibrium of Dilute UF3 Solutions Contained in Graphite*. ORNL-TM-4056. DOI: 10.2172/4605940.

19 Harder, B. R., G. Long, and W. P. Stanaway. “Compatibility and Processing Problems in the Use of Molten Uranium-Alkali Chloride Mixtures as Reactor Fuels.” *Nuclear Metallurgy*, Metallurgical Society of the American Institute of Mining, Metallurgical and Petroleum Engineers, 15 (1969) 405-32.

20 Koger, J. W., 1972. *Alloy Compatibility with LiF-BeF2 Salts Containing ThF4 and UF4*. ORNL-TM-4286. DOI:10.2172/4381831.

21 Thamer, B. J. 1966. *Corrosion Tests of Hastelloy N, Inconel 600, and Hymu-80 Exposed to UCl3-KCl at 900 °C*. LA-3476-MS. DOI: 10.2172/4556216.

22 McCoy, Jr., H. E. 1978. *Status of Materials Development for Molten Salt Reactors*. ORNL-TM-5920. DOI:10.2172/5195742.

23 Keiser, J. R. 1977. *Status of Tellurium-Hastelloy N Studies in Molten Fluoride Salts*. ORNL-TM-6002. DOI:10.2172/7295251.

24 Federer, J. I., and Poteat, L. E. 1971. *Study of the Adherence of Tungsten and Molybdenum Coatings*. ORNL-TM-3609. DOI:10.2172/4677916.

25 Briggs, R. B. 1964. *Molten-Salt Reactor Program Semiannual Progress Report for Period Ending July 31, 1964*. ORNL-3708, pp. 241-244. DOI:10.2172/4676587.

26. Smith, A. N. and M. Bell. 1969. *Distribution of Decay Heat in the MSBR Off-Gas System.” In Molten-Salt Reactor Program Semiannual Progress Report for Period Ending February 28*. ORNL-4396, pp. 94–95.

27. Briggs, R. B. *Molten-Salt Reactor Program Semiannual Progress Report for Period Ending July 31, 1963*. ORNL-3529, p. 125-129. DOI:10.2172/4106714.