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Technical Assessment of Storage Options for Molten Salt Reactor Waste

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TECHNICAL ASSESSMENT OF STORAGE OPTIONS FOR MOLTEN SALT REACTOR WASTE— FINAL REPORT

Prepared for

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EXECUTIVE SUMMARY

To prepare for safety reviews and pre-licensing interactions for molten salt reactor (MSR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff is evaluating available information to identify potential storage technologies that could be used by a potential licensee to manage irradiated fuel salt wastes generated from liquid-fueled MSR designs. To support that effort, CNWRA has conducted an assessment of potential safety and technical considerations for interim storage of liquid-fueled molten salt reactor (MSR) fuel salt wastes. The assessment, documented in this report, considers publicly available technical information on MSR fuel salt waste characteristics, different levels of processing, potential waste forms and storage methods, and potential storage technologies. The assessment considered both fluoride and chloride salts.

Waste characteristics and waste forms depend on the level of post-discharge processing. Three potential levels of post-discharge processing of MSR fuel waste were assessed: (i) no processing (unprocessed and unstabilized), (ii) stabilized unseparated waste, and (iii) stabilized separated waste. The specific materials that would remain in waste streams after processing depend on the processing that would be applied, including any extractions performed for reuse of materials, which would remove materials from certain waste streams. Similarly, the choice of waste form depends on factors such as the MSR reactor design, fuel composition, degree of processing based on fuel cycle, and associated waste streams and material compositions. Glass, ceramic, and metallic composite forms have been primarily identified to immobilize MSR fuel salt waste. The unprocessed and unstabilized fuel salt waste form can potentially be stored for decades after discharge. The advantage of not processing the waste immediately is that there is some flexibility for transferring the fuel salt waste from one containment to another, or for further processing at a future time. Material characteristics of unprocessed fuel salt waste suggest potential technical considerations, such as the potential for gaseous radiolysis products; historical challenges in storing unprocessed discharged MSR fluoride-based fuel salt waste at MSRE have been documented (Peretz, 1996). The characteristics of the unprocessed and unstabilized forms suggest that practical deployments would proceed with further processing options (e.g., stabilization, separations) to achieve a more durable and manageable long-term waste form.

An assessment of potential safety and technical considerations for the safe interim storage of MSR fuel salt wastes and associated waste forms was also conducted. The report provides a better understanding of MSR fuel waste radionuclide inventories for various design and operation options, as well as information on materials properties such as radiation effects on potential waste forms .

REFERENCE

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

ABBREVIATION/ACRONYMS

ARPA-E	US Department of Energy's Advanced Research Projects Agency-Energy
ASTM	American Society for Testing and Materials
CFR	Title 10 of the *Code of Federal Regulations*
CNWRA®	Center for Nuclear Waste Regulatory Analyses
CURIE	Converting UNF Radioisotopes into Energy
CWF	ceramic waste form
DOE	U.S. Department of Energy
DSWF	dehalogenated salt waste forms
DSF	dry storage facility
DSS	dry storage system
EPRI	Electric Power Research Institute
Fe-P-O	iron phosphate
GBS	glass-bonded sodalite
GBZ	glass-bonded zeolite
halmet	halide-metal composites
INL	Idaho National Laboratory
LABS	lanthanide borosilicate
LWR	light-water reactor
MCRE	Molten Chloride Reactor Experiment
MSRE	Molten-Salt Reactor Experiment
MSR	Molten Salt Reactor
NAS	National Academy of Sciences
NRC	U.S. Nuclear Regulatory Commission
NWTRB	Nuclear Waste Technical Review Board
ONWARDS	Optimizing Nuclear Waste and Advanced Reactor Disposal Systems

ORNL	Oak Ridge National Laboratory
Pb-Te-O	lead tellurite
RH-TRU	remote handled-transuranic
SAP	silica aluminophosphate
SNF	spent nuclear fuel
SSC	structures, systems, and components
SS	stainless steel
UNF	used nuclear fuel
USHYZ	ultrastable H-Y zeolite
ZIT	zinc-in-titania

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QUALITY OF DATA, ANALYSES, AND CODE DEVELOPMENT DATA

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

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

1 INTRODUCTION

To prepare for safety reviews and pre-licensing interactions for molten salt reactor (MSR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff is evaluating available information to identify potential storage technologies that could be used by a potential licensee to manage irradiated fuel salt wastes generated from liquid-fueled MSR designs. Unlike light-water reactors (LWRs), liquid-fueled MSRs use molten salt as the coolant and fuel, and implementation of MSRs would require management of different waste streams. Potential challenges include the technology for providing canisters and casks to be used for long-term storage of the waste from these advanced reactors, as well as the potential variety of waste streams and waste forms that could result from processing options described in the technical literature.

The various MSR designs have different fuels and processing options, leading to a variety of potential waste compositions and forms. Previous reports prepared by the Center for Nuclear Waste Regulatory Analyses (CNWRA®) for the NRC (Adams et al., 2023; LaPlante et al., 2024; 2025) describe the historical experience with MSR technology, current MSR designs and processing concepts, applicable information from technical reports, and potential challenges and information needs associated with storage, transportation, and processing. This report extends that effort by further exploring possible MSR waste streams and associated waste forms and their characteristics, possible storage methods, and potential technical considerations and information needs for the safe storage of these materials.

Chapter 2 of this report summarizes the MSR fuel salt waste characteristics associated with different levels of processing and possible MSR fuel salt waste forms and storage methods. Chapter 3 describes potential safety and technical considerations for interim storage of MSR fuel salt wastes. Chapter 4 provides a summary of the report.

2 POTENTIAL FUEL SALT WASTE FORMS AND STORAGE METHODS FOR MOLTEN SALT REACTORS

2.1 Fuel Salt Waste Characteristics

A variety of formulations for MSR fuel salt have been described (McFarlane et al., 2019). Designs and fuels are typically grouped by (i) fuels composed of fluoride salts and (ii) fuels composed of chloride salts (NAS, 2022; Arm et al., 2020; McFarlane et al., 2019; Riley et al., 2018). Fluoride salt designs exist for both thermal neutron spectrum and fast neutron spectrum reactors, with thermal neutron spectrum reactors currently nearer utilization in the United States (Adams et al., 2023). Chloride salt designs are all intended to operate in the fast neutron spectrum (Adams et al., 2023), due to high neutron absorption by chlorine in the thermal spectrum (He et al., 2023). Example fuel salt chemistries based on the literature on fluoride-based and chloride-based salts are shown in Table 2-1.

Table 2-1. Example fuel and coolant chemistries for reactor designs considered in this assessment

Reactor Design Concept	Example Reactor	Fuel Chemistry	Coolant Chemistry	Reference
Fluoride-based Thermal Spectrum	Molten Salt Reactor Experiment	$^{7}LiF-BeF_{2}-ZrF_{4}-UF_{4}$ (65-29-5-1 mol%, respectively)	$2(^{7}LiF)-BeF_{2}$	Riley et al. (2018) Serp et al. (2014)
Chloride-based Fast Spectrum	Molten Chloride Reactor Experiment	$UCl_{3}-NaCl$ (36-64%, respectively)	$UCl_{3}-NaCl$ (33-67%, respectively)	Latkowski (2021) DOE (2023) Walls and Spencer (2025)

DOE "Final Environmental Assessment for the Molten Chloride Reactor Experiment (MCRE) Project", INL/RPT-22-68976, August 2023.
Riley, B., J. McFarlane, G. DelCul, J. Vienna, C. Contescu, L. Hay, A. Savino, and H. Adkins. "Identification of Potential Waste Processing and Waste Form Options for Molten Salt Reactors." NTRD-MSR-2018-000379.
PNNL-27723. 2018.
Serp, J., M. Allibert, O. Beneš, S. Delpech, O. Feynberg, V. Ghetta, D. Heuer, D. Holcomb, V. Ignatiev, J.L. Kloosterman, L. Luzzi, E. Merle-Lucotte, J. Uhlír, R. Yoshioka, and D. Zhimin. "The Molten Salt Reactor (MSR) in Generation IV: Overview and Perspectives." Progress in Nuclear Energy 77 (Supplement C):308-19. 2014. <https://doi.org/10.1016/j.pnucene.2014.02.014>
Latkowski, J. "TerraPower's Molten Chloride Fast Reactor (MCFR)." TerraPower. February 22, 2021.
<https://www.nationalacademies.org/documents/embed/link/LF2255DA3DD1C41C0A42D3BEF0989ACAECE3053A6A9B/file/DB0D308269688B2BD7B1AF60BAA143D48890C2DE80BB?noSaveAs=1> (Accessed 22 September 2022).
Walls, M. and K. Spencer. "Molten Chloride Reactor Experiment." INL/MIS-23-73611-Revision-0. July 2023.
<https://inldigitallibrary.inl.gov/sites/sti/sti/Sort_66862.pdf> (Accessed 19 February 2025).

As with other nuclear reactor fuels, the fission reaction and other processes following from irradiation in a reactor generate a wide variety of additional radioactive isotopes. In a nuclear physics context, these materials are typically classified as actinides, transuranics (actinides with atomic numbers greater than 92), fission products, and associated decay products, as shown in the box to the right of this page. These three classifications are typically used in the regulation and management of radioactive wastes. Because chemistry is central to MSR fuel salt waste processing concepts, element sets that are based on chemical characteristics (addressed in the structure of the periodic table) are typically considered in waste processing research in addition to actinides. These additional element sets include noble metals (Ru, Rh, Pd, Ag, Os, Ir, Pt, Au), halogens (F, Cl, Br, I, At), rare earths (Sc, Y, La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu), alkali (Li, Na, K, Rb, Cs, Fr), alkali earths (Be, Mg, Ca, Sr, Ba, Ra), and lanthanides (rare earth elements from La to Lu). These element groups encompass various fission products and their associated decay products. Fission products in irradiated reactor fuel contribute to high radiation fields that typically require controls to maintain radiological safety.

Riley et al., 2021 note that the physical, chemical, and radioactive characteristics of the waste stream are important in designing waste forms. For halide salt wastes, important characteristics highlighted for waste form formulation and associated processing include:

Radioactive Materials in Irradiated Reactor Fuel

Actinides include Ac, Th, Pa, U, Np, Pu, Am, Cm, Bk, and Cf. Many actinide isotopes have long half-lives relative to the half-lives of most fission products. (This list does not include the higher-atomic number actinides with short-lived isotopes.)

Transuranics are a subset of actinides beyond U (Np, Pu, Am, Cm, Bk, and Cf) in the periodic table.

Fission products and their associated **decay products** are numerous but include, for example, isotopes of Sr, Tc, I, and Cs. Most fission products have short half-lives relative to actinides. A notable exception is the fission product Tc-99, which has a half-life of 210,000 years.

- Decay heat
- Concentrations of total halogen, alkali, alkaline earths, rare earths, transition metals, U, Pu, and other actinides
- Ratios of halogens (e.g., Cl:F:I) and alkalis (e.g., Li:Na:K:Cs) and, to a lesser extent, alkaline earths, rare earths, and transition metals
- Concentrations of long-lived isotopes (e.g., I-129, Cl-36, Se-79, Tc-99, Pu-242, Np-237); and
- Fissile material content (e.g., U-233, U-235, Pu-239, Pu-241)

These characteristics in MSR fuel salt waste would vary depending on the reactor design, fuel formulation, operational characteristics (burnup, fueling approach, and online processing), as well as any post-discharge fuel waste processing.

2.2 Potential Waste Forms for Fuel Salt Waste

The fuel salt waste form depends on the amount and type of processing performed after the waste is generated. Processing concepts and technologies for MSR fuel salt wastes, including treatment and immobilization, are being evaluated to produce waste forms acceptable for waste management. The various potential waste forms for storage of fuel salt waste have been previously described in a recent NRC-sponsored report by LaPlante et al. (2024). Accordingly, in this section, a brief description of potential waste forms for fuel salt waste is provided, and more detailed information that has largely been documented in the previous reports is included in Appendix A.

Some of the waste forms involve immobilization, where the liquid salt waste is converted into a stable solid form for long-term storage and is resistant to deliquescence and dissolution. The primary potential waste forms that have been identified as likely for fuel salt wastes are variations of glass waste forms [i.e., phosphate, tellurite, and lanthanide borosilicate (LABS) glasses], ceramic waste forms [i.e., titanate, sodalite, apatite, and silica aluminophosphate (SAP)], and metallic composite waste forms [i.e., ceramic-metal (cermet) composites and halide-metal (halmet) composites] (Riley et al., 2018, 2019; Riley, 2020; Arm et al., 2020; Carlson et al., 2021; EPRI, 2022; Matteo et al., 2023). Arm et al. (2020) tabulate in detail waste form options applicable to immobilization of MSR fuel salts, with delineation between fluoride and chloride salts.

One important consideration is the extent of finality for a waste form. Unprocessed and unstabilized MSR fuel salt waste can be considered a temporary waste form because it has characteristics that present challenges for interim storage and potentially for disposal. The long-term performance of the containment (e.g., can, canister, or container) is a key consideration. Stabilized waste forms generally appear to be intended as final forms that are not easily re-processed, though some stabilized waste forms may retain the option for future conversion to a more durable final form.

2.3 Potential Storage Methods for Fuel Salt Waste

MSR designs typically include one or more drain tanks that are intended to temporarily store the fuel salt when an operating reactor is in shutdown mode. The duration of temporary tank storage would likely depend on the design and operational approach of individual proposals, the availability of fuel waste processing capabilities, and the broader fuel cycle concept associated with the specific implementation of any particular MSR technology. Storage tanks have the benefit of limiting handling and provide the capability for reheating and transferring the fuel salt waste to other containers to facilitate continued material management, including storage, transportation, and further processing (either onsite or offsite). Storage tanks are less favorable for long-term storage because unstabilized fuel salt waste has limited durability, contains reactive (e.g., corrosive) halides, and for some systems there could be gas generation from radiolysis.

Fuel salt drain tanks used at the MSRE were described as thick walled and composed of Hastelloy N, a special alloy Oak Ridge National Laboratory (ORNL) created for the MSRE program to provide strength at high temperatures and corrosion resistance to the reactor’s eutectic fluoride fuel salt (Notz, 1985). The tanks were put into service when the reactor first started in 1965 and have continued to contain the fuel salt post-shutdown (1969) until the present. The only notable exception to containment is the documented migration of fluorine and uranium hexafluoride gases into the connected off-gas system (Peretz, 1996; McMillan, 2019).

Thermal energy storage at concentrating solar power plants provides additional potentially informative experience with the storage of molten salts. Though the solar salt has a different composition and its chemistry is not complicated by irradiation impacts (e.g., fission product ingrowth and decay), the potential for corrosion of storage vessels exists. Generation 2 concentrating solar power plants with a thermal energy storage system rely on two metal storage tanks—one cold tank at 290 °C [554 °F] and one hot tank at 565 °C [1,049 °F]—to house the solar salt inventory as both the heat transfer fluid and the thermal energy storage medium. Carbon steel is used for the cold tank, and a specialized stainless steel (SS347H) is used for the hot tank. The tank shells are in direct contact with the molten solar salt at these temperatures. Several types of failures of SS347H have been reported (Hong et al., 2023) and two instances of hot tank failures have been reported (Mehos et al., 2020), likely due to thermal stresses and low-cycle fatigue in the SS 347H floor. The low-cycle fatigue failures are likely due to the metallurgical vulnerability of SS 347H to stress relaxation cracking at the hot tank's temperature, where the alloy's microstructure is susceptible to elemental diffusion/segregation leading to a phase segregation and loss of strength along grain boundaries (Hong et al., 2023). While the post-discharge storage of MSR fuel salt wastes would likely be at lower temperatures (i.e., less than 290 °C [554 °F]), which would be expected to reduce the rate of corrosion, the solar experience reinforces the importance of selecting materials that will perform as intended under the expected operating conditions. Generation 3 concentrating solar power plant designs (presently undergoing research and development) use a molten chloride salt storage system at 720 °C [1,328] and employ a different tank design (Turchi, et al., 2021), where both tanks are internally insulated with refractory liners. Information and experience from the Generation 3 solar power plant effort, when it becomes available, may further inform the storage of MSR fuel salt wastes.

Conceptually, longer-term storage of the fuel salt waste is expected to be facilitated by eventual transfer to a container, canister, or can regardless of the level of processing. Placing waste materials in containers that are compatibly sized to work with an existing or new canister or cask system is one possible way MSR fuel salt waste might be managed.

Several processes have been proposed or investigated for MSR canister or cask systems. Idaho National Laboratory (INL) (working with partners) has described the concept of designing and manufacturing irradiated salt containers to receive irradiated fuel salt from the Molten Chloride Reactor Experiment (MCRE) project to facilitate long-term storage and protection of workers and the environment from radiological contamination (DOE, 2023). These irradiated salt containers are expected to be cylindrical containers with an outside diameter of 0.23 meters [9 inches] and a height of 1.0 meter [40 inches], would be capable of utilizing an inert cover gas over the irradiated fuel salt, and would be designed for compatibility with a standard DOE hazardous waste container.

Another packaging concept with general design information related to transportation of unstabilized MSR fuel salt waste was described by ORNL in the context of MSRE fuel salt waste disposition alternatives (Peretz, 1996). ORNL proposed filling an inner Hastelloy-N can with molten salt and allowing it to cool and solidify. The can would be placed inside a shielded container, and three shielded waste containers could then be placed inside a remote handled transuranic (RH-TRU) canister. The canning option was not pursued at ORNL because it was associated with a broader remedial action alternative that was not selected.

For potential commercial nuclear power projects, dry storage systems would be certified by NRC in accordance with the requirements in 10 CFR Part 72. Stabilized (or separated and stabilized) fuel wastes might also be stabilized in a can or canister that facilitates continued management as previously envisioned for borosilicate glass waste forms for high-level waste (NWTRB, 2017). Once MSR fuel salt waste has been placed in a container, conceptually the storage of filled containers could be similar to existing irradiated material storage approaches (e.g., dry cask storage system, vault system). If needed, container and/or waste form loading could be adjusted to satisfy design limits such as those related to decay heat, criticality control, and radiation safety.

3 POTENTIAL SAFETY AND TECHNICAL CONSIDERATIONS FOR INTERIM STORAGE OF MSR FUEL SALT WASTES

Potential safety and technical considerations were assessed for safe interim storage of salt wastes, while considering the material properties, potential waste forms, and possible storage methods described in Chapter 2. The assessment was focused on technical topics applicable to storage cask and facility safety reviews that are most affected by differences in material properties and waste forms relative to those typically encountered in corresponding LWR SNF safety reviews. Thermal, criticality, materials, confinement, and waste management topics were assessed. Safety topics such as shielding, structural, and accidents were not included because they apply more directly to storage cask systems rather than waste forms.

3.1 Thermal

A thermal evaluation addresses heat transfer and flow characteristics and decay heat removal systems of a dry storage system (DSS) (NRC, 2020). A thermal evaluation ensures that the storage container and fuel material temperatures of a DSS remain within the allowable limits for normal, off normal, and accident conditions.

LaPlante et al. (2024) assessed potential technical challenges and information needs associated with certification of storage containers for MSR fuel salt wastes. The authors noted that reactive gases may be a consideration for fuel salt waste, depending on reactor design and operation (e.g., fluorine and uranium hexafluoride for fluoride-based fuel salt wastes). Reactive gases that differ from gases in LWR spent fuel may involve different considerations including additional design or components (e.g., use of “getters”) and would have different properties and potentially different thermal characteristics and effects. Improved characterization of MSR fuel waste radionuclide inventories for various design and operation options would help to bound the range of possible inventories, including contributors to decay heat, and facilitate comparisons with LWR spent fuel and other wastes that can inform preparations for safety reviews. Further analysis of possible waste forms and their characteristics, including thermal characteristics, was also recommended to help reduce uncertainties about likely options that may be pursued in future applications.

Waste form processing that involves dehalogenation would address the process of radiolysis-induced gas generation in MSR fuel salt waste storage that was previously observed with the storage of MSRE fuel salt waste (Peretz, 1996).

If needed, decay heat (once known) could be attenuated by adjusting the package waste loading. One notable characteristic of cermet waste forms is they are more heat tolerant relative to ceramic waste forms due to the higher thermal conductivity of the metallic phase and can accept much higher fission product loadings than glasses or ceramics (Evarts et al., 2024). No evidence of thermal testing of waste forms for stress effects or thermal fracturing was identified in the literature reviewed for this report.

3.2 Criticality

A criticality evaluation is used to ensure SNF proposed to be placed into dry storage under 10 CFR Part 72 remains subcritical under normal, off normal, and accident conditions involving handling, packaging, transfer, and storage (NRC, 2020). The combination of fissile material and neutron sources presents the potential for criticality in a potential MSR fuel salt waste storage system or facility.

LaPlante et al. (2024) assessed potential technical challenges and information needs associated with certification of storage containers for MSR fuel salt wastes. The authors noted that variation in possible reactor designs and operational characteristics implies variation in fissile material inventories. Additionally, some materials in MSR fuel salt waste can contribute to elevated neutron activity. Notz (1985) highlighted elevated neutron activity in MSRE fuel salts from alpha neutron reactions due to the presence of Be-9, F-19, and Li-7. Another potential criticality hazard is associated with the radiolysis of fluoride-based fuels. This reaction creates fluorine gas and uranium hexafluoride, which, if not properly controlled, can migrate, deposit in ventilation systems, and pose a potential criticality safety concern (Peretz, 1996). Unprocessed salts are hygroscopic (absorb water from air), and this theoretically could potentially increase reactivity and the potential for criticality. Dehalogenation might reduce the hygroscopic nature of the material and would also address the fluorine gas concern. Arm et al. (2020) recommended consideration of developing waste forms with integral criticality control, such as (i) including a combination of managing of the mass and distribution of fissile material within the encapsulating and stabilizing matrix, (ii) placement and longevity of neutron absorbers, and (iii) strategies for the waste form performing a moderator exclusion or moderator displacement function.

Separations processing for uranium alone or uranium and plutonium would reduce the reactivity of the fuel waste and the potential for criticality relative to unseparated fuel salt waste. Waste forms with lower waste loading would have less concentrated fissile material and lower potential for criticality relative to waste forms with higher waste loading. The potential for phase separation upon cooling remains a technical consideration that exists for unprocessed and unseparated fuel salt waste and has been evaluated for other waste forms. The potential for reconfiguration and concentration of fissile material can impact criticality safety.

3.3 Materials

A materials evaluation addresses materials performance of the structures, systems, and components (SSCs) used for dry storage. The materials review ensures that the materials performance of storage system SSCs (i) is adequate under all credible loads and environments for normal, off-normal, and accident conditions, and (ii) will not pose operational problems with respect to its removal from storage under normal and off-normal conditions (NRC, 2020). Specific materials topics related to waste forms include the potential for material phase separation; reactivity and corrosion; radiation effects on materials; and adverse environmental factors such as water ingress.

3.3.1 Phase Separation

The specific topic of phase separation, when applied to glass, refers to the process where the homogeneous glass melt separates into two or more distinct phases during cooling. This phenomenon is undesirable because it can negatively impact the chemical durability and overall performance of the glass. For borosilicate glass-based waste forms used to immobilize high-level waste, the phase stability of the glasses is of concern during melting and long-term storage (Jain and Pan, 2000). Phase instability in borosilicate glasses can be induced by either liquid-liquid phase separation or crystallization on cooling from the melt. Both processes can result in the development of an inhomogeneous microstructure that may affect the reliability of the waste glass process and product performance.

For phosphate glasses developed for the immobilization of radioactive waste, Chong and Riley (2024) noted that concerns with phosphate glass waste form production include (i) phase separation during vitrification, (ii) crystallization during slow cooling after melt casting into storage canisters, and (iii) reduced waste form chemical durability. Chemical durability studies of the phosphate glass waste forms have shown that varying the glass composition and structure of phosphate glass can have significant impact on minimizing the phase separation and enhancing the chemical durability. Phase separation was also observed in the silica aluminophosphate (SAP) ceramic waste form, where Li/P-rich droplets were seen in the glassy aluminoborosilicate network (Riley, 2020). Lee et al. (2019) investigated the structure of the upgraded-SAP (U-SAP) waste form and showed that U-SAP waste form exhibited high durability because the droplets were encapsulated by durable silicate glass. However, phase separation in potential waste forms for immobilization of MSR fuel salt wastes, that result in the development of layers or a stratified solid, have not been reported.

3.3.2 Reactivity and Corrosion

Ensuring compatibility between MSR fuel salt waste forms and the storage container is critical for the safety of interim storage of MSR fuel salt wastes. EPRI (2022) noted that MSR fuel salt wastes pose several challenges for meeting the confinement requirements, including possible release of potentially corrosive gases (e.g., uranium hexafluoride and fluorine) and incompatibility of the wastes with the materials of fabrication of the storage container. The presence of corrosive fluoride or chloride salts poses long-term material compatibility challenges for storage canisters. Therefore, further investigation is warranted into the compatibility of materials between the storage container and the MSR fuel salt wastes due to the chemical characteristics of the wastes.

A materials evaluation includes the review of (i) degradation and corrosion from environmental conditions and (ii) other chemical degradation processes caused by reactions among the contents or between the contents and the storage container components (NRC, 2020). LaPlante et al. (2024) assessed potential technical challenges and information needs associated with certification of storage containers for MSR fuel salt wastes. The authors identified the need for thorough evaluation of the chemical interactions of air and water with MSR fuel salt waste or associated storage container components that are relied upon for providing structural support and confinement safety functions.

3.3.3 Radiation Effects

For storage of MSR fuel salt wastes, one of the primary concerns involves the accumulation of radiolytically produced gases (e.g., uranium hexafluoride and fluorine) from the salts during storage conditions [EPRI (2022)]. The storage containers could be over-pressurized if these gases are not vented over storage periods. The potential for over-pressurization of storage containers because of the release of radiolytic gases from molten salts may need to be addressed. Another concern is associated with radiolytically produced fluorine gas, which can be highly corrosive. The presence of fluorine gases may increase the rate of corrosion on storage containers or other components with the potential to compromise the confinement boundary with sustained exposure. Corrosion of the storage container internal surfaces due to the presence of fluorine gases may need further evaluation.

Studies of radiation amorphization that could lead to cracking in potential waste forms for immobilization of MSR fuel salt wastes were not found in technical literature searches conducted for this report.

3.3.4 Environmental Factors Including Water Ingress

No discussion of environmental factors (e.g., water ingress) that could affect the performance of potential MSR fuel salt waste forms was found in technical literature searches conducted for this report.

3.4 Confinement

A confinement review evaluates whether (i) the confinement features of the DSS and dry storage facility (DSF) are designed to ensure that radiological releases to the environment would be within the limits established by the regulations and (ii) the waste material will be sufficiently protected against degradation (NRC, 2020).

LaPlante et al. (2024) assessed potential technical challenges and information needs associated with certification of storage containers for MSR fuel salt wastes, noting that differences in the properties and radionuclide inventory of the MSR fuel waste material and its response to thermal conditions (e.g., melting and crystallization/solidification) may require different considerations, data, or information compared to estimated release fractions in typical LWR SNF evaluations.

Melting applies to unprocessed and unstabilized MSR fuel salt waste and would be addressed by stabilized waste forms that have much higher melting points. Forms with increasing durability would appear to lower the release fraction and dispersibility of stored contents under credible accident conditions; however, specific studies on how the potential MSR waste forms address these factors were not identified.

3.5 Waste Management Considerations

The selection of a waste form can impact future waste management considerations. One important consideration is whether the waste form is temporary or final. Unprocessed and unstabilized MSR fuel salt waste can be considered a temporary or interim waste form because of inherent characteristics (hygroscopicity, solubility, durability, reactivity/corrosivity/gas generation) that may not meet potential future waste acceptance criteria for final disposal. The unprocessed and unstabilized form can be remelted, repackaged, and processed at some future time after a period of storage on the order of several decades, possibly longer depending on the long-term performance of the containment (e.g., can, canister, container). Stabilized waste forms appear to be intended as final forms that are not easily re-processed; however, as described above, INL is considering mixing electrorefiner salt wastes with zeolites as an intermediate waste form to mitigate the potential for salt deliquescence and consequently corrosion while retaining the option for future conversion to a more durable final form.

4 SUMMARY

An assessment of potential safety and technical considerations for interim storage of liquid-fueled MSR fuel salt wastes was conducted, considering available technical information on MSR fuel salt waste characteristics, different levels of processing, potential waste forms and storage methods, and potential storage technologies that could be used by a potential licensee to manage these wastes. Most MSR designs include fuels composed of either fluoride salts or chloride salts. Because the degree of processing is central to understanding waste characteristics and compatible waste forms, this assessment considers waste form options that have been demonstrated or studied for MSR applications in the context of three potential levels of post-discharge MSR fuel waste processing, including (i) no processing (unprocessed and unstabilized waste), (ii) stabilizing unseparated waste, and (iii) stabilizing separated waste.

The unprocessed and unstabilized fuel salt waste form can potentially be stored for decades after discharge and provides some flexibility for transfer from one containment to another or for further processing at a future time. Material characteristics of unprocessed fuel salt waste suggest some potential long-term challenges in storing discharged MSR fluoride-based fuel salt waste at the Oak Ridge Molten Salt Reactor Experiment (Peretz, 1996). The characteristics of the unprocessed and unstabilized form suggest that practical deployments would proceed with further processing options (e.g., stabilization, separations) to achieve a more durable and manageable long-term waste form.

The specific materials remaining in waste streams after processing would depend on the processing that is applied, including any extractions performed for reuse of materials that would remove the materials from specific waste streams. Important waste characteristics highlighted for waste form formulation and associated processing include (i) decay heat; (ii) concentrations and specific ratios of halogen, alkali, alkaline earths, rare earths, and transition metals; (iii) concentrations of U, Pu, other actinides, transition metals, and long-lived isotopes; and (iv) fissile material content.

The primary waste forms identified in the technical literature that could potentially be used to immobilize MSR fuel wastes include variations of glass waste forms [i.e., phosphate, tellurite, and lanthanide borosilicate (LABS) glasses], ceramic waste forms [i.e., titanate, sodalite, apatite, and silica aluminophosphate (SAP)], and metallic composite waste forms [i.e., ceramic-metal (cermet) composites and halide-metal (halmet) composites].

Waste form options are limited for stabilized and unseparated MSR fuel salt waste in the presence of halides. The glass-bonded sodalite ceramic waste form (GBS-CWF) is an option being explored for chloride-based fuel waste and the glass-bonded apatite form is being explored for fluoride-based fuel waste stabilization. Waste loading was found to be lower for these glass-bonded ceramic waste forms (and waste form volume was higher) relative to other waste forms, including those that are applicable to dehalogenated waste, thereby providing some incentive for further processing.

Potential waste forms that are being explored to address separated MSR fuel waste streams focus on addressing electrochemical processing waste streams that involve chloride-specific wastes. Of these, the ultrastable H-Y zeolite (USHYZ) process that incorporates dehalogenation with ceramic waste form production addresses waste management challenges associated with retaining halides in the waste form. The iron phosphate (Fe-P-O) glass waste form also incorporates dehalogenation and allows relatively high waste loading.

The variety of potential MSR fuel waste forms being evaluated are a result of (i) the variety of options for reactor design and associated fuel composition and (ii) the degree of processing, which is affected by fuel cycle considerations and the related waste streams and material compositions. The number of options under consideration for how MSR technology might be implemented presents a challenge for narrowing the focus to specific waste forms that might be utilized in future projects.

The assessment of potential safety and technical considerations for the safe interim storage of MSR fuel salt wastes focused on technical topics applicable to storage cask and facility safety reviews that are most affected by differences in material properties and waste forms relative to typical light-water reactor spent nuclear fuel-focused considerations. These topics include thermal, criticality, materials, confinement, and waste management.

- Technical considerations associated with a *thermal evaluation* included improved estimates of MSR fuel waste radionuclide inventories for various design and operation options that would help to bound the range of possible inventories, including contributors to decay heat, and facilitate comparisons with LWR spent fuel and other wastes that can inform preparations for safety reviews. No evidence of thermal testing of waste forms for stress effects or thermal fracturing was identified in the literature reviewed for this report.

- Technical considerations associated with a *criticality evaluation* included processing with separations for either uranium alone or jointly for uranium and plutonium, which would reduce (i) the reactivity of the fuel waste and (ii) the criticality potential relative to unseparated fuel salt waste. The potential for reconfiguration and concentration of fissile material can affect safety concerns. Design features that result in moderator exclusion could also be considered.

- Technical considerations associated with a *materials evaluation* focused on processes related to waste forms, including the potential for phase separation, reactivity, and corrosion; radiation effects on materials; and environmental factors such as water ingress. Phase separation has been noted as a potential concern in developing MSR compatible waste forms (Section 3.3.1); however, the development of layers or a stratified solid have not been reported in the studies examined for this report. Regarding reactivity and corrosion, the presence of corrosive fluoride or chloride salts poses long-term material compatibility challenges for storage canisters that warrant further investigation. The accumulation of radiolytically produced gases (e.g., uranium hexafluoride and fluorine) generated from fluoride-based salts during storage conditions was also noted as a potential concern regarding increasing internal pressure and from the potential for enhancing corrosion. No studies related to the potential effects of water ingress were identified for this report.

- Technical considerations associated with a *confinement evaluation* focused on the potential for phase changes (e.g., melting) of unprocessed and unstabilized MSR fuel salt waste under potential accident conditions. This could be addressed by employing stabilized waste forms that have much higher melting points. Information on how waste forms affect the release fractions and dispersibility of stored contents under credible accident conditions is needed but was not identified.

Assessment of waste management options focused on how the selection of a waste form can impact future waste management considerations. Finality of the waste form is one important consideration. Unprocessed and unstabilized MSR fuel salt waste can be considered a temporary waste form because it has characteristics (hygroscopicity, solubility, durability, reactivity/corrosivity/gas generation) that present potential challenges for interim storage and ultimate disposal. The unprocessed and unstabilized form can be remelted, repackaged, and processed at some future time after a period of storage on the order of several decades, possibly longer depending on the long-term performance of the containment (e.g., can, canister, container). Stabilized waste forms appear to be intended as final forms that are not easily re-processed; however, INL is considering mixing electrorefiner salt wastes with zeolites as an intermediate waste form to mitigate the potential for salt deliquescence and consequently corrosion while retaining the option for future conversion to a more durable final form.

In summary, this report aids the NRC staff's understanding of the potential challenges associated with the storage of molten salt reactor spent fuel salts and associated waste forms,. It provides a better understanding of MSR fuel waste radionuclide inventories for various design and operation options, as well as information on materials properties such as radiation effects on potential waste forms.

5 REFERENCES

Adams, G., P. LaPlante, and Yi-Ming Pan. “Assessment of the Current State of Knowledge on Storage and Transportation of Molten Salt Reactor Waste—Final Report.” NRC Adams Accession No. ML23188A168. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2023.

Arm, S.T., D.E. Holcomb, R. Howard, and B. Riley. “Status of Fast Spectrum Molten Salt Reactor Waste Management Practice.” December 2020. <https://www.osti.gov/servlets/purl/1761520/> (Accessed April 3, 2023).

Carlson, K., L. Gardner, J. Moon, B. Riley, J. Amoroso, and D. Chidambaram. “Molten Salt Reactors and Electrochemical Reprocessing: Synthesis and Chemical Durability of Potential Waste Forms for Metal and Salt Waste Streams.” *International Materials Reviews.* Vol. 66. pp. 339–363. 2021.

Chong, S. and B. Riley. “Compositional Effects on the Chemical Durabilities of Aluminophosphate Glasses: A Review.” *Journal of Non-Crystalline Solids*. X. Vol. 21. pp. 100–205. 2024.

DOE. “Final Environmental Assessment for the Molten Chloride Reactor Experiment (MCRE) Project.” DOE/ID-2209, Revision 0, INL/RPT-22-68976. Idaho Falls, Idaho: U.S. Department of Energy, DOE Idaho Operations Office. 2023. <https://www.energy.gov/sites/default/files/2023-08/final-ea-2209-molten-chloride-reactor-2023-08.pdf> (Accessed 24 March 2024).

DOE. “Converting UNF Radioisotope Inventories into Energy Workshop.” DOE Advanced Research Projects Agency-Energy. 2021a.

DOE. “Optimizing Nuclear Waste and Advanced Reactor Disposal Systems.” DOE Advanced Research Projects Agency-Energy. 2021b

EPRI. “Evaluation of Existing Storage Technology for Irradiated Advanced Reactor Fuel: Options for Onsite Containerized Storage of Irradiated Fuel and High-Level Waste Arising from Advanced Reactors.” Palo Alto, California: Electric Power Research Institute. 2022.

Ebert, W. and J. Fortner. “Analyses of Iron Phosphate Glasses for Dehalogenated Salt Waste.” ANL/CFCT-19/5. Lemont, Illinois: Argonne National Laboratory. 2019. <https://publications.anl.gov/anlpubs/2019/07/153602.pdf> (Accessed 2 May 2025).

Ebert, W.L., C.T. Snyder, B.J. Riley, and S.M. Frank. “Designing Advanced Ceramic Waste Forms for Electrochemical Processing Salt Waste.” Argonne National Laboratory, Pacific Northwest National Laboratory, Idaho National Laboratory. 2016. <https://publications.anl.gov/anlpubs/2016/04/126582.pdf> (Accessed 26 March 2024).

Evarts J.S., S. Chong, J.M. Oshiro, B.J. Riley, R.M. Asmussen, and J.S. McCloy. “Ceramic-metal (cermet) Composites: A Review of Key Properties and Synthesis Methods Focused on Nuclear Waste Immobilization.” *Industrial and Engineering Chemistry Research*. Vol. 63, No. 14. pp. 6,003–6,023. 2024.

Gardner, D., M. Wasnik, B. Riley, S. Chong, M. Simpson, and K Carlson. “Synthesis and Characterization of Sintered H–Y Zeolite-Derived Waste Forms for Dehalogenated Electrorefiner Salt.” *Ceramics International*. Vol. 46. pp. 17,707–17,716. 2020.

Harward, A., L. Gardner, C.M. Decker Oldham, K. Carlson, T.S. Yoo, G. Fredrickson, M. Patterson, and M. Simpson. “Water Sorption/Desorption Characteristics of Eutectic LiCl-KCl Salt-Occluded Zeolites.” *Journal of Nuclear Fuel Cycle Waste Technology*. Vol. 20. pp.259–268. 2022.

He, L.Y., Y. Cui, L. Chen, S.P. Xia, L.Y. Hu, Y. Zou, and R. Yan. “Effect of Reprocessing on Neutrons of a Molten Chloride Salt Fast Reactor.” *Nuclear Science and Techniques*. Vol. 34, Article 46. 2023. <https://doi.org/10.1007/s41365-023-01186-3> (Accessed 12 February 2025).

Hong, Y., T. Pickle, J. Vidal, C. Augustine, and Z. Yu. “Impact of Plate Thickness and Joint Geometry on Residual Stresses in 347H Stainless Steel Welds.” *Welding Journal*. Vol. 102, 279–292. 2023. <https://doi.org/10.29391/2023.102.021> (Accessed 29 April 2025).

Jain, V. and Y. Pan. ”Glass Melt Chemistry and Product Qualification.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. ADAMS Accession No. ML033640523. 2000.

LaPlante, P., B. Dasgupta, and Y. Pan. “Technical Assessment of Molten Salt Reactor Fuel Salt Waste Processing—Draft Final Report.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2025.

LaPlante, P, B. Dasgupta, Yi-Ming Pan, and G. Adams. “Storage and Transportation of Molten Salt Reactor Wastes: Identification of Technical Information Needs and Implications for Safety Review Guidance—Final Report.” NRC Adams Accession No. ML24256A003. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2024.

Lee, Ki Rak, B.J. Riley, H. Park, J. Choi, S.Y Han, J. Hur, J.A. Peterson, Z. Zhu, D.K. Schreiber, K. Kruska, and M.J. Olszta. “Investigation of Physical and Chemical Properties for Upgraded SAP (SiO_2-Al_2O_3-P_2O_5) Waste Form to Immobilize Radioactive Waste Salt.” *Journal of Nuclear Materials*. Vol. 515. 382p. 2019.

Matteo, E., L. Price, R. Pulido, P. Weck, A. Taconi, P. Mariner, T. Hadgu, H. Park, J. Greathouse, D. Sassani, and H. Alsaed. “Advanced Reactors Spent Fuel and Waste Streams Disposition Strategies.” SAND2023-08602R. Albuquerque, New Mexico: Sandia National Laboratory. 2023.

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

McMillan, B. “Molten Salt Reactor Experiment Project Initiatives.” Oak Ridge, Tennessee: Oak Ridge National Laboratory. 2019. <https://www.energy.gov/sites/prod/files/2019/11/f68/ORSSAB%20MSRE%20Presentation%20-%20Nov%202019.pdf> (Accessed 29 April 2025).

Mehos, M, H. Price, R. Cable, D. Kearney, B. Kelly, G. Kolb, and F. Morse. “Concentrating Solar Power Best Practices Study.” Golden, Colorado: National Renewable Energy Lab. 2020. <https://www.nrel.gov/docs/fy20osti/75763.pdf> (Accessed 29 April 2025).

NAS. “Merits and Viability of Different Nuclear Fuel Cycles and Technology Options and the Waste Aspects of Advanced Nuclear Reactors.” Washington, DC: The National Academies Press. 2022. <https://doi.org/10.17226/26500> (Accessed 10 May 2023).

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

NRC. NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities–Final Report”. ADAMS Accession No. ML20121A190. Washington DC: U.S. Nuclear Regulatory Commission. 2020.

NWTRB. “Vitrified High-Level Radioactive Waste.” Revision 1. Arlington, Virginia: U.S. Nuclear Waste Technical Review Board. 2017. (Accessed 26 March 2025).

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

Priebe, S. and K. Bateman. “The Ceramic Waste Form Process at Idaho National Laboratory.” *Nuclear Technology*. Vol. 162. pp. 199–207. 2008.

Riley, B., J. Vienna, and W. Ebert. “Road Map for Developing Iron Phosphate Waste Forms for Salt Wastes.” PNNL-30998. Richland, Washington: Pacific Northwest National Laboratory. 2021.

Riley, B. “Electrochemical Salt Wasteform Development: A Review of Salt Treatment and Immobilization Options.” *Industrial & Engineering Chemistry Research*. Vol. 59. pp. 9,760–9,774. 2020.

Riley, B., J. McFarlane, G. DelCul, J. Vienna, C. Contescu, and C. Forsberg. “Molten Salt Reactor Waste and Effluent Management Strategies: A Review.” *Nuclear Engineering and Design.* Vol. 345. pp. 94–109. 2019.

Riley, B., J. McFarlane, G. DelCul, J. Vienna, C. Contescu, L. Hay, A. Savino, and H. Adkins. “Identification of Potential Waste Processing and Waste Form Options for Molten Salt Reactors.” NTRD-MSR-2018-000379, PNNL-27723. Richland, Washington: Pacific Northwest National Laboratory. 2018.

Riley, B., J. Kroll, J. Peterson, D. Pierce, W. Ebert, B. Williams, M. Snyder, S. Frank, J. George, and K. Kruska. “Assessment of Lead Tellurite Glass for Immobilizing Electrochemical Salt Wastes from Used Nuclear Fuel Reprocessing. *Journal of Nuclear Materials*, Vol. 495. pp. 405–420. 2017.

Turchi, C.S, S. Gage, J. Martinek, S. Jape, K. Armijo, J. Coventry, J. Pye, C.-A. Asselineau, F. Venn, W. Logie, A. Fontalvo, S. Wang, R. McNaughton, D. Potter, T. Steinberg, and G. Will. “CSP Gen3: Liquid-Phase Pathway to SunShot.” Golden, Colorado: National Renewable Energy Lab. (NREL); Albuquerque, New Mexico: Sandia National Laboratory; Canberra, Australia: Australian National University; Newcastle, Australia: Commonwealth Scientific and Industrial Research Organization (CSIRO); Brisbane, Australia: Queensland University of Technology. 2021. Available at <https://docs.nrel.gov/docs/fy21osti/79323.pdf> (Accessed 17 August 2025).

Wasnik, M., A. Grant, K. Carlson, and M. Simpson. “Dechlorination of Molten Chloride Waste Salt from Electrorefining via Ion-Exchange Using Pelletized Ultra-Stable H-Y Zeolite in a Fluidized Particle Reactor.” *Journal of Radioanalytical and Nuclear Chemistry*. Vol. 320. pp. 309–322. 2019.

APPENDIX A

POTENTIAL WASTE FORMS FOR MOLTEN SALT REACTOR FUEL SALT WASTE

APPENDIX A—POTENTIAL WASTE FORMS FOR MOLTEN SALT REACTOR FUEL SALT WASTE

This appendix includes an expanded section on the Potential Waste Forms for Fuel Salt Waste. This information is based upon public information and builds upon other NRC-sponsored reports, which are cited within.

A.1 Potential Waste Forms for Fuel Salt Waste

The fuel salt waste form depends on the amount and type of processing performed after the waste is generated. Processing changes the waste form. Processing concepts and technologies for MSR fuel salt wastes, including treatment and immobilization, are being evaluated to produce waste forms acceptable for waste management. An important processing objective is to address all the various species present in the salts, including fission and activation products (halides, alkalis, alkaline earths, rare earths) and actinides. The specific materials remaining in waste streams after processing would depend on the processing that is applied, including any extractions performed for reuse of materials that would remove those materials from specific waste streams.

A variety of potential waste forms intended to immobilize MSR fuel wastes have been described in technical literature. Immobilization in this context means that the liquid salt waste is converted into a stable solid form for long-term storage, resistant to deliquescence and dissolution. The primary waste forms identified are variations of glass waste forms [i.e., phosphate, tellurite, and lanthanide borosilicate (LABS) glasses], ceramic waste forms [i.e., titanate, sodalite, apatite, and silica aluminophosphate (SAP)], and metallic composite waste forms [i.e., ceramic-metal (cermet) composites and halide-metal (halmet) composites] (Riley et al., 2018, 2019; Riley, 2020; Arm et al., 2020; Carlson et al., 2021; EPRI, 2022; Matteo et al., 2023). Arm et al. (2020) tabulate in detail waste form options applicable to immobilization of MSR fuel salts, with delineation between fluoride and chloride salts. Various waste forms have been experimentally demonstrated for MSR applications. These include the glass bonded sodalite (GBS) ceramic waste form (GBS-CWF), which is compatible with chloride salts and does not require dehalogenation. Iron phosphate (Fe-P-O) glass and ultrastable H-Y zeolite (USHYZ) following dehalogenation of the chloride salts have also been demonstrated for MSR applications. The Fe-P-O glass and USHYZ waste forms have not been demonstrated for fluoride salt wastes.

One of the primary issues with production of waste forms with acceptable chemical durability is high halide content (mainly chlorides or fluorides) in the salt streams (Arm et al., 2020; Riley, 2020). Halides do not integrate well into glass and ceramic waste forms due to limited solubility in the waste forms, resulting in a low waste loading of halide-rich salts. Dehalogenation processes (removal of halides from the MSR fuel salt wastes) have been considered crucial for removing the halide solubility limitation (Riley, 2020). Additionally, radiolysis of solidified halide salts containing fission products can lead to the production of gaseous byproducts (e.g., F_2). Hence, dehalogenation of MSR fuel salt wastes is deemed necessary to increase waste loading, enhance waste form stability, and ensure compatibility with waste forms and storage containers.

Waste form properties that may be relevant to the safety reviews for licensing and certification of storage systems or facilities include reactivity, durability, resistance to thermal or radiological effects, and compatibility with future processing/re-canning (if needed). While little information is available on waste form properties relevant to the safety reviews for interim storage of MSR fuel salt wastes, Riley (2020) compared the physical properties of several potential waste forms (i.e., GBS, lead tellurite (Pb-Te-O) glass, Fe-P-O glass, and USHYZ). Table 3 of Riley (2020) includes data on density, porosity, salt loading, salt cation loading, and waste form volume for these waste forms. The waste compositions are also provided in Table S1 (Supporting Information) of Riley (2020). A salt cation loading comparison based on volume calculations for full-salt waste forms shows that the phosphate process results in the highest waste loading with the lowest waste form volumes (>5X decrease over the 8 mass% GBS-CWF).

The U.S. Department of Energy's (DOE) Advanced Research Projects Agency-Energy (ARPA-E) Program has initiated programs related to advanced reactors, including Optimizing Nuclear Waste and Advanced Reactor Disposal Systems (ONWARDS) and Converting UNF Radioisotopes into Energy (CURIE) (DOE 2021a; 2021b). The ONWARDS and CURIE programs partly aim to develop novel waste forms (i.e., glass-bonded perovskite ceramic waste forms) for advanced reactor waste streams and reduce waste volumes from advanced reactors.

Because the degree of processing is central to understanding waste characteristics and compatible waste forms, the following subsections describe waste form options that have been demonstrated or studied for MSR applications in the context of three potential levels of post-discharge MSR fuel waste processing: (i) no processing (unprocessed and unstabilized), (ii) stabilizing unseparated waste, and (iii) stabilizing separated waste.

A.1.1 Waste Forms for Unprocessed Fuel Salt Waste

Unprocessed fuel salt waste (e.g., the fuel that is discharged from a reactor without further separation or stabilization processing) can be expected to have the chemical characteristics of the fuel salt with most if not all of the additional aforementioned actinide and fission, activation, and decay products (some operational separations might be applied to maintain optimal fission in the reactor). Unprocessed fuel salt waste is solid at lower temperatures (below the solidus temperature), but additional processing can improve its stability.

One approach to managing unprocessed fuel salt waste would be to transfer the molten salt to containment (e.g., container, canister, or can) and allow it to cool and solidify. The container, canister, or can, would provide additional structural support, containment, shielding and possibly other safety functions to allow for continued management of the material. Section 2.3 of this report provides additional information about possible storage methods for this material.

One benefit to storing unstabilized MSR fuel salt waste is that the fuel salt can be remelted and transferred to a different container or processed at a later time. Material characteristics of unprocessed fuel salt waste (hygroscopic, soluble in water, low durability, reactive, corrosive, gas generating) suggest some potential storage challenges. Historical challenges in storing unprocessed discharged MSR fluoride-based fuel salt waste at MSRE (Peretz, 1996) included gas generation and the mobilization of fissile material, corrosion and operability of storage and transfer equipment, post-storage transfer, and site remediation. A previous CNWRA report (Adams et al., 2023) described the MSRE storage experience in greater detail based on available documentation; that discussion is not repeated here. The material characteristics of the unprocessed and unstabilized forms suggest that practical deployments would proceed with further processing options (e.g., stabilization, separations) to achieve a more durable and manageable long-term waste form.

A.1.2 Waste Forms for Stabilizing Unseparated MSR Fuel Salt Waste

MSR fuel waste discharged from a reactor can be further stabilized with additional processing to create a more durable waste form. Stability in this context refers generally to the waste form maintaining its designed chemical and physical characteristics sufficiently to allow continued safe management of the waste (including storage). Because of the high halogen content expected for MSR fuel salts and low halogen solubility limits in sodium aluminoborosilicate glasses (Riley, 2020; Lee et al., 2019), the conventional borosilicate approach is not suitable for vitrifying these wastes. The glass-bonded sodalite ceramic waste form (GBS-CWF) is one of the most well-studied waste form options for immobilizing chloride-based MSR fuel salt waste and has also been demonstrated as a waste form for salt wastes from electrochemical reprocessing of Experimental Breeder Reactor II fuel (Riley et al., 2019).

A.1.2.1 Ceramic Waste Form

GBS-CWF is a stabilized form that is a multi-phase material where alkalis and halides from the salts are immobilized in sodalite, while the remaining fission products (such as alkaline earths and rare earths and actinides) are incorporated into the glass phase as oxides (Priebe and Bateman, 2008; Riley et al., 2019; Riley, 2020; Ebert et al., 2016). This waste form is fabricated by grinding the salt waste and zeolite 4A to a fine particle size (45-250 μm), mixing the ground salt and zeolite, and heating to 500 °C [932 °F] to occlude the salt into the structure of the zeolite. A borosilicate glass binder is added at a loading of ~26 mass %. The mixture is then heated to 925 °C [1,697 °F] to convert the mixture to the final GBS-CWF waste form. The waste forms have been made at full scale (i.e., a 400 kg cylindrical monolith that is ~0.679 m diameter × 3.12 m tall), and production of radiological waste forms has also been experimentally demonstrated in a hot cell on a smaller scale (Carlson et al., 2021). The sodalite structure can readily incorporate alkalis and halides (Riley et al., 2018). However, glass-bonded apatite is considered a better option than glass-bonded sodalite for fluoride-based MSR fuel salt wastes because the apatite structure can incorporate alkaline earths in addition to alkalis and halides (Riley et al., 2018).

Priebe and Bateman (2008) measured chemical durability of the GBS-CWF using the standard product consistency test, according to American Society for Testing and Materials (ASTM) C1308 Test Method A. The test results revealed that the release rates of all the matrix elements (i.e., silicon, aluminum, boron, and potassium) are at least an order of magnitude lower than a comparable environmental assessment glass that is used as the baseline for repository durability analyses. The performance of accelerated leach tests suggests that the GBS-CWF is a suitable option for immobilizing unprocessed chloride-based fuel wastes.

A.1.3 Waste Forms for Stabilizing Separated MSR Fuel Salt Waste

Available technologies are being explored for separating constituents in MSR fuel salt waste to address various objectives for reuse of materials and waste management. Post-discharge fuel processing would define the potential waste form and the characteristics of the fuel waste material(s) requiring interim storage. Separations that might be applied to a specific MSR fuel salt waste would depend on the processing objectives, which in turn would be influenced by the reactor design, the fuel cycle, and available disposal options. LaPlante et al. (2025) summarized processing objectives and technology concepts from available literature and formulated conceptual processing scenarios that ranged from limited to complex.

Dehalogenation would (i) address the potential for fluorine gas generation from radiolysis (for fluoride salt wastes) and (ii) result in waste compatible with more waste form options.

Electrochemical treatment (i.e., electrorefining and pyroprocessing) is one of many methods described by several authors (Arm et al., 2020; McFarlane et al., 2019; Riley et al., 2018; Peretz, 1996) as a means to conduct separations on molten fuel salt waste. Electrochemical processing has the potential to separate actinides from the electrolyte. The resulting electrolyte can be further processed by dehalogenation, producing reusable materials and waste streams that can each be aligned with compatible waste form materials.

Peretz (1996) broadly described the purpose of electrochemical treatment in the context of evaluating alternatives for processing Molten Salt Reactor Experiment (MSRE) waste as the extraction of actinides and fission products from the fuel/flush salt, concentrating the radioactive constituents in a small volume of high-level waste and leaving a large volume of salt residue containing trace amounts of radioactivity. More recent descriptions emphasize the capability to remove actinides from an electrolyte solution containing MSR fuel salt waste, namely removing uranium alone or uranium and plutonium for reuse as reactor fuel (Riley et al., 2017). The resulting electrolyte mixture contains mostly fission products. The electrolyte mixture can be (i) stabilized as waste at the end of its useful life or (ii) further processed by electrochemical processing and other methods to reuse the electrolyte while producing separate fission product waste streams to be stabilized into compatible waste forms.

MSR fuel waste processing and waste form options for different waste streams are reviewed by Riley et al. (2018, 2019) and Riley (2020). Riley (2020) summarizes salt processing and waste form options for immobilizing full (unpartitioned) electrorefiner waste salt or partitioned electrorefiner salt waste streams. A full-salt waste stream could be incorporated into GBS or a Pb-Te-O glass waste form with a significantly limited waste loading. Partition of the fuel wastes using dehalogenation to remove halides from the salt provides alternative waste form options including iron phosphate (Fe-P-O) glass, SAP, and glass-bonded ceramic waste forms. Other waste form options, such as LABS glass or zinc-in-titania (ZIT) glass-bonded ceramic waste form, can be used for immobilizing rare-earth fission products if they are separated from the fuel waste. More detailed descriptions of potential waste form options by material type are provided in the sections that follow.

A.1.3.1 Glass Waste Form

An iron phosphate glass (Fe-P-O glass) waste form has been developed to immobilize salt wastes from the electrochemical reprocessing of spent nuclear fuel (Carlson et al., 2021; Riley et al., 2021). The process for producing this waste form with electrochemical salt wastes involves the reaction of chloride salts with phosphate precursors at moderate temperatures less than 600 °C [1,112 °F] that drives off the halides from the salt as a volatile product and produces a Fe-P-O glass. The Fe-P-O glass is not chemically durable; iron oxide is added and vitrified at higher temperatures less than 1,200 °C [2,192 °F] to generate a glass waste form with acceptable chemical durability. This process results in dehalogenation of the waste salt, capture of the gaseous halide-bearing species, and immobilization of the residual salt components in the Fe-P-O glass waste form.

Chemical durability studies of the Fe-P-O glass waste form have demonstrated its performance as a matrix for radioactive salt cations (Ebert and Fortner, 2019; Carlson et al., 2021). Accelerated leach tests were conducted using a modified ASTM C1308 method with four sequential complete solution exchanges made at 24-h intervals. The test results revealed that

Fe-P-O glasses exhibit much lower dissolution rates compared to the GBS-CWF. The Fe-P-O glasses have been demonstrated as waste forms for immobilizing chloride-based salt wastes and may potentially be suitable for the treatment and immobilization of fluoride-based salt wastes.

A.1.3.2 Ceramic Waste Form

The ultrastable H-Y zeolite waste form is produced after a dechlorination process that employs the ion exchange properties of microporous zeolite frameworks developed for immobilizing salt wastes from electrochemical reprocessing (Wasnik et al., 2019; Gardner et al., 2020). Development of this two-step approach process is driven by the lack of waste forms available for salt immobilization at high salt loads. This is referred to as the USHYZ process. The first step in waste processing involves the removal of chlorine as hydrogen chloride gas via ion exchange reactions between USHYZ and halide salts. In the second step, the salt-cation-loaded zeolite is consolidated to produce a dense, monolithic assemblage of silicate-based phases for storage or final disposal. Glass-bonded zeolite (GBZ) or glass waste forms have been associated with this process (Riley et al., 2018). Wasnik et al. (2019) evaluated several processing variables to improve the degree of ion exchange and achieve dechlorination efficiencies of up to 95%.

Gardner et al. (2020) performed chemical durability experiments using a modified ASTM C1308 method in deionized water and different silica solutions. A comparison of release rates from the GBS-CWF and the USHYZ-derived dehalogenated salt waste forms (DSWF) showed that the cumulative releases for the DSWF are less than 33% of those measured for the GBS-CWF. The USHYZ process has been demonstrated for dechlorination of chloride salt wastes but has not yet been demonstrated for defluorination of MSR-relevant fluoride salt mixtures.

A comparison of the GBS, Fe-P-O glass, and USHYZ waste-form physical properties is provided by Riley (2020). It should be noted that little information is available on waste form properties relevant to the safety reviews for storage of MSR fuel-related wastes, such as resistance to thermal or radiological effects. Therefore, further investigation into waste form options and their properties for MSR fuel wastes is warranted.

A.1.3.3 Metallic Composite Waste Form

Evarts et al. (2024) describe ceramic-metal (cermet) composite waste forms as potentially playing a more limited but important role where other waste forms underperform, such as with residual halides, noble metals and high-heat producing radioisotope immobilization. A significant thermal conductivity of the composite was noted due to the high conductivity of the metallic phase. The increased thermal conductivity allows higher fission product loadings than glasses and ceramics.

A.1.3.4 Zeolite-4A Waste Form Intermediate

Processing to produce a more stable intermediate waste form could be applied to MSR fuel waste streams to facilitate storage and management while retaining the option for future conversion to a more durable final form. Electrorefiner waste salt, for example, is hygroscopic (it readily absorbs water) and can deliquesce into a corrosive saline solution. Therefore, it is desirable to stabilize the LiCl-KCl eutectic when that is the electrorefiner waste salt. Developing a glass-bonded sodalite (GBS), which is a water-resistant chemical waste form, results in a significant increase in volume. The effective salt density in GBS is only 0.15 gcm^{-3} compared to liquid LiCl-KCl, which is 1.7 gcm^{-3}. Therefore, researchers at INL are looking into occlusion into zeolite-4A as an option for limited term storage and to mitigate the chance of water sorption. Harward et al. (2022) demonstrated that salt-occluded zeolite (SOZ) with 20 weight percent LiCl-KCl has a lower capacity for absorbing moisture from the atmosphere than salt alone and protects the salt long-term against deliquescence and container corrosion. The purpose of performing this process would be to render the salt safe from deliquescence and prevent corrosion of the storage canisters.

A.1.4 Assessment of Available Waste Form Options

Overall, considering the information presented in the preceding sections, the available research indicates a variety of potential waste forms are being explored to address a variety of MSR implementation options and associated potential waste streams. The unprocessed and unstabilized salt form can be stored potentially for decades after discharge and provides some flexibility for allowing transfer from one containment to another or further processing at a future time. As described in Section A.1.1, material characteristics of unprocessed fuel salt waste suggest some potential technical considerations, such as the potential for gaseous radiolysis products. Historical challenges in storing unprocessed discharged MSR fluoride-based fuel salt waste at MSRE have been documented (Peretz, 1996). The characteristics of the unprocessed and unstabilized forms suggest that practical deployments would proceed with further processing options (e.g., stabilization, separations) to achieve a more durable and manageable long-term waste form.

The waste form options for stabilized and unseparated MSR fuel salt waste are limited by the presence of halides; however, GBS-CWF is an option being explored for chloride-based fuel waste and the glass-bonded apatite waste form is being explored for fluoride-based fuel waste stabilization. Waste loading was found to be lower for this glass bonded ceramic waste form resulting in a higher net volume relative to other waste forms, including those that are applicable to dehalogenated waste, thereby providing some incentive for further processing. The Pb-Te-O glass waste form is also applicable to unseparated fuel salt waste but was noted as costly due to its use of tellurium dioxide (TeO_2).

Many more potential waste forms are being explored that can address separated MSR fuel waste streams. Considerable focus of current waste form research is on addressing electrochemical processing waste streams that involve chloride-specific wastes. Of these, the USHYZ process, that incorporates dehalogenation with ceramic waste form production, addresses waste management challenges associated with retaining halides in the waste form. The Fe-P-O glass waste form also incorporates dehalogenation and favors high waste loading. A potential role for cermets includes electrochemical waste streams with residual halides, noble metals, and high heat producing radioisotopes, and cermets display capabilities for increased fission product waste loadings relative to ceramic and glass forms. Evident from the preceding discussion is that a variety of potential MSR fuel waste forms that are being evaluated are associated with a variety of options for reactor design and associated fuel composition, degree of processing (which in turn is affected by fuel cycle considerations), and the related waste streams and material compositions.

The number of options under consideration for how MSR technology might be implemented presents a challenge for narrowing the focus on specific waste forms that might be utilized in future projects.

A.2 References

Adams, G., P. LaPlante, and Yi-Ming Pan. "Assessment of the Current State of Knowledge on Storage and Transportation of Molten Salt Reactor Waste—Final Report." NRC Adams Accession No. ML23188A168. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2023.

Arm, S.T., D.E. Holcomb, R. Howard, and B. Riley. "Status of Fast Spectrum Molten Salt Reactor Waste Management Practice." December 2020. <https://www.osti.gov/servlets/purl/1761520/> (Accessed April 3, 2023).

Carlson, K., L. Gardner, J. Moon, B. Riley, J. Amoroso, and D. Chidambaram. "Molten Salt Reactors and Electrochemical Reprocessing: Synthesis and Chemical Durability of Potential Waste Forms for Metal and Salt Waste Streams." *International Materials Reviews.* Vol. 66. pp. 339–363. 2021.

DOE. "Converting UNF Radioisotope Inventories into Energy Workshop." DOE Advanced Research Projects Agency-Energy. 2021a.

DOE. "Optimizing Nuclear Waste and Advanced Reactor Disposal Systems." DOE Advanced Research Projects Agency-Energy. 2021b

Ebert, W., and J. Fortner. "Analyses of Iron Phosphate Glasses for Dehalogenated Salt Waste." ANL/CFCT-19/5. Lemont, Illinois: Argonne National Laboratory. 2019. <https://publications.anl.gov/anlpubs/2019/07/153602.pdf> (Accessed 2 May 2025).

Ebert, W.L., C.T. Snyder, B.J. Riley, and S.M. Frank. "Designing Advanced Ceramic Waste Forms for Electrochemical Processing Salt Waste." Argonne National Laboratory, Pacific Northwest National Laboratory, Idaho National Laboratory. 2016. <https://publications.anl.gov/anlpubs/2016/04/126582.pdf> (Accessed 26 March 2024).

EPRI. "Evaluation of Existing Storage Technology for Irradiated Advanced Reactor Fuel: Options for Onsite Containerized Storage of Irradiated Fuel and High-Level Waste Arising from Advanced Reactors." Palo Alto, California: Electric Power Research Institute. 2022.

Evarts J.S., S. Chong, J.M. Oshiro, B.J. Riley, R.M. Asmussen, and J.S. McCloy. "Ceramic-metal (cermet) Composites: A Review of Key Properties and Synthesis Methods Focused on Nuclear Waste Immobilization." *Industrial and Engineering Chemistry Research*. Vol. 63, No. 14. pp. 6,003–6,023. 2024.

Gardner, D., M. Wasnik, B. Riley, S. Chong, M. Simpson, and K Carlson. "Synthesis and Characterization of Sintered H–Y Zeolite-Derived Waste Forms for Dehalogenated Electrorefiner Salt." *Ceramics International*. Vol. 46. pp. 17,707–17,716. 2020.

Harward, A., L. Gardner, C.M. Decker Oldham, K. Carlson, T.S. Yoo, G. Fredrickson, M. Patterson, and M. Simpson. "Water Sorption/Desorption Characteristics of Eutectic LiCl-KCl Salt-Occluded Zeolites." *Journal of Nuclear Fuel Cycle Waste Technology*. Vol. 20. pp.259–268. 2022.

LaPlante, P., B. Dasgupta, and Y. Pan. “Technical Assessment of Molten Salt Reactor Fuel Salt Waste Processing—Draft Final Report.” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2025.

Lee, Ki Rak, B.J. Riley, H. Park, J. Choi, S.Y Han, J. Hur, J.A. Peterson, Z. Zhu, D.K. Schreiber, K. Kruska, and M.J. Olszta. “Investigation of Physical and Chemical Properties for Upgraded SAP (SiO_2-Al_2O_3-P_2O_5) Waste Form to Immobilize Radioactive Waste Salt.” *Journal of Nuclear Materials*. Vol. 515. 382p. 2019.

Matteo, E., L. Price, R. Pulido, P. Weck, A. Taconi, P. Mariner, T. Hadgu, H. Park, J. Greathouse, D. Sassani, and H. Alsaed. “Advanced Reactors Spent Fuel and Waste Streams Disposition Strategies.” SAND2023-08602R. Albuquerque, New Mexico: Sandia National Laboratory. 2023.

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

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

Priebe, S. and K. Bateman. “The Ceramic Waste Form Process at Idaho National Laboratory.” *Nuclear Technology*. Vol. 162. pp. 199–207. 2008.

Riley, B., J. Vienna, and W. Ebert. “Road Map for Developing Iron Phosphate Waste Forms for Salt Wastes.” PNNL-30998. Richland, Washington: Pacific Northwest National Laboratory. 2021.

Riley, B. “Electrochemical Salt Wasteform Development: A Review of Salt Treatment and Immobilization Options.” *Industrial & Engineering Chemistry Research*. Vol. 59. pp. 9,760–9,774. 2020.

Riley, B., J. McFarlane, G. DelCul, J. Vienna, C. Contescu, and C. Forsberg. “Molten Salt Reactor Waste and Effluent Management Strategies: A Review.” *Nuclear Engineering and Design.* Vol. 345. pp. 94–109. 2019.

Riley, B., J. McFarlane, G. DelCul, J. Vienna, C. Contescu, L. Hay, A. Savino, and H. Adkins. “Identification of Potential Waste Processing and Waste Form Options for Molten Salt Reactors.” NTRD-MSR-2018-000379, PNNL-27723. Richland, Washington: Pacific Northwest National Laboratory. 2018.

Riley, B., J. Kroll, J. Peterson, D. Pierce, W. Ebert, B. Williams, M. Snyder, S. Frank, J. George, and K. Kruska. “Assessment of Lead Tellurite Glass for Immobilizing Electrochemical Salt Wastes from Used Nuclear Fuel Reprocessing. *Journal of Nuclear Materials.* Vol. 495. pp. 405–420. 2017.

Wasnik, M., A. Grant, K. Carlson, and M. Simpson. “Dechlorination of Molten Chloride Waste Salt from Electrorefining via Ion-Exchange Using Pelletized Ultra-Stable H-Y Zeolite in a Fluidized Particle Reactor.” *Journal of Radioanalytical and Nuclear Chemistry*. Vol. 320. pp. 309–322. 2019.