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SAFEGUARDS MATERIAL CONTROL AND ACCOUNTANCY CONSIDERATIONS FOR MOLTEN SALT REACTORS

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ABSTRACT

Molten salt reactors are a class of nuclear reactor design with varying features and operational characteristics. Variations in design features include the physical, chemical, and isotopic composition of fresh and irradiated fuel; neutron energy spectra; breeding ratios; and capabilities for chemical processing. Specific designs within these variations affect the material control and accountancy approaches implemented for both domestic and international nuclear safeguards purposes. This paper identifies safeguards-relevant design features for salt-fueled molten salt reactors and describes how each feature might influence implementations of domestic and international nuclear safeguards. Additionally, this paper describes two categories of material control and accountancy approaches that could be appropriate for molten salt reactors: (1) a black box material-balance approach with measurements of the feed and waste streams combined with robust containment and surveillance of the nuclear material within the reactor or (2) an approach that quantifies the nuclear material throughout the reactor (i.e., a process-monitoring approach). This paper describes the benefits of these approaches, the challenges to meeting technical safeguards objectives, and considerations for future safeguards technologies.

INTRODUCTION

Advanced reactor design concepts have gained significant traction as countries around the world consider nuclear power plants to meet increased energy demand using carbon-free sources. Molten salt reactors (MSRs) were one of six promising non–light water reactor (LWR) technologies selected for recognition by the Generation IV International Forum in 2002 [1]. In December 2020, the United States (U.S.) Department of Energy announced that two of the five awards for risk reduction for future nuclear reactor demonstration projects were awarded to MSRs. The two MSR recipient teams were

(1) The Hermes Reduced-Scale Test Reactor: the Kairos Power, LLC, TRI-structural ISOtropic particle fuel (TRISO), pebble-fueled reactor that uses a low-pressure fluoride salt coolant
(2) The Molten Chloride Reactor Experiment: the Southern Company Services Inc. Molten Chloride Reactor Experiment, a fast-spectrum salt reactor related to TerraPower's Molten Chloride Fast Reactor, in which fissile material is dissolved in chloride salt [2]

These are examples of the two distinct types of MSRs. The first is a solid-fueled, molten salt-cooled MSR, with no fissile material in the salt coolant. The second is a salt-fueled MSR, in which the primary loop consists of fissile material dissolved in molten-salt coolant [3].

For domestic safeguards purposes in the United States (US), any operator licensed by the Nuclear Regulatory Commission (NRC) must maintain a material control and accountancy (MC&A)

program that tracks and verifies the special nuclear material (SNM; i.e., Pu, ^{233}U, or U enriched in the isotope ^{235}U) at the facility [4]. Internationally, states with comprehensive safeguards agreements with the International Atomic Energy Agency (IAEA) must declare quantities and locations of nuclear material (i.e., Pu, U, and Th) to the IAEA. The IAEA then independently verifies those quantities. Therefore, facility operators and some regulatory agencies (e.g., the European Atomic Energy Community [Euratom], the IAEA) will need methods and technologies to quantify fissile and fertile isotopes within MSR fuel. The fissile materials (and fertile materials in the case of the IAEA) will need to be quantified in both salt-cooled and salt-fueled reactors. Existing salt-cooled MSR concepts use tristructural isotropic particle (TRISO) fuel in pebble form. The quantities of nuclear material within the non-fissile salt coolant in salt-cooled reactors will be negligible. Therefore, the MC&A challenges for salt-cooled MSR designs will be similar to the challenges for other (e.g., gas-cooled) pebble-bed reactor designs [3].

The overall goal of US domestic safeguards is to ensure that SNM is not stolen or otherwise diverted from a facility [5]. Domestic safeguards incorporate both physical protection and MC&A to achieve this goal [6]. International safeguards, applied by the IAEA, are in place to ensure that the State itself is using nuclear material and facilities only for peaceful purposes. For States with comprehensive safeguards agreements, the IAEA’s generic safeguards objectives are

- to detect any diversion of declared nuclear material at declared facilities or locations outside facilities (LOFs) where nuclear material is customarily used;
- to detect any undeclared production or processing of nuclear material at declared facilities or LOFs; and
- to detect any undeclared nuclear material or activities in the State as a whole [7].

Nuclear material accountancy and containment and surveillance (e.g., methods to control nuclear material) are commonly used as safeguards measures to meet safeguards technical objectives in facilities. This paper identifies some design features that will be relevant when developing plans to account for nuclear material within a salt-fueled MSR facility. The goal of this paper is to provide some baseline assessments and identify challenges and opportunities that could be further developed to meet either domestic or international safeguards objectives for salt-fueled MSRs. The term “MC&A” is used to generically represent the accountancy and control of nuclear material—as it pertains to both domestic and international safeguards—with the understanding that the objectives of these safeguards, as well as the methods to achieve those objectives, are distinct. Additionally, though “nuclear material,” as defined by the IAEA, and “special nuclear material,” as defined by the NRC, are distinct, this paper uses the term “nuclear material” to generally refer to fissile and fertile materials (i.e., Pu; depleted, natural, and enriched U in ^{235}U; ^{233}U; and Th). Finally, this paper focuses only on salt-fueled MSRs.

Nuclear Material in MSRs

To identify MSR design features relevant to safeguards, this paper first identifies the types and forms (physical and chemical) of nuclear material present in existing design concepts. A wide variety of chemical and isotopic compositions exist for planned fuel in salt-fueled MSR designs. Most thermal-spectrum, salt-fueled MSR designs use fluoride-based salts, whereas some newer designs operating within the fast-neutron energy spectrum use chloride salts [8]. The Molten Salt Reactor Experiment was an approximately 8 MWth salt-fueled MSR that operated at Oak Ridge National Laboratory from 1965 to 1969 using a $^{7}Li_2BeF_4$ coolant salt with first ^{235}U, then ^{233}U, over its operating lifetime [9].

For MSRs without fuel fabrication processes in the facility, fuel salt will arrive at MSR facilities as a solid, within containers that can be item counted, because the melting temperatures of the salts are significantly higher than ambient temperature at atmospheric pressure (i.e., over 500 C) [10]. Some salt-fueled MSR designs may include fuel fabrication (e.g., combination of salt and nuclear material) at the MSR facility. In either design, fresh, unirradiated fuel will have low radioactivity levels and, for most salt-fueled MSR designs, is either low enriched uranium (LEU) fuel with less than 5 wt % ^{235}U—similar to fuel in LWRs—or high-assay LEU between 10 and 20 wt % ^{235}U. Some salt-fueled MSR designs plan to use Pu or spent Canada Deuterium Uranium (CANDU) fuel that has been chemically processed to remove fission products [3]. These fuels would be more radioactive than LWR and MSR fresh fuel, though not as radioactive as irradiated fuel with fission products. Fresh fuel salt will be melted into a liquid before it enters or—depending on the design—as it enters the reactor environment. Once the salt is drained from the system, for maintenance or end-of-life, the salt will solidify. The accessibility of the fuel salt will decrease once the fuel is in the operating reactor. The reactor environment is high in temperature and in radioactivity. The containment will likely be less robust for salt-fueled MSRs as compared to LWRs because of significantly lower operating pressures (i.e., just over atmosphere due to cover gas) [11]. Some designs include plans to place the reactor itself underground or even under water on the seabed. Another significant difference from current operating reactors is that makeup salt (i.e., fissile material in fuel) will be added while the reactor is operational to maintain criticality/power.

Quantifying nuclear material in irradiated fuel salt during operation will be especially challenging because the inventories of nuclear material types (e.g., ^{235}U, ^{239}Pu) will change over time due to transmutation. All fuel salt will contain fertile nuclear material (i.e., ^{238}U and/or ^{232}Th). When irradiated in the reactor, these isotopes will breed fissile material that will be mixed within the fuel salt with fission products, actinides produced through neutron capture, fluoride/chloride salt, activation products, and radioactive progeny. Many nondestructive assay (NDA) technologies used in safeguards detect the fission products, activation products, and transuranic actinides to verify the irradiation history of spent fuel assemblies (e.g., the Passive Gamma Emission Tomography system used in international safeguards) [12]. However, in salt-fueled MSRs, the highly radioactive environment and the changing concentrations caused by removal of gaseous fission products, through the cover gas and—if included—an off-gas system, introduces potential challenges to gamma and neutron detection for quantifying nuclear material in the bulk fuel salt. MSR salt is homogeneous, so a small sample taken from the system represents the isotopic and chemical makeup of all fuel salt at the time the sample was removed. Lastly, directing salt through online detection tools is an opportunity for online measurements that is not available in fixed-fuel systems.

PROPOSED GENERAL MC&A APPROACHES

There are two general categories of MC&A approaches that facility MC&A staff (for domestic safeguards) or the IAEA (for international safeguards) might apply for salt-fueled MSRs: (1) a "black box" material balance approach, in which the nuclear material content within all inputs and outputs of the reactor is thoroughly measured, coupled with robust containment and surveillance on the reactor itself, or (2) a process monitoring approach that aims to quantify the nuclear material throughout all facility processes, including within the reactor. Table 1 describes both approaches.

Table 1. Description of two general approaches to MC&A at MSRs [3]

General MC&A Approach	MC&A Aspects	Benefits	Challenges	Considerations for Safeguards Technologies
Black box material balance of inputs and outputs to the reactor	• Thorough accountancy of inputs (i.e., fresh and makeup fuel) and outputs (i.e., waste streams, irradiated nuclear fuel) without a focus on the material in the reactor itself • Robust containment and surveillance within the facility and material balance areas	• Most similar to existing LWR MC&A approaches • Could likely be done with minor, if any, modifications to existing technologies and techniques	• Would be difficult to apply to liquid-fueled designs with online chemical processing with separation of fissile material • May have large material unaccounted for (MUF)/inventory difference (ID) that might require further investigation (e.g., quantification of holdup) • Measured outputs will need to be compared with predicted values of fissile material produced in the reactor	• Might require new or modified technologies to quantify material in fresh fuel salt/concentrate, irradiated fuel salt, and/or waste streams • Reactor physics models will need to be developed, verified, and validated to produce predicted values of material quantities; some designs (e.g., those that operate in the fast spectrum) might need more accurate nuclear physics data (e.g., cross-sections) to achieve reasonable uncertainties
Material accountancy throughout the process streams	• Quantifies material throughout the process	• Leverages existing data tracked for the purposes of safety analysis and optimization of operations • Potentially more timely detection of diversion of nuclear material from process streams • Likely more accurate in determining location and quantities of nuclear material, especially holdup and other contributors to MUF/ID	• Measurements performed on salt with high temperature and radioactivity • May require integration of a sampling stream/port • Would require a revised approach by the NRC to utilize data from "process monitoring" to meet requirements in special nuclear material (SNM) categories other than Category I • May be costly to integrate technologies if measurements are not already being taken for other purposes	• Might require new or modified technologies to quantify SNM in fresh fuel salt/concentrate, irradiated fuel salt, and/or waste streams • Might require development of new online or in-process monitoring technologies • Drives a "by-design" approach to the integration of technologies and resultant data streams

Black Box Material Balance Approach

The black box MC&A approach is similar to the one used for LWRs and other facilities, where the nuclear material exists in an accountable physical item. The IAEA implements a similar methodology to this approach in enrichment facilities. It would be reasonable to extrapolate this approach to solid-fueled MSRs, in which TRISO fuel pebbles could be individually accounted for or (more likely) groups of fuel pebbles within a vessel could be accounted for before entering and after exiting the reactor. This approach could also be reasonably extrapolated to salt-fueled MSRs, in which the fuel is contained in distinct fuel assemblies.

However, the black box approach has limitations for salt-fueled MSRs. Even if nuclear material can be accurately quantified in the feed fuel (i.e., initial fresh fuel salt and makeup fuel salt added to the system) and in the irradiated salt once it is drained from the system (i.e., in tanks), this quantification will not provide enough information to meet safeguards goals unless there is information as to what quantities should be in the irradiated fuel. In LWRs, this expected value can found using well-developed, validated depletion modeling tools [13]. Additionally, because LWR assemblies are large, heavy, and highly radioactive, in most cases the goals are practically met as long as the operators or IAEA can verify that all assemblies are accounted for (e.g., using containment and surveillance), and—in the case of international safeguards—that the assemblies are radioactive and contain U and Pu relatively consistent with burnup (e.g., using Cherenkov viewing devices). However, item accounting for salt-fueled MSRs is ineffective. Salt-fueled MSR modeling tools must incorporate the dynamic aspect of fuel moving in the system, online refueling with makeup salt, and online removal of some isotopes through, for example, the off-gas systems [14]. Additionally, some MSR concepts incorporate online chemical separation for recycling of fissile material. Although efforts are currently being funded to develop these modeling tools, with no operating MSRs (research, test, or commercial), limited data are available for validation. Furthermore, the accuracy of these codes is limited by the accuracy of the nuclear data (i.e., values of nuclear physics–related properties associated with specific isotopes). Some MSR designs operate within a fast-neutron energy spectrum—as opposed to LWRs, which operate within a thermal neutron energy spectrum—and it is unknown whether the existing, more limited, and less validated nuclear data in those energy regions will have uncertainties low enough for practical purposes related to estimating nuclear material quantities in irradiated MSR fuel salt[3].

Even if such codes are developed, measurement of irradiated fuel at periodicities determined by the accessibility of the salt as it is drained from the system for maintenance is unlikely to be frequent enough to ensure timely detection of diversion of nuclear material from the salt. Commercial MSR designers and operators would benefit from few and infrequent complete salt drains from the system—potentially once every few years—to maintain a high capacity factor. Therefore, data from within the operational process stream (i.e., process monitoring) will likely need to be included in MC&A programs or IAEA safeguards approaches at salt-fueled MSR facilities [15].

Process Monitoring

Many salt-fueled MSR designs will likely incorporate the ability for facility operators to monitor operational parameters to optimize operations and measure the thermochemical and thermophysical properties of the fuel salt. This capability would ensure that the parameters remain within the established limits necessary to satisfy the reactor safety bases or to perform fundamental safety functions. For example, one approach could be to extract a small quantity of fuel salt from the reactor environment through a sampling line. Analysis of the sampled material could enable the determination of fissile material quantities and actinide concentrations in near real time. Destructive analysis (i.e., mass spectrometry) techniques and/or in situ nondestructive analysis detection systems (e.g., gamma and/or neutron, hybrid K-edge densitometry) could be used to determine the isotopic composition of the salt in the process stream(s). Much of this operational parameter data could provide relevant information to determine the material quantities and locations for MC&A purposes. This methodology is often referred to in the technical community as "process monitoring." However, NRC regulations in 10 CFR Part 74 use that terminology in a very specific context associated with the robust MC&A requirements mandatory for Category I, strategic SNM. Applying the technical approach of material accountancy throughout the process streams would provide less overall uncertainty as to the location and quantities of the SNM within the facility. Additionally, the designs would likely already incorporate measurement systems to produce these data for non-safeguards purposes. The NRC would benefit from more accurate and efficient MC&A implementation at MSRs by encouraging facility designers—especially of salt-fueled MSRs—to incorporate the methodology of material accountancy throughout the process streams (i.e., process monitoring), while not necessarily requiring all obligations associated with the term "process monitoring" used in 10 CFR Part 74 for Category I, strategic SNM. If vendors of salt-fueled MSRs, in any State, plan to construct and/or operate in Non-Nuclear Weapon States, designers will need to incorporate international safeguards considerations into their designs to accommodate the IAEA's need to independently verify quantities and locations of nuclear material. In facilities with large throughputs and the capability to separate "direct use nuclear material" (i.e., salt-fueled MSRs with online chemical separation of fissile material), the IAEA often implements online measurement systems to independently verify quantities of the material within the process streams (e.g., Japan's Rokkasho reprocessing facility, Republic of Korea's pyroprocessing facilities).

Combined Approach

It is possible that a domestic or international safeguards approach for a salt-fueled MSR could employ a black box material balance approach initially while gathering data to validate reactor physics codes that could then be used in the future to produce the expected values. In this approach, measurements would be taken on in-process material, though not necessarily used in a material balance evaluation. The goal of these measurements would be to provide additional information on quantities throughout the process, but the safeguards technical objectives (in the case of international safeguards) would be met through containment and surveillance. This combined approach could be considered a monitored operations approach. Once more data have been gathered from operating salt-fueled MSRs, and reactor physics codes have been validated, then the expected quantities of nuclear material in irradiated fuel salt could guide material balance evaluations.

It is very unlikely that data from one technology will be able to accurately quantify every relevant isotope of interest in each of these locations within the process stream. Instead, an effective MC&A program or safeguards approach is likely to use data from multiple technologies to inform a mass balance evaluation across the entire facility. In some cases, technologies might not be available to provide the relevant data within the time period and uncertainty level desired to meet either domestic safeguards requirements or IAEA technical objectives [16]. In those cases, containment and surveillance can be employed to ensure that material is controlled even if it cannot be accurately accounted for. This paper does not fully explore containment and surveillance aspects that will be included in any MC&A program or safeguards approach. To measure the quantities of each relevant isotope within the fuel salt, in the absence of a technology that directly measures each quantity within the entirety of the system, combinations of the following variables will be needed in each location identified within the nuclear material process stream: (1) mass of the salt (might be calculated using volume and density); (2) elemental composition of the salt (e.g., U percentage within the salt); and (3) for U and potentially Pu, isotopic assay of the relevant isotope within the element (e.g., U percentage that is ^{235}U/enrichment). Some existing technologies robust enough to provide accurate results in MSR environments might provide information for one or two of the variables identified herein. Although using an individual technology would not provide enough information to quantify the nuclear material within the salt, it could be coupled with information from other technologies or used with containment and surveillance in an overall MC&A plan to provide assurance that no diversion from the system occurred.

SAFEGUARDS-RELEVANT DESIGN FEATURES

Such wide variation across different salt-fueled MSR design concepts makes it impossible to generalize a domestic or international safeguards approach to meet specified objectives. Table 2 is a list of some of the design features of salt-fueled MSRs that the authors have identified as safeguards relevant. Features are organized into four main categories related to (1) fuel selection, (2) operational plans or practices, (3) physics of the reactor design, and (4) other systems included in the design. Table 2 also includes a brief description of the features and how each feature might be relevant to domestic and/or international safeguards.

CONCLUSIONS

Salt-fueled MSR concepts currently include a wide range of design features. This paper identifies many design features that are relevant to domestic and/or international safeguards. Many of these design features are necessary and intrinsic to the design itself; however, some of them could potentially be altered to incorporate more accurate characterization of the fuel for safeguards purposes. For example, fresh fuel inventory could be limited and/or stored in one secure location that promotes easier inventorying by the IAEA.

This paper also identifies general MC&A approaches that might be used in a broader safeguards approach. For designers and operators (for domestic safeguards purposes) and the IAEA (for international safeguards purposes) to develop effective and efficient safeguards approaches, further research is required to determine how effective a black box approach can be on a specific design. Approaches for some less complicated designs, with lower throughput and low power (i.e., low potential for misuse by irradiating undeclared fertile material – of concern in international safeguards), could include measurements of fresh fuel and irradiated salt drained from the system and use only containment and surveillance on the reactor itself to meet the technical objectives. Approaches for higher power, more complex salt-fueled MSRs could benefit significantly from measurements of the irradiated salt during operation. Further assessment of nuclear material quantities—based on concentrations, throughput, and neutron physics considerations—present in specific designs should be performed to determine an appropriate approach for a specific salt-fueled MSR. Any additional information that designers could share without revealing proprietary information (e.g., planned method and frequency of refueling) would help the research and development communities provide more detailed recommendations for developing effective and efficient MC&A approaches.

Table 2. Salt-fueled MSR design features and their relevance to domestic and international safeguards

Design Feature Category	**Design Feature**	**Description**	**Relevance to Domestic Safeguards**	**Relevance to International Safeguards**
Fuel	Fissile material content (type, physical/chemical form, quantity, enrichment) in fresh fuel salt	Designs include different plans for fissile material in the fresh fuel, including <5 wt % ^{235}U, 10–20 wt % ^{235}U, Th, Pu, and other actinides.	Fissile material is accountable material; LEU > 10% enriched has different regulations than LEU < 5% in US domestic safeguards.	Fissile material types, quantities, enrichments, and locations must be declared to, and independently verified by, the IAEA.
	Fertile material content (type, quantity) in fresh fuel salt	In salt-fueled breeder MSRs, fertile material will be added to salt in either the primary salt loop or a separate blanket salt loop. Burner reactor designs will also contain fertile material (e.g., ^{238}U).	Fertile material is not accountable material but is helpful in estimating/bounding the quantities of SNM that will be created through irradiation.	Fertile material types, quantities, and locations must be declared to, and can be independently verified by, the IAEA.
Operational plans/practices	Inventory of fresh fuel salt held at the facility	Facility operators will likely plan to store different amounts of fresh fuel onsite.	Inventories and containment and surveillance of this material will require MC&A resources.	Inventories and containment and surveillance of this material will require IAEA and facility operator resources.
	Method and frequency in which makeup salt is added	Designs will likely incorporate different plans for adding makeup salt.	MC&A plans will require either surveillance or measurements to verify that no SNM is diverted during refueling.	Material balance evaluations require quantification of the nuclear material added to the system during refueling. The method and frequency with which salt is added will determine what technologies are practical for the IAEA to use to monitor fuel additions.
	Frequency in which reactor components are replaced	Components within MSR designs will have to be replaced periodically (e.g., in thermal spectrum MSRs, the graphite moderator blocks will have to be replaced every few years). When reactor components are replaced, fuel salt will be drained from the system and stored in a tank(s).	MC&A plans should incorporate considerations to ensure no diversion of SNM during replacement of reactor components.	Accounting for nuclear material during maintenance will likely require significant IAEA resources. These could be opportunities for the IAEA to perform inventories of irradiated salt in tank(s).

Design Feature Category	Design Feature	Description	Relevance to Domestic Safeguards	Relevance to International Safeguards
Physics of the reactor	Power of the reactor	Some salt-fueled MSR designs are planned as significantly smaller (e.g., <300 MWth) than large, commercial LWRs (~1,000 MWth); other designs exist for larger (~1,000 MWth) salt-fueled MSRs.	The power of the reactor impacts the throughput of fissile material, which impacts the total quantity of fissile material on the site.	The power of the reactor impacts the throughput of fissile and fertile material and bounds the amount of undeclared fissile material that could be produced at the facility; this impacts IAEA resources to ensure timely detection of diversion/misuse of the facility.
	Breeding ratio (connected with neutron energy spectrum)	The breeding ratio determines how effectively the reactor is transmuting fertile material into fissile material.	The breeding ratio will impact the total quantity of fissile material within the reactor over the core lifetime.	The breeding ratio will impact the total quantities of fissile and fertile material within the reactor over the core lifetime; both must be declared to the IAEA.
	Neutron energy spectrum (connected with breeding ratio)	The neutron energy spectrum in which the reactor primarily operates impacts how much Pu is produced in U/Pu fuel-cycle reactors.	Thermal energy spectrum MSRs will typically have lower amounts of Pu in their waste; Pu waste characterization will likely be a part of a facility's MC&A plan.	Thermal energy spectrum MSRs will typically have lower amounts of Pu in their waste; IAEA frequency and intensity of inspections will be impacted by the quantities of nuclear material.
Other systems	Chemical processing of salt to separate fissile material and re-use as makeup fuel	Some salt-fueled MSR designs (especially those with high breeding ratios) plan to separate fissile material from the salt and use it as makeup feed to refuel the reactor during operation. This approach is used in some U/Pu and Th/U fuel-cycle designs.	Nuclear material in this recycle stream, where fissile material is separated, stored, and returned to the reactor as makeup fuel, may need to be accurately quantified.	IAEA might have to independently verify quantities of nuclear material within decay tanks or streams of fissile material being separated. In Th fuel cycles, this may include verification of ^{233}Pa, which some MSR concepts plan to separate.
	Off-gas system	Some designs include an off-gas system to actively filter out gaseous fission and activation products. These systems sometimes include decay tanks.	Online measurements could occur in the off-gas system to provide data relevant to MC&A.	NDA measurements of fission products and/or ratios of certain actinides are sometimes used to determine burnup or other factors that are relevant in quantifying nuclear material in irradiated fuel. Cover gas and off-gas systems will impact the quantities of these gasses.

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