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Technical Assessment of Molten Salt Reactor Fuel Salt Waste Processing

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TECHNICAL ASSESSMENT OF MOLTEN SALT REACTOR FUEL SALT WASTE PROCESSING —FINAL REPORT

Prepared for

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

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EXECUTIVE SUMMARY

To prepare for pre-application interactions and potential licensing reviews for molten salt reactor (MSR) technologies, the U.S. Nuclear Regulatory Commission (NRC) staff is evaluating available information associated with possible future processing of MSR 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 would require management of unique waste streams. Processing of MSR fuel salt waste may be needed to remove fission products or actinides, recover fissile and fertile material, and stabilize the waste. Variations of MSR designs involve different fuel characteristics and processing options that lead to a variety of potential waste compositions and forms. Previous reports by the Center for Nuclear Waste Regulatory Analyses (Adams et al., 2023; LaPlante et al., 2024) described the historical experience with MSR technology, current MSR designs and processing concepts, applicable information from technical reports, and information needs for transportation and storage. This report extends those efforts by identifying objectives and potential scenarios for processing the waste with the aim of executing a particular waste management strategy and identifying potential additional technical considerations and associated information needs for processing facility safety evaluations.

Potential technical considerations and information needs associated with regulating aspects of the management of MSR waste depend on the characteristics of wastes and waste forms that inform this evaluation. This report includes details on MSR processing concepts and the characteristics of their outputs, based on the technologies that have been described in technical reports. This report also describes potential processing scenarios and evaluates associated technical considerations and information needs within the context of safety-related topics.

While various formulations for MSR fuel salt have been described (McFarlane et al., 2019), salt-fueled reactor designs and their fuels are typically grouped by fuels composed of fluoride salts and 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 at nearer term for utilization in the United States (Adams et al., 2023). Chloride salt designs are 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). This report focuses on fuel salts and associated waste salts for the two groups of MSR designs for which the most information is available and are closest to deployment: fluoride-based thermal reactors and chloride-based fast reactors. These two categories are reasonably representative of the liquid-fueled MSR designs that are presently being pursued and that might be pursued in the future.

Processing of MSR fuel salt wastes may be conducted to produce waste forms acceptable for waste management. Ongoing research and analysis are addressing, to some degree, the potential for waste management strategies to be affected by different species that may be present in the salts, such as halides, alkalis, alkaline earths, rare earths, and actinides. Possible waste forms being considered include glass waste forms, ceramic waste forms, mineral waste forms, ceramic-metal composite (cermet) waste forms, and halide-metal composites (halmet) waste forms (Riley et al., 2018, 2019; Riley, 2020; Arm et al., 2020; Carlson et al., 2021).

Technical considerations and information needs that are aligned with MSR fuel salt waste processing attributes and associated hazards include the following topics:

- Likely near-term potential options for applying MSR technologies

- Details on material characteristics (including added materials), processing steps, effluents, and waste streams

- Elevated neutron activity and the importance of addressing subcriticality throughout the processing steps

- Generation of gases from MSR fuel salt waste and from waste processing, including, where applicable, fluorine, chlorine, uranium hexafluoride, fission product gases, acids, and anhydrous hydrogen fluoride

- Generation of radioactive particulates from processing operations

- Corrosion of facility equipment and structures from contact with processed materials

- Chemical hazards, including those from beryllium fluoride

- Applicability of existing codes and standards

- Heat is a key hazard at all stages of MSR fuel salt waste processing and a potential contributor to fires if the necessary conditions for fire are not properly controlled.

- Potential accident scenarios, release fractions, and atmospheric modeling of releases

- Safe decommissioning considerations

The following recommendations are provided to address the needs identified in this report:

- Focus future MSR processing-related research activities on evaluating technologies that are most likely to be used by early adopters of MSR technology.

- Engage potential applicants during pre-licensing interactions regarding plans for waste management activities involving processing to improve and narrow the likely approaches that would be used and the associated information needs.

- Monitor technical developments regarding potential processing methods and waste forms to identify the field of potential options most likely to be utilized in future projects.

- Monitor technical developments regarding radionuclide inventory and source terms for MSR technologies to help clarify potential technical considerations.

- Conduct further research or monitoring of developments regarding MSR processing-specific accident scenarios to improve risk insights.

- Evaluate the applicability of codes and standards that may be applied to the design of an MSR fuel salt waste processing facility.

This report documents an initial screening of topics pertaining to the safe processing of MSR fuel salt waste, with the objective of highlighting topics that may benefit from further technical assessment. The information provided aims to further inform the NRC staff in evaluating whether additional preparation or guidance may be beneficial to accommodate the increased industry interest in potential applications of MSR technology.

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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.

ABBREVIATION/ACRONYMS

ASTM	American Society for Testing and Materials
CNWRA®	Center for Nuclear Waste Regulatory Analyses
CWF	ceramic waste form
DOE	U.S. Department of Energy
DSWF	dehalogenated salt waste forms
EBR-II	Experimental Breeder Reactor-II
GBS	Glass-bonded sodalite
halmet	halide-metal composites
HEPA	high-efficiency particulate air
INL	Idaho National Laboratory
LANL	Los Alamos National Laboratory
LLRW	low-level radioactive waste
LWR	light-water reactor
MSRE	Molten-Salt Reactor Experiment
MSR	Molten Salt Reactor
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
SNF	spent nuclear fuel
SRS	Savannah River Site
USHYZ	ultrastable H-Y zeolite

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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 processing technologies that could be used to manage irradiated fuel salt wastes generated from liquid-fueled MSR designs. Liquid-fueled MSRs use molten salt as the coolant and fuel, and implementation of MSRs would require management of different waste streams. Fuel salt waste processing may be needed to remove fission products or actinides, recover fissile and fertile material, and stabilize the waste. MSR technology was demonstrated in an experimental setting in the 1960s (Haubenreich and Engel, 1970), and several companies are currently pursuing development of new advanced reactors with molten salts in the design (Adams et al., 2023).

The various liquid-fueled 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 for the NRC (Adams et al., 2023; LaPlante et al., 2024) described historical experience with MSR technology, current MSR design and processing concepts, applicable information from technical reports, and storage and transportation information needs. This report extends that effort by identifying reasonable scenarios for processing and identifying potential technical considerations and associated information needs for future guidance development pertaining to processing facility safety evaluations.

Chapter 2 of this report summarizes the various MSR fuel salt waste processing objectives, concepts, and methods. Chapter 3 describes potential waste processing scenarios and the characteristics of wastes and applicable waste forms that have been described in technical reports, informing the assessment of information needs documented in Chapter 4. Chapter 5 summarizes results and conclusions.

2 MOLTEN SALT REACTOR FUEL SALT WASTE AND BACK-END PROCESSING CONCEPTS

The characteristics of molten salt reactor (MSR) fuel salt waste would vary depending on the reactor design, operational considerations, and processing options that are implemented. To inform the assessment of potential information needs associated with licensing a future MSR fuel salt waste processing facility, this chapter summarizes information about MSR designs (Section 2.1), associated fuel and waste materials (Section 2.2), processing objectives (Section 2.3), fuel salt waste processing methods (Section 2.4), and waste form processing technologies (Section 2.5). Descriptions within these sections address important variations that affect the types of materials, processing, and associated facilities that can inform the assessment of potential information needs to help the U.S. Nuclear Regulatory Commission (NRC) staff prepare for potential future licensing reviews.

2.1 Molten Salt Reactor Experience and General Concepts

The degree of processing that has been described in technical reports varies depending on several characteristics, including reactor design, operational characteristics, and fuel cycle concept. Some MSR designs do not currently include online processing or involve removal of fuel salt waste for offsite processing or disposal. However, as future proposals involving processing of MSR fuel salt waste are possible, the characteristics of unprocessed fuel salt waste are described here in a manner that also informs preparations for reviews related to licensing a processing facility.

Adams et al. (2023) summarized the experience with on-site storage and waste management considerations and planning for the MSR fuel salt (fuel and coolant) waste at Oak Ridge National Laboratory (ORNL) after the Molten Salt Reactor Experiment (MSRE) was shut down in 1969. Some storage of unprocessed MSR fuel salt waste at a processing facility can be reasonably expected, unless the processing facility is designed and operated to process all incoming materials immediately without storage.

Potential safety issues associated with storage of the MSRE fuel salt waste at ORNL were identified, including criticality safety issues, chemical and radiological issues, and material issues (Sautman, 1995; Peretz, 1996; Atz and Joseph, 2022; Price, 2022). Storage of the solidified fluoride-based fuel salt at the MSRE resulted in radioactive material migration, which was detected in the drain tank piping and the off-gas system. The source of the radiation was volatile uranium hexafluoride complexes formed via radiolytic decomposition of the solidified fuel salt. Under certain conditions, storage of MSR fuel salt could result in accumulation of fissile material in off-gas filters due to uranium hexafluoride mobilization, which at ORNL presented a criticality risk that prompted the removal of uranium from the solidified fuel salt. In addition to criticality safety concerns, the potential hazards associated with high radiation levels and reactive fluorine gas were worker safety considerations. Following shutdown of the MSRE, ORNL observed the production of fluorine gas from the irradiation of the solid fuel salt (Peretz, 1996; Haubenreich, 1970). Irradiation produced fluorine radicals that then combined to form fluorine gas that would diffuse to the salt surface. The presence of moisture and fluorine gas in an off-gas system increased the potential for stress corrosion cracking in off-gas system piping. The buildup of corrosion products in the drain line also caused a blockage in the drain line and postponed the effort to transfer the MSRE fuel salt waste from the existing drain line to salt canisters.

Specific scenarios for management of the MSRE salts in an ORNL analysis (Peretz, 1996) included reuse at Los Alamos National Laboratory (LANL); separation of the uranium with chemical stabilization of the remaining salt; calcination and possibly eventual vitrification of the salt at Idaho National Laboratory (INL); blending of the salt into the feed stream for the Defense Waste Processing Facility at Savannah River Site (SRS); and construction of an electrorefining system or other new process at ORNL or another site.

The description in the ORNL analysis of applying electrorefining to the MSRE fuel salt waste includes detailed steps, the waste materials that would be expected from each step, and how such an approach could be integrated with the Experimental Breeder Reactor-II (EBR-II) electrorefining process at INL. Although there is substantial recent and ongoing focused research related to MSR fuel salt waste processing methods, the only detailed descriptions of complete processing systems that were identified in publicly available references are in the ORNL alternatives analysis for the disposition of MSRE fuel salt waste (Peretz 1996) and within the long-standing U.S. Department of Energy (DOE) effort to process EBR-II spent fuel using electrorefining and associated waste form processing (DOE, 1996; Priebe and Bateman, 2008). Therefore, these publicly detailed cases are given particular attention in this report, along with pertinent aspects of some recent research.

2.2 Fuel Materials and Waste

While 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 compositions 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 Composition	Coolant Composition	Reference
Fluoride-based Thermal Spectrum	Molten Salt Reactor Experiment	$^{7}\text{LiF-BeF}_2\text{-ZrF}_4\text{-UF}_4$ (65-29-5-1 mol%, respectively)	$2(^{7}\text{LiF})\text{-BeF}_2$	Riley et al. (2018) Serp et al. (2014)
Chloride-based Fast Spectrum	Molten Chloride Reactor Experiment	$\text{UCl}_3\text{-NaCl}$ (36-64%, respectively)	$\text{UCl}_3\text{-NaCl}$ (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.

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As with other nuclear reactor fuels, following irradiation in a reactor, the resulting fission reaction and other processes generate a wide variety of additional radioactive isotopes that are typically classified into sets. In a nuclear physics context, these materials are typically classified as actinides, transuranics, fission products, and associated decay products (see Text Box), and these terms are typically used in the regulation and management of nuclear wastes. Because chemistry is central to MSR fuel salt waste processing concepts, other element sets that are based on chemical characteristics (addressed in the structure of the periodic table) are also typically considered in waste processing research. This includes, in addition to actinides, noble metals (ruthenium, rhodium, palladium, silver, osmium, iridium, platinum, gold), halogens (fluorine, chlorine, bromine, iodine, astatine), rare earths (scandium, yttrium, lanthanum, cerium, praseodymium, neodymium, promethium, samarium, europium, gadolinium, terbium, dysprosium, holmium, erbium, thulium, ytterbium, lutetium), alkali (lithium, sodium, potassium, rubidium, cesium, francium), alkali earths (beryllium, magnesium, calcium, strontium, barium, radium), and lanthanides (rare earth elements from lanthanum to lutetium). In the physics context, fission products and their associated decay products are among the constituents of these element groups. Fission products in irradiated reactor fuel contribute to high radiation fields that typically require controls to maintain radiological safety.

Radioactive Materials in Irradiated Reactor Fuel

Actinides include actinium, thorium, protactinium, uranium, neptunium, plutonium, americium, curium, berkelium, and californium. Many 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 that includes elements beyond uranium (neptunium, plutonium, americium, curium, berkelium, and californium) in the periodic table.

Fission products and their associated **decay products** are numerous but include, for example, isotopes of strontium, technetium, iodine, and cesium. 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.

The current assessment examined additional information related to unprocessed MSR fuel salt waste characteristics, including radionuclide inventory, dose rates, fissile material content, and thermal power. Inventory estimates are included in studies by Krall et al. (2022); NAS (2022); Wheeler et al. (2021); EPRI (2021); and Choe et al. (2018). Choe et al. (2018) provided an analysis of fuel cycle options for the thermal spectrum Terrestrial Energy Integral Molten Salt Reactor that included information on generated wastes. EPRI (2021) estimated fuel salt fission product concentrations for a chloride fast reactor concept. NAS (2022) evaluated wastes generated (total annual masses of fission products, minor actinides, uranium, and plutonium) from a thorium-fueled thermal MSR with continuous processing. These provide general (e.g., non-radionuclide-specific activity or mass inventories) or relative/normalized results without itemizing detailed inventories.

Peretz (1996) described the calculated surface dose rate and fissile mass in postulated canisters of MSRE fuel salt waste as part of an evaluation of waste management alternatives. This and additional information from that report is summarized in Table 2-2. MSRE was a low power (8 MW thermal) experimental MSR that operated for 1.3 equivalent full power years (Haubenreich and Engel, 1970). MSRs with a higher power rating and longer operational period than the MSRE would be expected to generate larger inventories. Additionally, some of the values in Table 2-2 account for removal of uranium by fluorination (by addition of fluorine to uranium tetrafluoride, described in Section 2.4.4) and reflect decay during decades of storage. Although the MSRE analysis indicated criticality metrics were within acceptable limits, this conclusion was limited to fuel salt waste where the uranium had been removed by fluorination.

Table 2-2. Characteristics of MSRE Fuel Salt Waste and Flush Salt Waste in Three Postulated Cans Contained Within a Postulated Transuranic Waste Canister

Characteristic	Fuel Salt Waste	Flush Salt Waste
Volume of salt in canister (m^3)	0.235	0.24
Mass of salt in canister (kg)	581	533
Specific activity (Ci/L)*	13.4	0.22
Pu-239 equivalent activity (PE-Ci)	36	0.05
Pu-239 fissile gram equivalents (g)**	95	4
Surface dose rate (R/hr)***	10	0.36
Thermal power (W)****	<14.2	<0.25
Transuranic content (nCi/g)	61,900	1,240

Source: 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).

*Activity assumes decay to 1999 (fuel was discharged in 1969)

** Based on the sum of U-233, U-235, and Pu-239 inventories in fluorinated salt waste

***Calculated contact dose at side of salt can with 2 in. steel shielding. Unshielded surface doses are 270 R/hr for the fuel salt and 4.5 R/hr for the flush salt.

****Thermal power is not reduced to account for uranium migration or fluorination.

Notz (1985) presented physics-based ORIGEN depletion and decay calculations of the time-dependent MSRE fuel salt inventory of actinides and fission products starting at the time of discharge. More recently, Wheeler et al. (2021) conducted initial source term modeling for molten salt reactors and provided general inventory estimates for a hypothetical fluoride-based thermal spectrum MSR with an output of 1 GW of power after one year of operation. They also report a fuel salt volume equal to 1.47×10^5 liters and a full core inventory of 3.02×10^{10} Ci. For comparison, the reported MSRE volume of fuel salt is 1.88×10^3 liters and the inventory of fuel salt at the time of discharge was reported as 1.8×10^6 Ci (Notz, 1985). Therefore, for the hypothetical 1 GW of power MSR reactor, Wheeler estimated a fuel salt volume that was 78 times greater than the reported 8 $\mathrm{MW_t}$ MSRE and a full core fuel salt radioactivity inventory that was over 4 orders of magnitude larger than the reported MSRE fuel salt radioactivity inventory at discharge. The sparse available information on radionuclide inventories for MSR fuel salt waste relative to typical LWR spent nuclear fuel (SNF) and the wide variation in possible options for MSR designs and associated operational characteristics that can affect inventory indicate a potential need for additional depletion and decay calculations for the larger sized and more current MSR designs.

2.3 Fuel Salt Waste Processing Objectives and Fuel Cycle Considerations

Past evaluations have identified potential challenges to management of untreated waste MSR salts that prompt considerations for processing. These challenges are related to hazardous material content, corrosivity, and waste form performance (Riley et al., 2018). Available technologies are being explored for separating constituents in MSR fuel salt waste for various reasons, including facilitation of waste management and reuse of materials. Some objectives pertain to or implicitly assume an alternative or modified fuel cycle.

This section explores various reasons including those described in technical reports and literature for specific processing options. Understanding the objectives for processing informs the potential processing scenarios that are described in Chapter 3.

Facilitate Safe and Effective Waste Management

Processing MSR fuel presents the potential to address a variety of objectives applicable to storage and transportation of the material including:

- Remove or reduce chemicals or radioactive isotopes that present challenges for waste management. Such challenges can include removal of:
  - Reactive or corrosive materials such as halides (i.e., dehalogenation) that could affect waste form or packaging system or cask system integrity. For example, chloride can significantly accelerate general corrosion rates of engineering alloys (NRC, 2015).
  - Elements that are volatile or otherwise contribute to gas generation (e.g., fluorine, chlorine, iodine, xenon, and krypton).
  - Radionuclides that contribute significantly to decay heat. Notz (1985), in evaluating thermal power calculation results for MSRE fuel salt waste, indicated a rapid decline in thermal power during the first 100 years (which is dominated by fission products), with actinide activity controlling thermal power after 100 years. Rochman, et al. (2023), evaluated decay heat calculations for LWR SNF and noted the decay of fission products such as Rh-106, Pr-144 or Cs-134 are important contributors to the decay heat for short cooling times (less than a few years), and decays of Sr-90, Y-90 and Cs-137, Ba-137m are major contributors for the first tens of years of cooling time, whereas actinides such as Pu-238 or Am-241 represent the main sources of heat for 100–200 years of cooling time.
  - Radionuclides that contribute significantly to radiation levels (e.g., Cs-137, Ba-137m, Sr-90, and Y-90) (Peretz, 1996)
  - Fissile materials that present a potential criticality concern (e.g., U-233, U-235, and Pu-239) (Peretz, 1996)
  - Materials that have elevated toxicity (e.g., beryllium fluoride and fluorine) (McFarlane et al., 2019)
- Reduce the volume of waste that presents the greatest hazard by separating the most concentrated radioactive materials from the salt (nonradioactive salt constituents occupy the greatest volume in liquid-fueled MSR fuel salt waste).
- Separate the long-lived actinide isotopes and Tc-99 from the short-lived fission products to facilitate:
  - The concept of actinide reuse in fuel for fast reactors (involves change in the fuel cycle).
  - Shortening the time horizon for disposal performance of some waste streams.

- Produce waste materials with a composition compatible with processing for waste forms that have properties aligned with long-term waste management.
    - Waste constituents can negatively affect waste loading, durability, longevity, containment/confinement performance, and solubility characteristics of waste forms.

Facilitate Safe Reuse of Materials

Processing MSR fuel salt waste presents the potential to extract useful, valuable, and/or rare materials for reuse including:

- Fissile and fissionable materials including uranium, with or without transuranics, for reuse in fuel for reactors;
- Enriched isotopes for use in new MSR fuel, in particular Li-7 and Cl-37
- Coolant salt materials for reuse in an electrorefiner or reactor; and
- Rare elements such as platinum and ruthenium.

Fission products (e.g., lanthanides) can absorb neutrons and diminish a nuclear reactor's efficiency, so would need to be removed from any material intended for reuse in reactor fuel. Lanthanides have similar chemical characteristics as actinides and can be difficult to separate from actinides in spent fuel. Therefore, separating actinides from fuel salt waste to remove lanthanide contamination is feasible but presents some challenges (Hawthorne, 2023; McFarlane, 2023).

Reduce Cost

The complexity of facilities can affect construction, operation, and decommissioning costs. Although it's reasonable to consider a variety of potential processing options, some potential future license applicants may favor simpler and lower cost options over more complicated or more thorough processing options if they can be implemented safely and satisfy expected storage requirements.

2.4 Fuel Salt Waste Processing Concepts and Technologies

Adams et al. (2023) summarized high temperature electrochemical reprocessing (pyroprocessing) as a technology that allows separation of fissile material (e.g., actinides including uranium and plutonium) from fission products for reuse. The separated fission products could be incorporated into a ceramic waste form, such as within ceramic encapsulated sodalite or other forms, and then be disposed. Separation can result in a relatively larger quantity of residual salt that some authors have suggested could possibly be stabilized and disposed of as low-level radioactive waste (LLRW). Many processes are available for treating salt wastes from electrochemical reprocessing. Salt separation techniques include vacuum distillation, oxidative precipitation, reactive precipitation, melt crystallization, dehalogenation, phosphorylation, and zone freezing (Riley, 2020). Salt processing and waste form options for different MSR waste streams are reviewed by Riley et al. (2018, 2019; Riley, 2020). Arm et al. (2020) noted the selected equipment, methods, and procedures for separating solids from fuel salts have not been disclosed by developers to date; this is still the current status.

2.4.1 Electrochemical processing

Electrochemical treatment (i.e., electrorefining and pyroprocessing) is described by several authors (Arm et al., 2020; McFarlane et al., 2019; Riley et al., 2018; Peretz, 1996) as a means to conduct separations on molten fuel salt waste. Separations that would 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. Peretz (1996) broadly describes the purpose of electrochemical treatment in the context of evaluating alternatives for processing MSRE waste as the extraction of actinides and fission products from the fuel/flush salt, concentrating the radioactive constituents in a small volume of high-level waste, and leaving a large volume of fluoride salt residue containing trace amounts of radioactivity. Electrorefining is well matched for processing a halide salt containing uranium, other actinides, and fission products, and would decontaminate the salt matrix sufficiently for it to be disposed of as low-level waste (Peretz, 1996).

In more recent descriptions of electrorefining in the context of preparing for potential future MSR projects, Arm et al. (2020) indicate that the pyrochemical process has been amply demonstrated for chloride salt. They note that capture of fission products within zeolite ion exchange media has been demonstrated for the pyrochemical process, while they are separated as salts in the reductive separation process. In the latter case, they note a final waste form is likely achievable but needs to be explicitly identified. In evaluating the potential hazards associated with MSR processing, McFarlane et al. (2019) describe that the collected materials must be stored appropriately for possible recycling (in the case of the carrier salts[1]) or storage for eventual disposal (fission products). They further caution that electrochemical processing involves working with spent fuel (and the associated very high radiation levels); therefore, such operations must be performed remotely (McFarlane et al., 2019).

2.4.1.1 Experimental Breeder Reactor II Fuel Processing

The DOE EBR-II spent fuel processing program applies electrochemical processing methods to extract spent fuel uranium for re-use with plans to process the remaining waste streams into manageable forms. The electrochemical processing employs a chloride-based electrolyte and therefore the methods are, unless modified, more directly applicable to processing chloride-based MSR fuel salt waste. This electrochemical processing uses electrochemistry to separate the uranium from the salt in a metallic form.

DOE previously described the electrochemical processing approach for EBR-II spent fuel (DOE, 1996; Priebe and Bateman, 2008). The primary fluid is a molten mixture of two salts: lithium chloride and potassium chloride (i.e., LiCl-KCl). DOE indicated voltages are set to optimize the collection of uranium at the cathode. The uranium, transuranics, most of the fission products, and any bonded sodium, dissolve into the salt, forming chlorides of the various elements. The transuranics and fission products remain dissolved in the molten salt after the uranium-bearing cathode is removed. The uranium deposit is mechanically stripped from the cathode and stored for a few days in an argon cell in a canister until it is processed in the cathode processor.

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[1] MSR fuel salts consist of mixtures of fissile and possibly fertile materials, halides, and carrier salts. The carrier salts may include common nonradioactive industrial salts, such as beryllium fluoride (BeF_2), sodium chloride (NaCl), calcium chloride ($CaCl_2$), sodium fluoride (NaF), magnesium chloride ($MgCl_2$), zirconium fluoride (ZrF_4), zirconium chloride ($ZrCl_2$), and lithium fluoride (LiF). (McFarlane et al. 2019).

DOE indicated the cathode product is heated to approximately 1,300°C [2,372°F], melting both the uranium and the salt. The salt is more volatile than the uranium. Under vacuum conditions, the salt distills off the uranium and condenses in a receiver crucible for later recycling back to the electrorefiner. The uranium is melted in a furnace crucible and then solidified into an interim product ingot, which is briefly stored before final treatment in a casting furnace. The ingots are held in the facility pending removal for storage in a controlled storage facility until decisions regarding their final disposition are made. DOE considered the option of selling the uranium metal for commercial fuel use. Information regarding the purity of the uranium product and whether it includes other fissile or fissionable actinides was not provided in the referenced reports.

DOE had at one time planned to treat the electrolyte (at end of use) to remove the fission products and transuranic elements using zeolite ion exchange; however, a recent description of processing plans indicated this may have evolved to mixing ground electrolyte with zeolite to generate a sodalite waste form that would be mixed with glass frit and heated to form a glass ceramic composite material (e.g., glass-bonded sodalite). DOE is also considering other potential waste form options for this material (Section 2.5).

2.4.1.2 Molten Salt Reactor Experiment Processing Concept

Peretz (1996) described an electrochemical processing approach for MSRE fuel salt waste that was part of facility remediation planning. The approach was part of a broader remediation option that was not implemented; MSRE fuel salt waste was instead processed by fluorination (by addition of fluorine to uranium tetrafluoride, described in Section 2.4.4) to remove uranium and the waste remains in the drain tank pending further consideration of remediation options (McMillan, 2019). Additional information about the MSRE remediation options is summarized by Adams et al. (2023) and in detail by Peretz (1996). The processing approach described by Peretz (1996) is summarized here as a documented way to apply electrochemical processing to fluoride-based MSR fuel salt waste.

The first step in the processing is a separation step: removal of zirconium from the salt and replacement with lithium. An electrochemical reaction is employed and the resulting products (which plate out on the cathode) are zirconium metal dendrites, plus some metal fission and decay product waste (i.e., metal or metalloid materials that are less reactive than zirconium, including molybdenum, ruthenium, palladium, cadmium, rhodium, technetium, selenium, antimony, niobium, polonium, bismuth, lead, and thallium). The uranium, transuranics, and rare earth fission products stay dissolved in the salt, along with other fission products such as Sr-90. Peretz (1996) indicates this step would not be needed if fuel does not contain zirconium. At this step, the uranium has not been converted to a metallic form and has not been separated.

If uranium is to be separated by electrochemical processing, active metals (except the beryllium in BeF) are first removed from the salt by electrochemical titration with lithium into a liquid bismuth cathode. The resulting product would be a bismuth metal ingot containing uranium, transuranics, and rare earth fission products.

The continuation of uranium separation proceeds with the loaded liquid bismuth cathode ingot from the previous step being used as the anode of a second cell containing LiCl-KCl electrolyte. The resulting product would be a high purity uranium metal produced at the cathode of this second cell with the electrolyte containing the remaining material from the bismuth ingot including transuranics, rare earths, and thorium and its decay products. This electrolyte waste stream would require further processing to a final waste form. Peretz (1996) suggested the chloride electrolyte, LiCl-KCl, which is the same as the EBR-II chloride electrolyte, might be incorporated into a ceramic waste form similar to what has been proposed for EBR-II wastes.

An additional processing sequence may occur if uranium was already removed by other methods, such as fluorination (by addition of fluorine to uranium tetrafluoride to create a more easily separable product, described in Section 2.4.4). In that case, the previous two steps are skipped in favor of the following where the remainder of the plutonium, rare earth fission products, the Cs-137 and decay product Ba-137m, and the thorium (including daughters of Th-228 decay) are removed from the fluoride-based electrolyte into a bismuth cathode to generate waste in the form of a bismuth ingot. Peretz (1996) suggested that the bismuth ingot could then be processed into a glass-ceramic waste form.

The remaining fluoride electrolyte would still contain Sr-90 (and decay product Y-90). Sr-90 is a bone-seeking beta emitter that has an approximate 29-year half-life, which would present a radiological safety issue for the residual fluoride-based electrolyte material. As such, Peretz (1996) suggested its removal by applying another round of electrorefining with a nitride- or oxide-enhanced technique for extraction of the strontium at the cathode. It was noted the specific technique would need to be further developed.

If the Sr-90 were removed from the fluoride-based electrolyte, the resulting final waste is a matrix salt of the form $LiF\text{-}BeF_2$ (or $NaF\text{-}LiF\text{-}BeF_2$) stripped of actinides, transuranics, and fission products. Peretz (1996) suggested this waste salt material could be incorporated into fluorapatite and potentially disposed of as low-level radioactive waste.

2.4.2 Dehalogenation

Halogens (in particular fluoride and chloride) in MSR fuel salt waste complicate waste management and limit potential waste form options, limit waste loading, and potentially increase costs. For example, glass waste forms typically do not incorporate halides. Many ceramic waste forms can incorporate halides like chloride; however, the presence of halides can reduce the waste loading capacity of waste form materials (Riley, 2020). Removal of halides is therefore desirable for long-term waste management.

Riley et al. (2018) describes a process that has been evaluated by researchers at the University of Utah using a chloride salt simulant where ultrastable H-Y zeolite is reacted with LiCl-KCl eutectic salt containing fission products in a high-temperature blender. This process removes the halides from the salt by reaction with H^+ species to form volatile acids that are captured in a condenser. The processed waste is salt-occluded zeolite that could be incorporated into a low melting point glass such as bismuth silicate glass to create a composite waste form (such as a glass-bonded zeolite) or incorporated (by decomposition of the zeolite) into a high melting point glass. Riley suggested the process could be applied to defluorinate fluoride-based salts, but that has not been demonstrated. The context of this description is processing electrochemical salt waste that contains fission products; the described process would likely still be applicable to salt that contains both actinides and fission products (e.g., dehalogenation of otherwise unprocessed MSR fuel salt).

Riley et al. (2018) also describes the concept of dehalogenation with phosphates based on previous work where chloride salts and phosphoric acid (Siemer, 2012) or ammonium dihydrogen phosphate (Donze et al., 2000) react to generate ammonium chloride that can be collected and possibly recycled. The cations from the salt can be incorporated into a phosphate glass (Riley et al., 2018).

2.4.3 Vacuum Distillation

Vacuum distillation takes advantage of differences in boiling points of combined materials to achieve separation. The application of a vacuum reduces pressure, which lowers boiling points. DOE has described the proposed application of vacuum distillation to remove adherent salt (from the electrorefiner) that would contaminate the uranium deposited on the EBR-II processing cathode during the cathode processing step (DOE, 1996). DOE described the vacuum distillation apparatus as primarily consisting of a furnace with a vacuum system in an argon cell, containing a condensing trapping area in the vacuum system. DOE indicated the cathode product would be heated to approximately 1,300°C [2,372°F], melting both the uranium and the salt. The salt is more volatile than the uranium. Under vacuum conditions, the salt would distill off the uranium and be condensed in a receiver crucible for later recycling to the electrorefiner. The uranium would be melted in the furnace crucible and then solidified into an interim product ingot, which would be briefly stored before final treatment in the casting furnace. Vacuum distillation has also been described for recovering lithium fluoride (Riley et al., 2018).

2.4.4 Fluoride Volatility (Fluorination)

Fluoride volatility is a process technology developed to separate uranium fluoride and other materials from fluoride salt solutions (Peretz, 1996); the name comes from the volatility of uranium hexafluoride. This process has been used to separate uranium from carrier salt used in both the MSRE and its predecessor, the Aircraft Reactor Experiment. The MSRE carrier salt was composed of a mixture of LiF, BeF_2, and ZrF_4 (Peretz, 1996). Fluorination of uranium (to uranium hexafluoride) is thus another option for facilitating separation of uranium from fluoride-based MSR fuel salt waste if an electrochemical processor is not used.

The flowsheet for the process used to separate uranium from the MSRE is shown in Figure 2-1. As described by Peretz (1996), the salt was transferred from the drain tanks to the fuel storage tank in the adjacent fuel processing cell. The fuel storage tank, similar to drain tanks, was hung in a furnace on a support ring that rested on weigh cells. The tank capacity

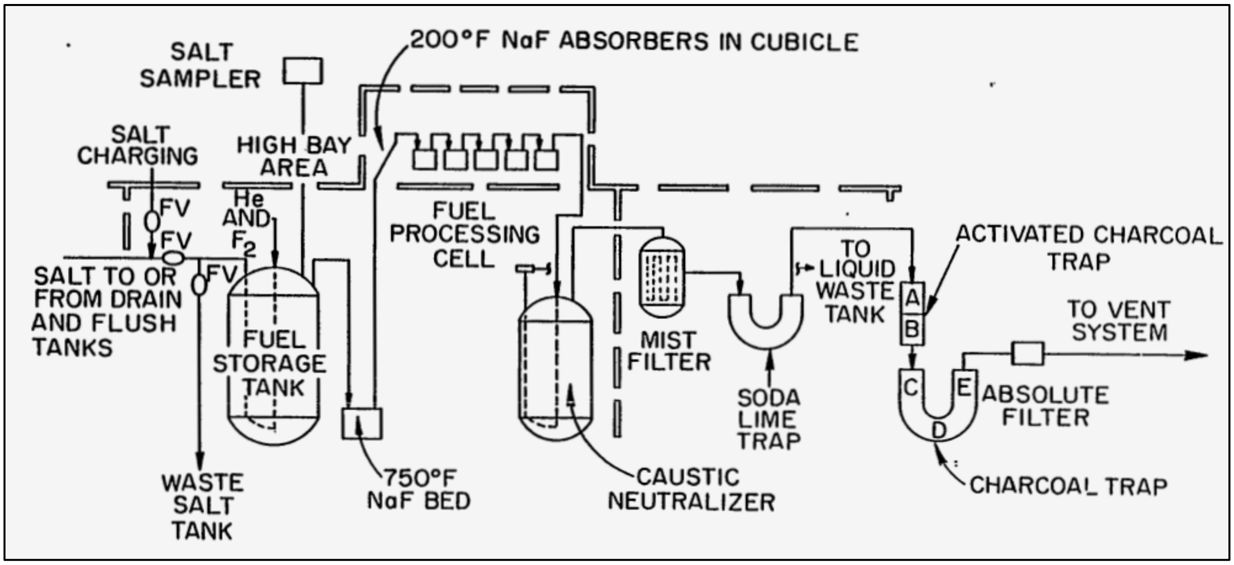

Figure 2-1. Fluorination Process for the Molten Salt Research Reactor Fuel Salt Waste (Peretz, 1996).

allowed the entire inventory of fuel or flush salt to be fluorinated in one batch with sufficient freeboard to minimize salt carryover in the outlet gas. The salt was sparged with either pure fluorine or a fluorine-helium mixture at a relatively high flow rate (40 L/min) to convert the uranium tetrafluoride to uranium pentafluoride as a chemical intermediate prior to uranium hexafluoride evolution. The outlet gas passed through a high-temperature {399 °C [750 °F]} sodium fluoride trap located near the floor of the fuel processing cell. At this temperature, volatile fission products collected on the trap and uranium passed through. Uranium was then collected on a set of lower-temperature {93 °C [200 °F]} sodium fluoride traps located in a cubicle in the high bay. Excess fluorine was then reacted with potassium hydroxide in a liquid caustic scrubber, and the outlet from the scrubber passed through a mist filter, a soda lime trap, several stages of charcoal, and a high-efficiency particulate air (HEPA) filter before being discharged to the MSRE ventilation system.

2.4.5 Vaporization/volatilization

Differing volatility of composite materials can be used for separations. McFarlane (2023) noted certain noble metal particles may need to be removed prior to encapsulation in a waste form and, in general, the noble metals appear to be easily separated from the bulk of the salt; however, separation of one noble metal from another may not be feasible or required. Long-lived Tc-99—which the author categorized as a noble metal—was noted as an exception and that it might be removed via volatility.

Riley et al. (2018) noted magnesium fluoride has been used for the selective trapping of volatile technetium fluorides and technetium oxyfluorides comingled with uranium hexafluoride, citing work by Golliher et al. (1963), Katz (1964), and Simmons (1996). They elaborated that magnesium fluoride can be agglomerated with water, pressed into pellets, dehydrated, and sintered at high temperatures to produce porous pellets for the selective trapping of technetium. Desorption of trapped Tc from the magnesium fluoride requires temperatures > 1,000 °C [1,832 °F]; however, trapped technetium can be removed by washing with water or dilute nitric acid and the magnesium fluoride can be reused after drying (Golliher et al., 1963).

2.4.6 Chemical Extraction

Chemical extraction can be accomplished based on the preferential solubility of a chemical within a mixture to a specific solvent. Riley et al. (2018) describes lithium fluoride as highly soluble in anhydrous hydrogen fluoride, while higher-charged cationic species are not. Therefore, they indicate anhydrous hydrogen fluoride could be used to recover the lithium fluoride from electrorefiner salt waste based on work conducted by Campbell and Cathers (1960).

2.5 Waste Form Processing Concepts and Technologies

Treatment and immobilization concepts and technologies for MSR fuel salt wastes are being evaluated to produce waste forms acceptable for waste management. An important processing objective is to address 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 depending on any extractions performed for reuse of materials, which would thus remove or reduce those materials from certain waste streams. A number of possible waste forms are being considered, including glass waste forms, ceramics or mineral waste forms, ceramic-metal composite (cermet) waste forms, and halide-metal composites (halmet) waste forms (Riley et al., 2018, 2019; Riley, 2020; Arm et al., 2020; Carlson et al., 2021).

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 application. This includes the glass bonded sodalite ceramic waste form (GBS-CWF) which is compatible with chloride salts and does not require dehalogenation. Iron phosphate glass and ultrastable H-Y zeolite (USHYZ), following dehalogenation of the chloride salts, have also been demonstrated for MSR application. These dehalogenated waste forms have not been demonstrated for fluoride salt wastes. Waste form properties that may be relevant to the safety reviews for licensing a processing facility (e.g., storage) include reactivity, durability, resistance to thermal or radiological effects, and compatibility with future processing/re-canning if needed. Further discussion of these waste properties is beyond the scope of this report. Descriptions of the waste form options and associated processing steps that have been demonstrated for MSR application are provided below.

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 wastes and has been demonstrated as a waste form for salt wastes from electrochemical reprocessing of Experimental Breeder Reactor II fuel (Riley et al., 2019). Because of the high chloride content expected for chloride-based MSR fuel salts, the conventional borosilicate approach is not suitable for vitrifying these wastes (Lee et al., 2019). Halogen solubility limits in sodium aluminoborosilicate glasses are very low. The GBS-CWF is a multi-phase material where alkalis and halides from the salts are immobilized within sodalite, while the remainder fission products such as alkaline earths and rare earths are incorporated into the glass phase as oxides (Priebe and Bateman, 2008; Riley et al., 2019; Riley, 2020). 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., 400 kg 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 can incorporate alkaline earths in addition to alkalis and halides, and is thus a better option for any fluoride--based MSR fuel salt wastes that include alkaline earths (Riley et al., 2018).

Iron phosphate glass waste form

The iron phosphate 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]} such that the halides are volatilized, leaving behind an iron phosphate glass. The iron phosphate glass is not chemically durable: iron oxide is added and vitrified at higher temperatures {but 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 iron phosphate glass waste form.

Ultrastable H-Y zeolite waste form

A dechlorination process that employs the ion exchange properties of microporous zeolite frameworks has been 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 that can accommodate high salt loading. This process is referred to as the USHYZ process. The first step in this process 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 ceramic (e.g., 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.

3 MOLTEN SALT REACTOR FUEL SALT WASTE PROCESSING SCENARIOS AND WASTE STREAMS

In this chapter, the various molten salt reactor (MSR) fuel salt waste processing objectives and technology concepts described in Chapter 2 are considered in formulating reasonable conceptual processing scenarios, which in turn inform the potential processing facility attributes described in Chapter 4, Section 4.1. The potential processing facility attributes support the information needs assessment, which is conducted for each review topic in Section 4.2. A range of processing options is assumed in the following scenarios, based on the range of possibilities for future licensing submittals. General fuel cycle modifications associated with each scenario (beyond the introduction of fuel salt waste processing) are noted.

3.1 Limited Processing with Dehalogenation, Consistent with Once-Through Fuel Cycle Waste Management

One limited processing scenario emphasizing low complexity (and presumably lower cost) includes dehalogenation of unprocessed MSR fuel salt waste consistent with the methods for dehalogenation described in Section 2.4, followed by solidification of the resulting radioactive waste stream. Dehalogenation is assumed in this scenario to facilitate eventual disposal. The unprocessed fuel salt waste would contain fission products, actinides, and decay products. The process of dehalogenation by ultrastable zeolite was described in Section 2.4 in the context of processing materials with only fission products. Therefore, dehalogenation of this waste stream by addition of phosphoric acid or ammonium dihydrogen phosphate and solidification of the resulting radioactive waste in phosphate glass waste form is assumed in this scenario. The processing would generate ammonium chloride (for chloride-based fuel salt waste) or ammonium fluoride (for fluoride-based fuel salt waste) that would be trapped and possibly recycled using available methods.

3.2 Electrochemical Recovery of Fissile Uranium for Re-Use Consistent with Existing Fuel Cycle

Electrochemical recovery of fissile uranium for re-use in the existing fuel cycle would be a way to recover and re-use uranium without assuming significant changes in the current fuel cycle or reactor fleet to utilize the material. Methods similar to Experimental Breeder Reactor-II (EBR-II) fuel processing (Section 2.4.1.1) for recovery of uranium from chloride-based fuel salt or Molten-Salt Reactor Experiment (MSRE) fuel processing (Section 2.4.1.2) for fluoride-based fuel salt are assumed for any potential electrochemical recovery. It is assumed here that any electrorefiner's chloride electrolytes, used in iterative batch processing of either fuel type, would be dehalogenated at the end of their useful life to facilitate disposal (like in the limited processing scenario in Section 3.1). The chloride electrolytes would contain fission products, actinides other than uranium, and decay products, and these would likely be encased in a phosphate glass waste form after dehalogenation. The remaining fluoride-based electrolyte (in the fluoride-based fuel case) would contain Sr-90: it would be advantageous for the purposes of radioactive waste management to remove the Sr-90 (if a technique is developed) and to then store the electrolyte until the Sr-90 is decayed, or to instead incorporate the electrolyte into a fluorapatite waste form.

3.3 Combined Electrochemical Recovery of Fissile Uranium and Fissionable Actinides for Re-Use in Fast Reactors Consistent with a Modified Fuel Cycle

Electrochemical recovery of fissile uranium and fissionable actinides for re-use in a modified fuel cycle that supports fast reactors would be a way to recover and re-use uranium and other fissionable actinides, assuming that the recovered materials can be processed into fuel and utilized by reactors. Methods similar to EBR-II fuel processing (Section 2.4.1.1) for recovery of uranium from chloride-based fuel salt or MSRE fuel processing (Section 2.4.1.2) are assumed for any potential electrochemical recovery, but the electrorefining process would need to be modified to co-deposit uranium and fissionable actinides at the cathode without depositing lanthanides. Co-depositing uranium and other actinides in this way is more challenging than depositing uranium alone and may require additional research (Hoover, 2023). Additionally, one method involves pairing the deposition cathode with a gas-evolving anode that produces hazardous chlorine or fluorine gas, depending on the salt composition (Hawthorne, 2023). The electrorefiner chloride electrolytes, used in iterative batch processing of either fuel type, would be dehalogenated at the end of their useful life to facilitate disposal (as in the limited processing scenario in Section 3.1). These electrorefiner chloride electrolytes (after uranium removal) would contain fission products and decay products and would likely be (i) dehalogenated by applying the ultrastable H-Y zeolite process described in Section 2.5 and (ii) encased in a glass-bonded zeolite waste form. The ultrastable H-Y zeolite process would release volatile acid (hydrochloric acid) that would be captured in a condenser. The remaining fluoride-based electrolyte (in the fluoride-based fuel case) would contain Sr-90, as in the previous scenario; again, it would be preferable to remove the Sr-90 (if a technique is developed) and to then store the electrolyte until the Sr-90 is decayed, or to instead incorporate the electrolyte into a fluorapatite waste form.

3.4 Electrochemical Recovery of Fissile Uranium and Separation of Actinides and Short-Lived Fission Products to Reduce Long-Term Waste Volume Consistent with a Modified Fuel Cycle

To facilitate back-end waste management, this scenario involves electrochemical co-recovery of fissile uranium and other actinides and separation of actinides and short-lived fission products to reduce the volume of waste that needs to be isolated from the environment over very long timeframes. In this scenario, the fuel cycle is potentially modified at the back end, changing the waste streams to be managed. This is a variant of the combined recovery scenario (Section 3.3) where an additional electrochemical separation of co-deposited uranium and actinides is executed to allow re-use of the uranium and create separate waste streams for fission products and actinides. The fission product waste stream is further processed by volatilization (Section 2.4.5) to recover long-lived Tc-99 so it can be combined or co-managed with the long-lived actinide waste stream. The resulting fission product waste stream is comprised of shorter-lived radioactive isotopes that can potentially be managed separately from the longer-lived waste stream. The radioactive waste streams under this scenario would be similar to the preceding combined recovery scenario (Section 3.3) with the addition of a separate long-lived actinide and Tc-99 waste stream that could be incorporated into a glass waste form.

3.5 Comprehensive Processing to Optimize Re-Use of Materials and Facilitate Waste Management Objectives Consistent with a Modified Fuel Cycle

To facilitate back-end waste management and optimize re-use of materials, this scenario is a variant of either the co-recovery scenario (Section 3.3) with optimized re-use of fissile and/or fissionable materials alongside waste management, or a variant of the complete separation scenario (Section 3.4) with optimized waste management. Recovery of useful materials includes Cl-37 and Li-7. The Cl-37 is added to MSR fuel salt to limit the unwanted activation of Cl-35 to Cl-36, and Li-7 is added to limit the unwanted activation of Li-6 to H-3 (tritium) (Riley et al., 2018). The chlorine can be removed and reused by the previously described dehalogenation methods (Section 2.4.2) that are already included in this processing scenario. An additional step is added for lithium containing fluoride-based MSR fuel salt waste, wherein lithium is recovered from residual electrolyte using chemical extraction with anhydrous hydrogen fluoride (Section 2.4.6). Waste streams and forms would be the same as described in Sections 3.3 and 3.4.

4 TECHNICAL CONSIDERATIONS AND INFORMATION NEEDS WITH LICENSING A PROCESSING FACILITY FOR MOLTEN SALT REACTOR FUEL SALT WASTES

This chapter contains an assessment of safety and performance considerations, and related information needs, to support potential regulatory actions pertaining to a facility that processes MSR waste. The assessment considers potential processing facility attributes and the associated activities that could be conducted. Section 4.1 describes the attributes and associated hazards of a potential processing facility within the context of typical areas of review, based on the processing scenarios in Chapter 3 and associated processing methods in Chapter 2. The scenarios and associated facility attributes are consistent with available information but should not be considered as an endorsement of any technology or approach. Section 4.2 uses these potential processing facility attributes and hazards to assess potential technical considerations and information needs pertaining to safety-related topics.

4.1 Potential Processing Facility Attributes and Associated Hazards for Typical Areas of Review

All identified processing scenarios for MSR fuel salt waste involve operations with irradiated materials and hazardous chemicals remotely manipulated in a high-radiation and high-temperature processing environment. Associated hazards include high radioactivity materials and associated high radiation fields, the potential for criticality, working with hazardous chemicals, the potential release of gases and particulates, and the potential for fire. The following subsection describes potential processing facility attributes and associated hazards for typical areas of review that are considered technically applicable to a salt processing facility.

4.1.1 Site Description

A general description of the MSR fuel waste processing facility site is generally required in NRCs regulations, with emphasis on those factors that have safety significance. The information includes descriptions of the site geography, meteorology, seismology, topography, demography, location, and prominent natural features (e.g., rivers, population data and distribution as a function of distance from the facility). The site description indicates the presence of nearby industrial, transportation and military facilities, railroads, airports, and dams that may pose a hazard to the facility. The site description also generally includes information characterizing natural phenomena (e.g., tornadoes, hurricanes, floods, and earthquakes) and other external hazards, including their likelihood of occurrence and design-basis natural events for the facility design.

Information required for review of a site description for MSR fuel waste processing facility is expected to be similar to other NRC licensed facilities, therefore no additional technical considerations are identified.

4.1.2 Facility Description

The NRC review of the facility description is to ensure that the description of the MSR fuel waste processing facility adequately identifies and describes general features. Information that would likely be provided consists of the facility location, arrangement of buildings on the facility site, the distance from the controlled area boundaries, and design information of the structures, systems, and components (including instrumentation and controls relied on to function during accident sequences to prevent or mitigate consequences). The facility is designed to provide adequate protection against natural hazards, fire and explosions, environmental effects, chemical risks, and is designed to provide criticality control and emergency capabilities.

Most MSR fuel waste processing facility structures would be expected to be similar to the other nuclear facilities licensed by NRC. Similar types of mechanical equipment and systems used in the nuclear facilities would be expected for the processing of incoming MSR waste to the facility and/or handling of waste forms generated at the facility (e.g., packaging, transfer, and temporary storage). Other unique technical considerations are not anticipated for most MSR structures. If the salt processing facility is away from a reactor site, there will be a need for a receiving area, which allows for the unloading and inspection of canisters from casks and transfer to storage. Depending on the radioactivity of the contents, these cans will need to be suitably shielded and monitored. Similarly, any waste forms generated at the end of the salt processing will need to be stored accordingly.

Systems and equipment associated with processing of reactor fuel waste and processes for ceramic or vitrified glass waste form have generally not been used at commercial U.S. nuclear facilities, to date. Therefore, there is no precedent for their review by NRC. Configuration, capacity, and design of these systems will be specific to a facility and process design, which will depend on the fuel salt (chloride or fluoride) and also types of waste streams generated and processed. Experience can be drawn from existing experimental and laboratory scale electrochemical processing facilities used in the DOE demonstration project of EBR-II spent fuel treatment (DOE, 1996; Priebe and Bateman, 2008), which consists of (i) an electrorefiner for reactor fuel to remove uranium in an electrochemical cell, (ii) cathode processors for removing entrained salt from the separated uranium by electro deposition; (iii) a casting furnace for forming the separated and further purified uranium product into ingots; and (iv) a ceramic waste process that blends and consolidates radioactive waste components, including actinides and fission products, into a glass-ceramic matrix. Technological readiness for processing MSR fuel waste for industry scale applications has not yet been demonstrated.

Additional technical considerations are anticipated to be needed for the design, operations, and safety features of processing equipment and systems for electrochemical processing of fuel waste. Sufficient information and understanding of facility and process equipment and operations will likely be needed for any safety analysis conducted by the licensee.

4.1.3 Description and Analysis of Facility Processes, Hazards, and Accident Sequences

Descriptions of facility processes, hazards, and accident sequences provide the necessary information to support hazard and consequence analyses that are important parts of a safety analysis.

Processes

The fuel waste process may be different from traditional fuel processes when the facility is processing MSR salts. A process description would likely include (i) separation technology of the MSR fuel waste; (ii) discussions of process design; (iii) general arrangement, functions, and operations; (iv) major components, equipment, and systems; (v) drawings, process schematics and flow sheets; and (vi) instrumentation, monitoring systems, and essential support systems (e.g., electric power system, ventilation system, fire protection system, etc.). The information should also include mechanical operations of receipt and transfer of incoming MSR fuel waste to the facility. Other operational processes at a MSR waste facility may include handling, packaging, transfer, temporary storage, and preparation for transportation of waste forms generated at the processing facility.

Hazard Analysis

An evaluation of potential hazards is generally needed to identify situations associated with process steps or activities which could lead to failure of safety related items, operator errors, or external events. The evaluation should include situations that could lead to accidental radiological and chemical release, such as nuclear criticality, fire or explosions. An evaluation will likely consider: (i) a list of hazardous materials (radioactive, fissile, flammable, and toxic), including intended maximum inventory amounts and locations of the hazardous material at the facility, (ii) identification of hazards that could result in initiating events, and (iii) an evaluation of subsequent accident sequences and consequences using appropriate process hazard analysis techniques.

McFarlane et al. (2019) discussed potential hazards associated with molten salt reactor fuel processing operations and proposed mitigation strategies. The discussion considered physical and chemical processes for chloride and fluoride fuel waste form preparations. The report also presented, for each processing activity, a qualitative judgment of risk to operations staff. Qualitative extents of risk to workers would be high for most hazardous activities identified, if unmitigated. Hazards postulated by McFarlane et al. (2019) include beryllium toxicity, corrosivity, heat, high radiation and contamination, and halogens. McFarlane et al. (2019) provides example mitigation strategies that include remote handling, instrumentation, and secondary containment. McFarlane et al. (2019) did not discuss any process design, equipment, systems, or components and did not elaborate on the hazard identification process but provided first order potential hazards with regards to MSR waste processing.

Accident Sequence

An accident sequence involves an initiating event and subsequent failure of hardware controls or manual procedures that allow the accident to propagate. It is likely that a description of engineered and administrative controls relied on, either to prevent (or reduce the likelihood of) event sequences or to mitigate consequences in an accident sequence, will need to be provided in a safety analysis.

Review of accident sequences of a MSR fuel waste processing facility and application of accident evaluation tools will likely require (i) a detailed description of the process; (ii) equipment, systems and instrumentations and controls including design information and reliability data; and (iii) the operational information. The technical information discussed in facility description and process should support evaluation of accident sequences; therefore no additional technical considerations are identified for review of accident sequences.

Irradiated MSR fuel salt from a reactor contains actinides and fission products. At the MSR waste processing facilities, electrorefining of MSR waste salt will recover uranium and produce waste streams that consist of Pu, transuranic elements, alkalis and alkaline earths, and some fission products. The uranium may be cast into ingots for further use and the remaining waste stream, if not further processed, will likely be classified as high-level waste. The active fission products and transuranic elements remaining in the electrorefiner salt are likely to be processed into ceramic waste forms.

Several processes at a MSR waste processing facility (e.g., electrorefining, cathode processor, casting furnace and fabrication of uranium ingots and ceramic waste form) may lend themselves to different potential initiating events, accident sequences, and consequences than fuel cycle facilities. Radionuclide source terms and inventory in each process system or at each process step of the facility may have the potential for high dose impacts to worker and public if not sufficiently mitigated.

4.1.4 Hazard and Consequence Analysis Methods

Process Hazard Analyses

The choice of a particular method or combination of methods for hazard analysis will depend on several factors, including the complexity of the process, the maturity of the design and available information, the results needed from the analysis, and the perceived risk of the process. For example, a Checklist Analysis, Preliminary Hazard Analysis, and/or What-If Analysis are generally used for broad identification and overviews of hazards. However, when perceived risk of accident sequences are high, a What-If/Checklist, Hazard and Operability Analysis, or Failure Modes and Effects Analysis is generally recommended. These methods collectively are found to be applicable to chemical and mechanical processes at nuclear facilities and are suitable for performing detailed analyses of a wide range of hazards, including radiological, chemical, and criticality analyses.

Hazard analysis techniques have been widely used for process safety in the chemical industry and for nuclear fuel cycle facilities; however, information on applications to electrochemical processes is not readily found in the literature. The methods are flexible and expected to be effective for MSR fuel waste processing facilities to evaluate hazards associated with electrochemical processes including mechanical operations related to handling, transfer, and storage of radioactive material within the facility. Applicants may need to use one or a combination of hazard analysis methodologies and demonstrate that all relevant hazards have been identified.

Consequence Analysis Methods

With respect to consequence analysis, the safety evaluation of the facility should generally include (i) the characteristics of releases of radioactive materials and hazardous chemicals and (ii) the consequences of those releases. The model(s) used, or approach(es) taken should (i) identify release scenarios, (ii) predict the characteristics of the source term and the airborne concentrations resulting from a release, and (iii) assess the impact on human health and adequacy of mitigative measures. Additionally, emergency response planning and post-accident analysis should be addressed. Analysis methodologies, examples, and data used in NUREG/CR-6410 (SAIC, 1998) may be helpful in modeling radiological and chemical hazards at MSR fuel processing facilities. Depending on the specifics of future proposed facilities, there may be a need for evaluation of current approaches or to develop new analysis techniques and data for calculating radiological and chemical consequences at MSR waste processing facilities.

4.1.5 Chemical Hazards and Consequence Standards

It is likely that a potential applicant will need to assess the consequences from acute chemical exposures to an individual from either radioactive waste material or other hazardous chemicals produced from the processing of radioactive material. While the specifics of potential chemical hazards to workers and the public would depend on the details of the proposed processing and associated controls at a specific facility, the description of materials and potentially applicable MSR fuel waste processing and waste form processing methods in Sections 2.2, 2.4 and 2.5 of this report indicate processing could involve materials that can be acutely hazardous to human health if not properly controlled. Fuel salt constituents that present unique and notable chemical hazards include beryllium fluoride, which has been associated with dermal and inhalation-based effects (e.g., berylliosis) (McFarlane et al., 2019). Additional chemical hazards are associated with materials that can be introduced or produced during processing operations. Notable hazardous constituents associated with processing include chlorine, fluorine, acids (such as phosphoric, hydrochloric, and hydrofluoric acid), and anhydrous hydrogen fluoride. Chemical hazards would need to be controlled through design and operational considerations such as adequate ventilation and filtration systems, process controls, monitoring, and personal protective equipment.

If batch sizes are small, then chemical hazards may be constrained to a worker safety concern, which is relevant for NRC regulated radiological safety if chemical hazards interfere with workers ability to perform radiological safety functions. Depending on the chosen approach to chemical consequence assessment, a relevant performance requirement for limiting high consequence chemical hazards may be to ensure that exposures are unlikely or mitigated for acute exposures that could (i) endanger the life of a worker and (ii) lead to irreversible or other serious, long-lasting health effects to individuals located outside the controlled area. An associated performance requirement for limiting intermediate consequence chemical hazards may be to ensure that exposures are unlikely or mitigated for acute exposure that could (i) lead to irreversible or other serious, long-lasting health effects to a worker; and (ii) cause mild transient health effects to any individual located outside the controlled area.

4.1.6 Radiation Protection

As described in Section 2.2 of this report, irradiation of fuel in a reactor generates radioactive isotopes, including actinides, fission products, activation products, and associated decay products. Inventory calculations for MSRE fuel salt (Peretz, 1996) show radioactivity in discharged fuel salt is initially dominated by short-lived fission products (Ce-144, Pr-144, and Pm-147) for about 3 years after discharge. Thereafter the radioactivity is dominated by longer-lived fission products (e.g., Cs-137, Ba-137m, Sr-90, and Y-90) for several hundred years that follow. Notz (1985) highlighted significant radiation hazards associated with the stored discharged fuel salt, noting gamma and neutron dose rates within the reactor and storage cells on the order of 10 Gy/hour (1,000 rad/hour). A potential MSR fuel salt waste processing facility would have high gamma and neutron radiation fields predominantly from (but not limited to) these fission products. This radiation would need to be controlled through design and operational considerations such as shielding, distance, exposure time limits, and remote handling and operations.

Other radiological hazards include potential release of gaseous fission products from MSR fuel salt waste (such as iodine, cesium, xenon, and krypton) and potential radioactive particulates from processing operations such as electrochemical processing (such as mechanical removal of separated materials from electrodes) and processing that involves grinding of fuel salt waste materials (such as methods for producing the glass-bonded sodalite ceramic waste form described in Section 2.5). Inhalation hazards would need to be controlled through design and operational considerations such as adequate ventilation and filtration systems, process controls, monitoring, and personal protective equipment.

4.1.7 Criticality Safety

The combination of fissile material and neutron sources presents the potential for criticality in a potential MSR fuel salt waste processing facility. 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. Fissile material (including isotopes of uranium and plutonium) in the MSR fuel salt waste would be moved or flowed through processing steps that can involve separations and increasing concentrations of some materials, including fissile materials. Therefore, the specific materials, their concentrations, and their geometries may exist in a state of flux. 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). Overall, subcriticality at a processing facility would need to be maintained by the application of adequate controls through, for example, engineering (such as passive engineered geometry control) and appropriate administrative measures.

4.1.8 Fire Safety

Fire hazards are likely to be present at various areas of the facility during the processing, mechanical handling, and storage of fuel waste. The potential consequences of fire events at a facility include destruction of confinement barriers, failure or damage to utility equipment, and loss of criticality controls. Potential fire events within the facility can be caused by short circuits or equivalent events involving electrical equipment, ignition, or combustion of fixed or transient combustibles, equipment that operates at high temperatures, and ignition of flammable/reactive chemicals initiated by process upset or external natural and man-made events.

A fire or explosion at an MSR facility creates a potential risk for release and dispersal of radioactive materials or chemicals. Safety related items should be designed and located to continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Heat-resistant and noncombustible materials should be used wherever practical throughout the facility, particularly in locations vital (i) to the functioning of confinement barriers and systems, (ii) to methods of controlling radioactive materials within a facility, and (iii) to the maintenance of safety control functions. The adverse effects of fires and explosions on safety related items can be minimized by providing systems with sufficient capacity and capability for detecting and suppressing explosions and fires and by using systems for transmitting alarms to one or more central control areas. Adverse effects may result from normal operation, malfunction, or failure of a fire protection system. It is important to recognize these potentially adverse effects and eliminate or mitigate them through proper design and installation.

McFarlane et al. (2019) identified heat as a key hazard at all stages of MSR fuel salt waste processing. Heat could be a contributor to fire-induced accidents if the necessary conditions for fire are not properly controlled. Design information including safety controls and instrumentation of the equipment and system used in the electrochemical processing and operated at high temperature (such as electrorefiner, cathode processor, casting furnace, calcination furnace) should consider addressing potential fire hazards in detail. MSR fuel waste processing is performed at high temperature (McFarlane et al., 2019). The electrorefining separation of fission products and actinides from MSR salt is performed in a molten state, expected to be at atemperature greater than 500°C [932 °F]. For example, techniques proposed for actinide recovery from fluoride salts include roasting with Ca(OH)2 at temperatures up to 1,000°C [1,832 °F], heating with NaOH at temperatures up to 500°C [932 °F], and silicate roasting at 850°C [1,562 °F] (McFarlane et al., 2019). High temperature is also expected during blending of waste salt, zeolite, and glass during conversion to ceramic waste forms. Given the high temperatures, design information, including safety controls and instrumentation of the equipment used in the electrochemical processing should be provided by licensees for the evaluation of potential fire hazards.

To inform fire protection program design, systems and features, potential resources include NFPA 801, *Standard for Fire Protection for Facilities Handling Radioactive Materials*, and other NFPA publications.

4.2 Assessment of Potential Technical Considerations and Information Needs

To prepare for the possibility of licensing a facility that processes MSR fuel salt waste, potential technical considerations and information needs were determined by evaluating the materials, processing concepts and scenarios, and associated hazards described in this report. This assessment was focused on technical considerations that are related to safety, generally, and not in the context of a specific regulatory framework for licensing such a facility. These technical considerations and information needs are summarized as follows:

- A wide variety of options for MSR technology and associated fuel compositions are currently being evaluated by vendors and other researchers. A range of potential operating conditions and fuel cycle considerations are also being explored. These considerations affect potential processing needs and associated technologies that could be pursued by potential applicants. Clarity regarding the MSR technologies, operating conditions, and the chemical and radiological characteristics of the MSR fuel salt waste(s) that would be processed by a potential facility would inform preparations for NRC licensing safety reviews.

- Any potential MSR fuel salt waste processing facility would be working with irradiated reactor fuel and therefore would have high gamma and neutron radiation fields, predominantly from fission products. Adequate shielding and remote handling, operations, and maintenance would be important for maintaining worker safety under these conditions.

- The evaluation of corrosion and other reactions on processing equipment may be an important consideration regarding the safe lifecycle of important processing facility structures and components.

- The additional potential for gas generation from MSR fuel salt waste and from waste processing needs to be addressed in processing facility design and administrative controls. This should include consideration of, for example, fission product gases, fluorine, uranium hexafluoride, chlorine, acids, and anhydrous hydrogen fluoride.

- The unique nature of an MSR fuel salt waste processing facility suggests assessment of the applicability of existing codes and standards for evaluating safety may be merited.

- Heat is a key hazard at all stages of MSR fuel salt waste processing

- In an evaluation of accident consequences, the melting point of unprocessed MSR fuel salt waste could affect accident release fractions and atmospheric modeling of releases.

- Design of any processing facility should include considerations for eventual safe decommissioning.

It should be noted that salt processing facilities may have unique features that differ from currently regulated reactors and fuel facilities. Some of the unique challenges which could be encountered in salt processing facilities are:

- Processing can change the material characteristics of MSR fuel salt waste with regards to chemical composition, chemical reactivity, potential for gas generation, and homogeneity of processed materials and waste forms. A detailed accounting of feed material characteristics (including added materials), processing steps, effluents, and waste streams would be needed to evaluate the safety of a specific processing approach.

- The potential for evolving changes in fuel salt chemistry, fissile material concentrations and properties (due to decay and radiolysis), neutron sources, and geometry during processing should be addressed, as applicable, in assessments that would demonstrate a subcritical condition is maintained at all times during processing.

- Elevated neutron activity may occur in some fuel salt wastes from alpha neutron reactions due to the presence of Be-9, F-19, and Li-7. This can affect shielding and criticality considerations for a potential processing facility.

- Radiolysis of fluoride-based fuels creating fluorine gas and uranium hexafluoride was seen in the storage of the MSRE fuel salt waste. If not properly controlled, these species could potentially migrate, deposit in ventilation systems, and pose a potential criticality safety concern at a potential processing facility.

- Radioactive particulates could be generated from processing operations such as electrochemical processing (such as during mechanical removal of separated materials from electrodes) as well as during processing that involves grinding of MSR fuel salt waste materials (such as a method for producing the glass-bonded sodalite ceramic waste form described in Section 2.5). Particulates derived from irradiated reactor fuel materials could potentially impact worker and public safety. This would need to be addressed through adequate controls to prevent exposures and dispersal under both normal and accident conditions.

- Fuel salt constituents that present unique and notable chemical hazards that need to be addressed include beryllium fluoride, which has been associated with dermal and inhalation-based effects (e.g., berylliosis). Such chronic exposure hazards have not been considered previously in NRC regulatory practice.

Sufficient technical information to support license applications would be expected to be provided by applicants, but some internal NRC efforts may be appropriate to prepare for or enhance the staff capabilities to conduct future licensing reviews. The following recommendations are provided to address needs identified in this report:

- Staff should strongly encourage early interaction with potential applicants. The interaction should focus on:

    - Understanding the current design of processes and facility, focusing on operations that are significantly different from processes and facilities previously licensed. The discussion should include any successful experience at the pilot plant or industrial scale at DOE or foreign facilities.

    - Understanding any results of the preliminary safety analysis, including dominant hazards/risk to workers or the public and the risk levels. Also discuss analytical methods including use of existing codes/methods and the development and use of new codes/methods.

    - Understanding the expected applicant's thoughts/plans for the regulatory approach, including planned regulatory framework, approaches for demonstrating compliance with regulatory requirements, and expected exemption requests.

- Focus future MSR processing-related research activities on evaluating technologies that are most likely to be used by early adopters of MSR technology, hence the need for robust preapplication engagement to identify plans for waste management activities involving processing to improve and narrow the likely approaches that would be used and the associated information needs. There is a wide variety of potential design and operational options associated with MSR technology and related processing options. Focusing preparations on technologies most likely to be proposed by future applicants can help limit the associated information needs.

- Monitor technical developments regarding:

    - potential processing methods and waste forms to identify the field of potential options most likely to be utilized in future projects.

    - radionuclide inventory and source terms for MSR technologies

- Conduct further research or monitoring of developments regarding MSR processing-specific accident scenarios to improve risk insights.

- Evaluate the applicability of codes and standards that may be applied to the design of an MSR fuel salt waste processing facility.

5 SUMMARY AND CONCLUSIONS

Information regarding (i) MSR fuel salt waste materials, (ii) possible processing concepts, technologies, and methods, and (iii) potential scenarios for processing the waste was evaluated to identify potential technical considerations and associated information needs to inform preparations for processing facility safety evaluations.

MSR fuel salt wastes were considered for two near-term groups of MSR designs with the most available information: fluoride-based thermal reactors and chloride-based fast reactors. These two categories are reasonably representative of the liquid-fueled MSR designs that are presently being pursued and that might be pursued in the future.

A set of waste processing objectives was derived from reviewing applicable literature, revealing a range of MSR fuel salt processing concepts and technologies that potentially could be used to meet these waste processing objectives. A range of processing scenarios that were aligned with these waste processing objectives were examined to provide insights regarding potential processing facility attributes and hazards; these scenarios informed the assessment of potential technical considerations and information needs.

All identified processing scenarios for MSR fuel salt waste involve remote handling and operations with irradiated materials and hazardous chemicals, with processes occurring in a high-radiation and high-temperature environment. Associated hazards include working with high radioactivity materials and associated high radiation fields, the potential for criticality, working with hazardous chemicals, the potential release of gases and particulates, and the potential for fire.

Technical considerations and information needs developed from identified MSR fuel salt waste processing attributes and associated hazards include:

- Likely near-term potential options for applying MSR technologies.
- Details on material characteristics (including both the waste materials and materials added to the wastes), processing steps, effluents, and waste streams.
- Elevated neutron activity and the importance of addressing subcriticality throughout the processing steps.
- Generation of radioactive particulates from processing operations.
- Generation of gases from MSR fuel salt waste and from waste processing, including, where applicable, fluorine gas, uranium hexafluoride, fission product gases, chlorine, acids, and anhydrous hydrogen fluoride.
- Corrosion of facility equipment and structures from contact with processed materials.
- Chemical hazards including beryllium fluoride.
- Applicability of existing codes and standards.
- Heat is a key hazard at all stages of MSR fuel salt waste processing and a potential contributor to fires if the necessary conditions for fire are not properly controlled

- Potential accident scenarios, release fractions, and atmospheric modeling of releases.
- Safe decommissioning considerations.

The following recommendations are provided to address the needs identified in this report:

- Focus future MSR processing-related research activities on evaluating technologies that are most likely to be used by early adopters of MSR technology.
- Engage potential applicants early in pre-application interactions regarding plans for waste management processing activities to understand significant differences from previously licensed processes and facilities. Discussions should include any successful experiences at the pilot plant or industrial scale at DOE or foreign facilities, as well as any preliminary assessments of dominant hazards (and associated risk levels) to workers or the public.
- Conduct further research or monitoring of technical developments regarding potential processing methods and waste forms to identify the field of potential options most likely to be utilized in future projects.
- Conduct further research or monitoring of technical developments regarding radionuclide inventory and source terms for MSR technologies to help clarify potential technical considerations.
- Conduct further research or monitoring of developments regarding MSR processing-specific accident scenarios to improve risk insights.
- Evaluate the applicability of analytical methods or codes and standards that may be applied to the design of an MSR fuel salt waste processing facility.

The information provided in this report aims to further inform the NRC staff in evaluating whether additional preparation or guidance may be needed to address the increased industry interest in advanced reactors, including potential applications of MSR technology and associated fuel salt waste processing.

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