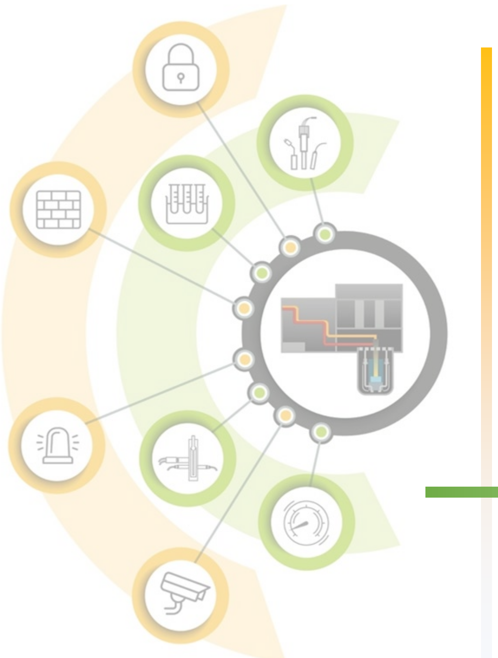

MC&A for MSRs:
FY2021 Report

Prepared for
US Department of Energy

Michael P. Dion, M. Scott Greenwood, Karen K. Hogue, Sean E. O’Brien,
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Physics, Nuclear Nonproliferation, and Nuclear Energy and Fuel Cycles Divisions
Prepared for the Advanced Reactor Safeguards Program

MC&A FOR MSRS: FY2021 REPORT

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September 2021

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CONTENTS

ACRONYMS

ARS	Advanced Reactor Safeguards
CFR	Code of Federal Regulations
DA	destructive assay
GADRAS	Gamma Detector Response and Analysis Software
HX	heat exchanger
LWR	light water reactor
MC&A	Material Control and Accountability
MSR	molten salt reactor
MSDR	Molten Salt Demonstration Reactor (MSDR)
NDA	nondestructive assay
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
SNM	special nuclear material
TID	tamper indicating device
TRANSFORM	Transient Simulation Framework Modules

1. INTRODUCTION

There is significant domestic and international interest, investment, and research and development momentum to pursue advanced nuclear reactor technologies. Molten salt reactor (MSR) concepts display the largest variability in fuel type and design features among the current advanced concepts. MSRs have been proposed with various core designs, sizes (power), and fuel cycles. Salt-fueled[1] molten salt systems represent the only advanced reactor type with fuel that is not in a solid form during operation. These "liquid-fueled" MSRs are unique from perspectives of fuel fabrication, spent irradiated fuel and waste components, licensing, and material control and accountability (MC&A) including the potential of fissile material holdup. The liquid fuel salt is the defining distinction in comparison to other advanced reactors that propose TRI-structural ISOtropic particle fuel pebbles, various coolant options (e.g., molten salts or metals, high temperature gas), or small modular alternatives using solid fuel variants including both light water reactors and non-light water reactors.

MSRs are appealing to the nuclear energy industry because of the diverse reactor characteristics they can support including various neutron energy spectra, fueling requirements, fuel cycles, and/or fuel utilization. However, because of the significant deviation and diversity of a salt-fueled system compared to traditional solid fuel light water-cooled reactors (LWRs), the history, regulatory licensing framework, modeling capabilities, and supporting engineering technology are either lacking or, in some cases, nonexistent. Therefore, the research community is actively supporting advanced MSR development on many of these fronts [1, 2, 3, 4, 5, 6, 7, 8, 9] in particular to assist MSR vendors with licensing requirements. ORNL is leading the research and development of respective MC&A approaches for salt-fueled MSRs.

This report summarizes the research performed at Oak Ridge National Laboratory (ORNL) under the US Department of Energy, Office of Nuclear Energy, Advanced Reactor Safeguards (ARS) program to investigate safeguards and security by design concepts, licensing and regulatory considerations, and dynamic system-level modeling to understand radioisotope concentrations for salt-fueled MSRs. The report builds upon the previous research and literature [10, 11, 12, 13, 14, 15, 16], identifies the MC&A challenges inherent to a salt-fueled MSR, reviews current regulatory frameworks for LWRs and their applicability towards salt-fueled MSRs, summarizes the status and progress of an MSR dynamic modeling tool, and discusses a prospective MC&A approach based on the Molten Salt Demonstration Reactor (MSDR) model.

The report is organized leading to the development of a generalized MC&A approach for the MSDR model. The following outline provides a brief introduction to the content of each section.

- **Section 1.1** introduces the concept of a salt-fueled MSR and discusses the features that enable such wide design variety. The design features, a description, and the safeguards relevance of those features are compiled in Table 1.
- **Section 2** provides a general introduction to the NRC regulatory and licensing framework for power reactors. The NRC categorization of special nuclear material (SNM) is introduced. Finally, discussions with the MC&A group from the NRC are summarized which provide guidance and feedback during the development of an MC&A for salt-fueled MSRs.
- **Section 3** is a summary of the progress and modifications made to TRANSFORM used for modeling these dynamic nuclear systems. Significant effort has been made to supplement

[1] In this report the term 'salt-fueled' is chosen to define an MSR in which the fuel (the fissile and fertile content) is in liquid form dissolved in the molten salt contained in the primary fuel loop. This is analogous to other terminology like 'liquid-fueled', etc. This is a major distinction in comparison to salt-cooled and metal-cooled advanced reactor designs that are solid-fueled.

TRANSFORM with capabilities needed to understand radionuclide inventories which can be used to inform measurement locations and frequency. In addition, the nuclear data set has been expanded and this has been verified to be in good agreement with similar simulations using SCALE/ORIGEN.

- **Section 4** contains specific information about the MSDR model. An analysis of a 5-y closed fuel cycle (no fueling or removal of fuel) with 1 day resolution is presented. The concentrations of SNM in the fuel salt are provided. The radionuclide inventories are used to model the dose from several geometries and demonstrate the high radioactivity of fresh fuel salt. Also, because the concentration of SNM in the fuel salt is relatively low, it demands large amounts of highly radioactive molten salt removal for any significant SNM quantities.
- **Section 5** presents a prospective MC&A plan for the MSDR. The generalized approach has elements that are likely applicable to many if not all MSRs. The reactor facility is divided into three Control Areas:
    1. fresh fuel salt storage,
    2. fuel salt in reactor containment, and
    3. irradiated fuel salt and waste storage

  For each control area, the MC&A objectives, the SNM in the control area, the physical form of SNM, the measurement environment, and the MC&A Elements are extensively defined. This information is compiled in Table 3.

1.1 PROPERTIES OF A SALT-FUELED MSR

The current NRC regulations and licensing requirements have been developed to support the current fleet of commercial LWRs. Applying the existing requirements to advanced reactor concepts may introduce licensing challenges. Salt-fueled MSRs may pose the most complicated MC&A approach considering the advanced reactor designs due to their extraordinary properties and the distinct difference of their fuel cycle (e.g., online refueling) and fuel form. Properties that make salt-fueled MSRs appealing from an operational perspective -e.g., ease of fueling, online refueling, etc. - introduce challenges in accurately accounting for and controlling fissile material throughout the process stream. Understanding how to apply current domestic safeguards requirements to salt-fueled MSRs presents a unique opportunity for research into MC&A approaches that will ensure special nuclear material is not stolen or otherwise diverted.

MSRs are advanced nuclear energy systems that can utilize various fuels and fuel cycles. The variations of salt-fueled MSR designs are made through combinations of the fuel, the neutron spectrum, and operational considerations such as chemical processing/separations and refueling methods and frequency. The parameters that makeup an MSR design that have safeguards relevance are captured in Figure 1. The design-specific fuel, salt compound and the neutron energy spectrum, together with specific parameters related to ‘Operational & Support Systems,’ create the context for considerations of a domestic safeguards approach on a per-design basis. It is important to understand the variability of salt-fueled MSR designs because it impacts (i) how regulations might apply to salt-fueled MSRs, and (ii) development of effective design-specific MC&A plans to meet NRC domestic safeguards requirements. For this report, a model based on the MSDR [18] was used as an exemplar to determine the first steps in the development of a prospective MC&A plan for a plausible salt-fueled MSR design.

The existing domestic nuclear regulatory framework was developed and revised over the years to support the licensing of LWRs and supporting fuel cycle facilities of a once-through, uranium/plutonium fuel cycle. The most significant difference comparing a salt-fueled MSR to an LWR is the physical state of the fuel itself. LWR fuel bundles are composed of sintered pellets arranged in fuel rods, and numerous fuel rods are combined to complete the fuel bundle assembly. The fuel assembly is loaded into the reactor core and removed as a unit as necessary per the reactor design. The supporting technology for LWR fuel development is a mature process. LWR fuel must be qualified, and the NRC regulations define a fuel

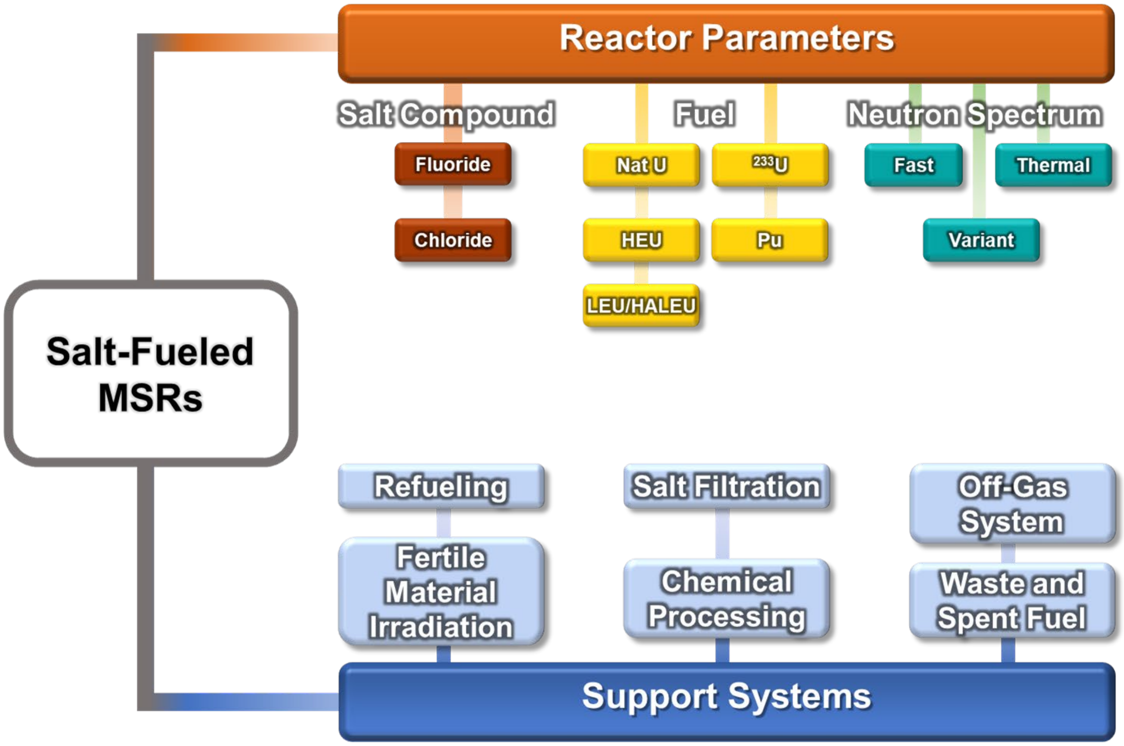

Figure 1. This figure captures some of the numerous options that salt-fueled MSR designs can contain. The interface and combination between the fuel salt selection and the operational & support systems create the potential for various designs. These details of fuel, spectrum, off-gas, etc. will potentially impact the safeguards approach for that reactor design.

system for LWRs as "The fuel system is defined as consisting of guide tubes or thimbles; fuel rods with fuel pellets, insulator pellets, cladding, springs, end closures, fill gas, …" [19] Conversely, a salt-fueled MSR contains none of these characteristics; the fuel system is comprised of fuel salt bound solely by the primary fuel loop. The fission products are contained by the boundaries defined by the primary fuel circuit itself not in the matrix of fuel pellets and then further restricted in a fuel rod. The fission products will either be (intentionally) removed by sparging and/or filtering or remain soluble in the molten salt. Therefore, fuel qualification and various supporting documentation for these systems will have to be revised. For more information on the proposed fuel qualification for salt-fueled MSRs, the reader is directed to Holcomb, et al. [7].

It should not be overlooked that a salt-fueled system has appealing safety and economic aspects from an operational perspective to a prospective vendor. These include:

- Online refueling reduces operational downtime,
- Fresh fuel salt synthesis (at the reactor site) removes the need for (custom) fuel fabrication facilities,
- Low pressure operation reduces cost of structural materials and containment,
    - The molten salt is the heat transfer coolant. There is no water or pressurized steam greatly reducing the risk of explosion [20].
- The fuel material is already molten so there is no risk of 'meltdown' and, if necessary, the fuel-salt can be quickly drained from the system and placed in a non-favorable geometry (i.e., drain tank).
    - Loss of electrical power to a plant does not affect this safety feature.

A significant amount of research and discussion has been made on the definition of ‘inherent’ safety features. For a more in-depth discussion of the topics above and others, the reader is directed to the references [20, 21].

The specific system design features of an MSR facility are influential in developing an effective safeguards approach including an MC&A plan. For simplification and consideration of the MSR fuel discussion, the components or process steps that could be present in an MSR facility and are most influential to an MC&A approach are categorized in Table 1 [13].

Table 1. MSR design features, description, and the relevance to a domestic safeguards plan.

	Design Feature	Description	Relevance to Domestic Safeguards
Fuel	Fissile material (content) in fresh fuel salt: type, quantity, physical and chemical form, enrichment	MSR fuels include LEU (<20 wt % ^{235}U including HALEU 5–20 wt % ^{235}U), HEU (>20 wt % ^{235}U), ^{233}U, Pu.	Most fissile material is defined as SNM and is accountable. Regulations vary based on U enrichment and quantities of material.
	Fertile material (content) in fresh fuel salt: type, quantity, physical and chemical form	Fertile material is added to the primary fuel loop or blanket salt to produce fissile material. Ex: $^{238}U \rightarrow ^{239}Pu$, $^{232}Th \rightarrow ^{233}U$.	Fertile material is not accountable but is useful in predictive calculations for accountable fissile material production.
Operations	Inventory of fresh fuel salt	Amount and physical and chemical form of fresh fuel stored onsite.	MC&A approaches will likely require inventories and physical surveillance of the material.
	Refueling details, frequency of refueling, and adding makeup fuel salt.	Addition of fuel salt is necessary. The frequency and how fuel salt will be added are dependent on design.	Surveillance during refueling will provide confidence of no diversion. Measurements will provide verification and quantification of declared addition and confidence that there was no diversion of material.
	Planned and unplanned maintenance	Replacement of components will be necessary and unplanned issues may arise, resulting in complete drains and flushes of the fuel salt.	Surveillance and documented records during maintenance periods will increase confidence that there is no diversion. Measurements can ensure all accountable SNM in fuel salt is accounted for.
Reactor Conditions	Power, fuel burnup	Variety of sizes are planned to range from <300 MW_{th} to 1000 MW_{th}. Inventory of fissile material at startup.	The power coupled with design features influence quantities and throughput of fissile material. Operational time, and power need to be documented.
	Breeding ratio	A ‘breeder’ has a ratio >1. It is the ratio of production of fissile isotopes divided by the rate of destruction of fissile isotopes during operation.	Breeding fissile material impacts overall fissile material production and burnup. Inventories of SNM may be found with predictive codes and compared to records.
	Neutron energy spectrum	The proposed energy spectrum dictates salt type, refueling needs, fissile material production, etc.	The proposed energy spectrum impacts refueling plans, and fissile material production.
Supporting Systems	Off-Gas	Insoluble gaseous fission products (i.e., ^{135}Xe) are removed for reactivity control in some designs.	Removal of these elements from the fuel salt are (more) favorable for radiometric measurements. Trace fuel salt containing SNM will be entrained in this system so MC&A plans may need to consider the SNM located here.
	Chemical processing and separations of fuel salt	Some designs plan to separate the fissile material generated from fertile material.	Additional MC&A considerations may be needed if fissile material is separated, stored, and returned to the reactor as makeup fuel, or outside of the primary fuel loop the SNM will need to be accurately quantified.

This report presents guidelines and considerations for an MC&A approach by incorporating feedback from NRC MC&A staff and considers current MC&A requirements and regulatory framework for salt-fueled MSRs. The majority of the challenges of implementing MC&A are due to the nature of the molten fuel-salt and how an MC&A plan will need to incorporate both item accounting methods (i.e., counting containers of fresh fuel salt) and bulk accounting methods, where the fissile (and fertile) materials are dynamic in abundance (concentration) and location (mobile in the system) in the primary fuel circuit. Online fueling and removal of fuel-salt from the system are also concerns that need to be considered. The objective is to develop documentation that identifies the systems and practices that require recognition during the licensing application stage with the NRC. A document provided to a vendor during the licensing stage can alleviate barriers, reduce license review stages, and identify shortcomings of the respective vendors application. This paper presents a generalized MC&A approach for salt-fueled MSRs that can become a starting point for any vendor in evaluation of their specific design. This prospective plan is a baseline for the vast array of design options and the supporting systems (e.g., off-gas removal, online processing, salt filtration, decay tanks), as shown in Figure 1.

2. DOMESTIC REGULATIONS AND LICENSING

The NRC issues domestic licensing to all LWR nuclear power plants (NPPs), other processing facilities (e.g., enrichment and fuel fabrication facilities), and parties who possess SNM[2] [22]. SNM is defined by the NRC as "plutonium, uranium-233, uranium enriched in the isotope U-233 or in the isotope U-235, and any other material which the Commission, pursuant to the provisions of section 51 of the Atomic Energy Act of 1954, as amended, determines to be special nuclear material, but does not include source material" [23]. Because MSRs are designed to produce electrical power through nuclear fission as a public utility, the US NRC Regulatory Guide 1.206, "Application for Nuclear Power Plants", is applicable as a roadmap of the rules, regulations, and other documents necessary in an NPP licensing application [19]. From a licensing perspective, the NRC's regulatory framework e.g., Title 10 Code of Federal Regulations (CFR) Part 50 [22], and Regulatory Guide 1.206 - Section C.I.4.2 [19] describe information that must be provided to the NRC by the applicant pertaining to the reactor components, site information, and the fuel itself. For LWR NPPs, licensing applications are submitted to the NRC under 10 CFR Part 50 – Domestic Licensing of Production and Utilization Facilities or Part 52 - Licenses, Certifications, and Approvals for Nuclear Power Plants. NPPs can be categorized as a utilization facility: "*Any nuclear reactor other than one designed or used primarily for the formation of plutonium or U–233*" and many are licensed in this way. Categorizing a power reactor as a utilization facility has direct implications to the safeguards and MC&A requirements. Because of exemptions and other regulatory guidelines developed, LWR NPPs are not required to have an MC&A program (see further discussion below).

The objective of an MC&A program at a facility is to ensure that the special nuclear material (SNM) at a facility is secure, documented, and protected. Furthermore, if there is loss or diversion of material, the MC&A elements put in place should identify this change to allow for timely recovery of the material. MC&A is one component of a safeguards program to "ensure that SNM within a licensed facility is not stolen or otherwise diverted from the facility[3]." The terminology applicable for developing a safeguards program and MC&A details is contained in Parts 70 - Domestic Licensing of Special Nuclear Material, 73 - Physical Protection of Plants and Materials, and 74 - Material Control and Accounting of Special Nuclear Material.

SNM is categorized by the NRC based on the risk of that material being used directly or to produce nuclear material for a fissile explosive. The three NRC-defined safeguard categories are:

1. *Strategic SNM (SSNM)*: This is defined as ^{233}U, Pu, and ^{235}U enriched to 20 wt% or more.

[2] https://www.nrc.gov/about-nrc/regulatory/licensing.html
[3] https://www.nrc.gov/materials/fuel-cycle-fac/nuclear-mat-ctrl-acctng.html

- Category I: SSNM in any combination of quantities of:
  - ≥2 kg Pu,
  - ≥5 kg ^{235}U (in material enriched to 20 wt% or more),
  - ≥2 kg ^{233}U, or
  - ≥5 kg in any combination of the *formula quantity* = (g contained ^{235}U) + 2.5×(g ^{233}U + g Pu)

2. *Moderate Strategic Significance*:
   - Less than a formula quantity of SSNM but more than:
     - 1 kg of ^{235}U enriched to ≥20 wt%,
     - more than 0.5 kg of ^{233}U or Pu, or
     - more than 1 kg = (g contained ^{235}U) + 2×(g ^{233}U + g Pu)
   - ≥10 kg ^{235}U enriched to ≥10 wt% but less than 20 wt%
3. *Low Strategic Significance*:
   - Less than an amount of moderate strategic significant but
     - >15 g ^{235}U enriched to ≥20 wt%,
     - 15 g of ^{233}U,
     - 15 g of Pu,
     - or the combination of 15 g = (g contained ^{235}U) + (g Pu) + (g ^{233}U)
   - <10 kg but >1 kg of ^{235}U enriched to ≥10 wt% but <20 wt%
   - ≥10 kg ^{235}U enriched greater than natural but <10 wt%

Physical protection requirements are influenced by the category of SNM that stored and processed at a facility. While these categories are explicitly defined, the NRC does have the ability to adjust the category of SNM, and thereby of the facility, with justification. For example, the NRC might consider a facility a Category II facility even if it contains greater than 2 kg of Pu if the Pu is proven to be inaccessible or have other impediments to diversion (i.e., size, weight or radiation dose and exposure). The NRC requires license applications for possession of SNM in quantities exceeding one effective kilogram[4] to include "a full description of the applicant's program for control and accounting of such special nuclear material" to demonstrate how compliance with specific requirements in CFR Title 10, Part 74 will be met[5]. This MC&A plan is submitted in license applications in the form of a Fundamental Nuclear Material Control plan [24]. The exceptions to this requirement are: (i) applications for use of the SNM as sealed sources, (ii) applications for uses involved in the operation of a nuclear reactor licensed pursuant to Part 50 (or, in practice, Part 52) of CFR Title 10, and (iii) applications for uses involved in a waste disposal operation. Currently, commercial NPP applicants in the US (i.e., LWR designs) fall under exception (ii) and therefore, do not have to submit an MC&A plan as a part of their licensing application.

The categorization and type of SNM at a facility influences the MC&A requirements pertaining to loss diversion and measurement controls. The MC&A requirements are summarized in 10 CFR Part 74 [24]. Generally, the following controls are provided:

- Low Strategic Significance (Subpart C) requirements are designed for licensees who possess more than one effective kilogram of low strategic significance SNM. There are special guidelines for uranium enrichment facilities who produce this material.
  - Three objectives: (1) Confirm the presence of special nuclear material; (2) Resolve indications of missing material; and (3) Aid in the investigation and recovery of missing material.

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[4] An effective kilogram of SNM is defined as (i) Pu or ^{233}U: their weight in kilograms; (ii) U with an enrichment in the isotope ^{235}U of 1 % and above: its element weight in kilograms multiplied by the square of its enrichment expressed as a decimal weight fraction; and (iii) U with an enrichment in the isotope ^{235}U below 1%: its element weight in kilograms multiplied by 0.0001 [33].

[5] https://www.nrc.gov/reading-rm/doc-collections/cfr/part070/full-text.html

- Moderate Strategic Significance (Subpart D) requirements are designed for licensees that possess SNM of moderate strategic significance or SNM in a quantity exceeding one effective kilogram of SSNM for various facilities (e.g., irradiate fuel reprocessing, waste disposal).
  - Requires internal control, inventory, and records of SNM. Further, an *item control program* must be maintained to:
    - "Assures that SNM items are stored and handled, or subsequently measured, in a manner such that unauthorized removal of 200 grams or more of plutonium or uranium-233 or 300 grams or more of uranium-235, as one or more whole items and/or as SNM removed from containers, will be detected."
- SSNM (Subpart E) are generally, the highest MC&A requirements.
  - Process monitoring for internal transfers, storage, and processing of SNM. Detection capabilities are characterized but exceptions are provided for various circumstances.
  - Item monitoring, alarm resolution, quality assurance requirements

It is not currently defined, by precedent or current regulations, how a salt-fueled MSR will be licensed. This report includes considerations for the aspects of an MC&A plan that the authors feel may be required by the NRC for the future licensing of a salt-fueled MSR. This report uses a model of the MSDR to explore SNM inventories throughout the process stream within the reactor containment and as a representative salt-fueled MSR design concept to develop a prospective MC&A approach.

2.1 MC&A DISCUSSIONS WITH THE NRC

The ARS national technical director, project staff at ORNL, and the NRC's MC&A team held two meetings, one on February 4, 2021 and the second on August 16, 2021. The objectives of the meetings were to (i) better understand the NRC's current position on MSR MC&A regulations, (ii) engage with the NRC's MC&A staff to identify specific MC&A challenges and potential approaches, and (iii) ensure that the ORNL (and ARS') research efforts are on track to meet the identified challenges of MC&A requirements during the licensing stage of potential salt-fueled MSR vendors.

At the February meeting, ORNL presented a technical briefing on MC&A Considerations for Molten Salt Reactors, and covered MSR design concepts, how they differ from LWRs related to MC&A, general MC&A approaches that might be applicable, and key high-level design features across MSR concepts that will be relevant for domestic safeguards. The ARS project team identified the following high-level design features as especially relevant to domestic NRC safeguards: (i) type of SNM, (ii) physical and chemical form of SNM (including the radioactivity of material in which the SNM is located and the concentration of SNM within the material), (iii) quantities of SNM, and (iv) accessibility of SNM.

In the August meeting, the NRC provided feedback on an ORNL report titled Domestic Safeguards Material Control and Accountancy Considerations for Molten Salt Reactors [6]. The NRC staff noted that it is very helpful to have reports that articulate the breadth of design variations in a class of advanced reactors and appreciated the inclusion of proposed challenges and potential solutions for MC&A elements and challenges for salt-fueled MSRs. ORNL provided an update on the progress of this work, and the NRC and project team discussed specific implementation challenges, including (i) NRC input on what design features are appropriate for a representative salt-fueled MSR to inform modeling decisions, (ii) how dynamic physical inventories might be performed, and (iii) whether containment and surveillance, coupled with measurements of fuel salt outside of containment, might be enough to meet NRC objectives for MC&A of salt-fueled MSRs. At this phase, these challenges are ongoing discussions and there is no definitive guidance as to explicitly what might be required by the NRC and how vendors might best meet NRC safeguards objectives. The NRC welcomed frequent, recurring meetings in FY22.

Engagement with the NRC has helped the ARS project team understand the status of NRC regulations and how applicants for salt-fueled MSR facilities might fit into those, as well as establishing an open line of communication where the NRC MC&A team can provide feedback on ongoing research activities related to establishing potential MC&A approaches. The NRC is not pursuing any new rule-makings at this time that would, for example, establish new regulations for specific types of facilities, like reprocessing, or, by extension, salt-fueled MSRs. Therefore, at this time, any license applications would have to work within the existing regulations and provide justifications for any potential exemptions for these new facilities.

3. DYNAMIC MODELING, DATA PROCESSING, SPECTRAL SIMULATIONS

An MSR modeling tool, TRANSFORM, has been developed at ORNL to address the dynamic nature of MSRs [4]. TRANSFORM tracks individual isotopes in the MSR system as trace substances. It allows components, like off-gas systems and decay tanks, to be integrated into the model to increase simulation fidelity from a system-level perspective. Previous research developed scripts and code to extract and parse the TRANSFORM output and perform subsequent modeling on the data structure [12]. TRANSFORM and the processing of the fission product source terms have been used to investigate the utility of online gamma-ray spectroscopy on components in the MSDR model [12, 14, 15].

3.1 MSDR DEFINITION IN TRANSFORM

The MSDR model in TRANSFORM is based on the concept development of Bettis et al. [18] with some modifications [25]. The model is a 550-megawatt thermal (MW_{th}) graphite-moderated, thermal-spectrum, loop-style MSR. Figure 2 contains the major components of the two types of MSRs: an integral (left) and loop style (right) MSR. The major difference between the two designs is the location of the primary heat exchanger (HX). Since the primary HX is located inside the reactor core boundary in the integral type, the fuel salt remains bound by this containment. Whereas, in the loop style design, the fuel salt is contained in a primary fuel loop flowing from the reactor core to the primary HX.

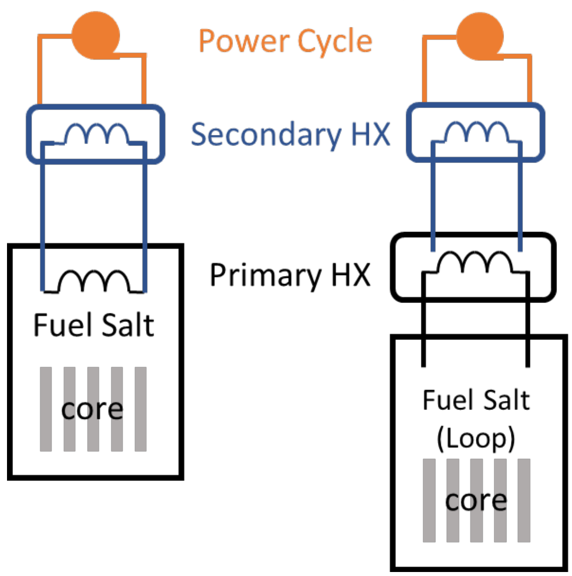

Figure 2. Depiction of the major components of an MSR and the distinction of an integral (left) and a loop style (right) design. Compare the placement of the primary heat exchanger and note the containment of the primary fuel salt in the integral graphic.

The MSDR TRANSFORM model in this research was based on a U/Pu fuel cycle but any fuel cycle defined by the user can be studied. The main components of the MSDR are the primary fuel loop, primary coolant loop, off-gas system and drain tank, and the decay heat removal system. Early MSDR TRANSFORM simulations were developed with a modified point-kinetics model which tracked a subset of fission products and precursor groups. In total, 91 isotopes (stable and radioactive) were tracked throughout that early model. This approach was sufficient to gain an understanding of reactivity feedback and functionality. However, for the investigation of using fission product concentrations for online gamma spectroscopy, it lacked the necessary short-lived fission product inventories. In addition, it did not include neutron interactions that cause transmutation of actinides in the fuel salt, and further, tracking of the actinides was not included. Therefore, TRANSFORM was upgraded to integrate nuclear data from the SCALE software package and update its tracking kinetics [16].

3.2 TRANSFORM MODIFICATIONS FOR RADIONUCLIDE CONCENTRATIONS

In previous work [4], a modified point-kinetics model was developed and implemented, which allowed for tracking of delayed neutron precursor groups and user defined isotope sets. That work was revisited in stages and modified, as noted previously, to provide a comprehensive treatment of a more exhaustive list of fission products, fertile and fissile isotopes, and subsequent transmutation through neutron reactions. The revised model leverages the nuclear data from SCALE and performs all parent-daughter calculations (both decay and fission) within TRANSFORM. This capability was demonstrated and summarized in more depth in [16]. That current research and publication presented fixed and non-fixed fuel scenarios with comparisons to ORIGEN where possible for verification. A direct comparison, especially for the non-fixed fuel model, between TRANSFORM and SCALE/ORIGEN is not trivial because of how the two codes approach computational fluid flow (Eulerian versus Lagrangian, respectively). Therefore, a transformation was applied to allow an evaluation of the simulation results; specifically, for mass chain 135. A result of this analysis is shown for ^{135}Xe in Figure 3. The modification and enhancement of fission products and other nuclides tracked in TRANSFORM have been incorporated into the MSDR model (termed the modified TRANSFORM MSDR) and are discussed in this report. Forty-two actinides are tracked and populated through various neutron interactions including isotopes of Th, Pa, U, Np, Pu, Am, and Cm. In total 150 fission products (both stable and radioactive), salt components and reaction products (e.g., isotopes of Li, ^{9}Be, ^{6}He, ^{4}He, ^{3}H), and six neutron precursor groups are also tracked as trace substances.

3.3 TRANSFORM DATA PROCESSING

A Python interface was developed to extract and process the TRANSFORM (i.e., Modelica) output files (*.mat) generated during the MSDR simulation [26]. This script enables access to all the parameters of the simulation, including temperatures, flow rates, and concentrations of the trace substances. This is used to generate the time-dependent isotope concentrations within the MSDR model. The script produces an output file for subsequent processing for detector response modeling in Geant4 [27] and the Gamma Detector Response and Analysis Software (GADRAS) [28].

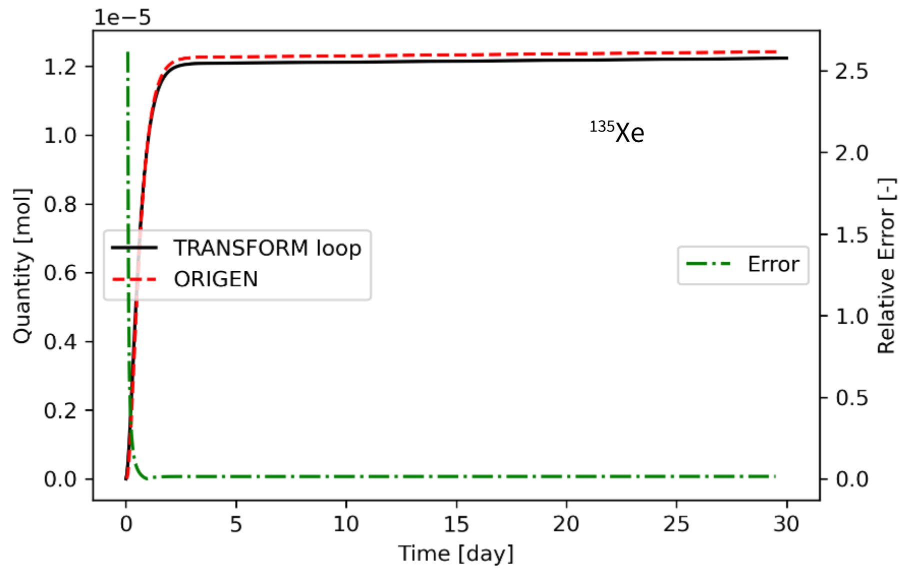

Figure 3. Non-fixed fuel comparison between TRANSFORM and ORIGEN with the modified TRANSFORM approach for ^{135}Xe.

For timely processing and production of detector response simulations, the GADRAS application program interface, included in all releases of the software, was implemented and used as a console and graphical user interface application. Then a detector and simulated measurement parameters (e.g., dwell time, distance to model, energy calibration) were defined. The graphical user interface application also contains a tool that tracks gamma-ray photopeaks specific to fission product radionuclides from the calculated synthetic spectra. The results of modeling the gamma-ray spectra and tracking of photopeaks were described in previous results [12] [15].

4. RADIONUCLIDE TRENDS OF THE MODIFIED TRANSFORM MSDR

The results presented in this section are from the modified TRANSFORM MSDR model where actinide neutron transmutation interactions are included, the actinides are tracked, and there is an extended fission product library. Because the inventories (production and depletion) of isotopes will be changing frequently it is important to understand these dynamic trends. Quantifying the SNM in the reactor system using the TRANSFORM model can provide insight into key measurement points and optimal frequency of those measurements. The MC&A approach may require measurements and observations during operation to confirm the amount and location of SNM while in reactor containment.

A simplified drawing of the main components of the MSDR model is given in Figure 4. Isotopic inventories are available at the locations marked with a checkbox. These locations are the pipe segments of the primary fuel loop, the primary HX, the reactor core, the pump bowl, and the off-gas system.

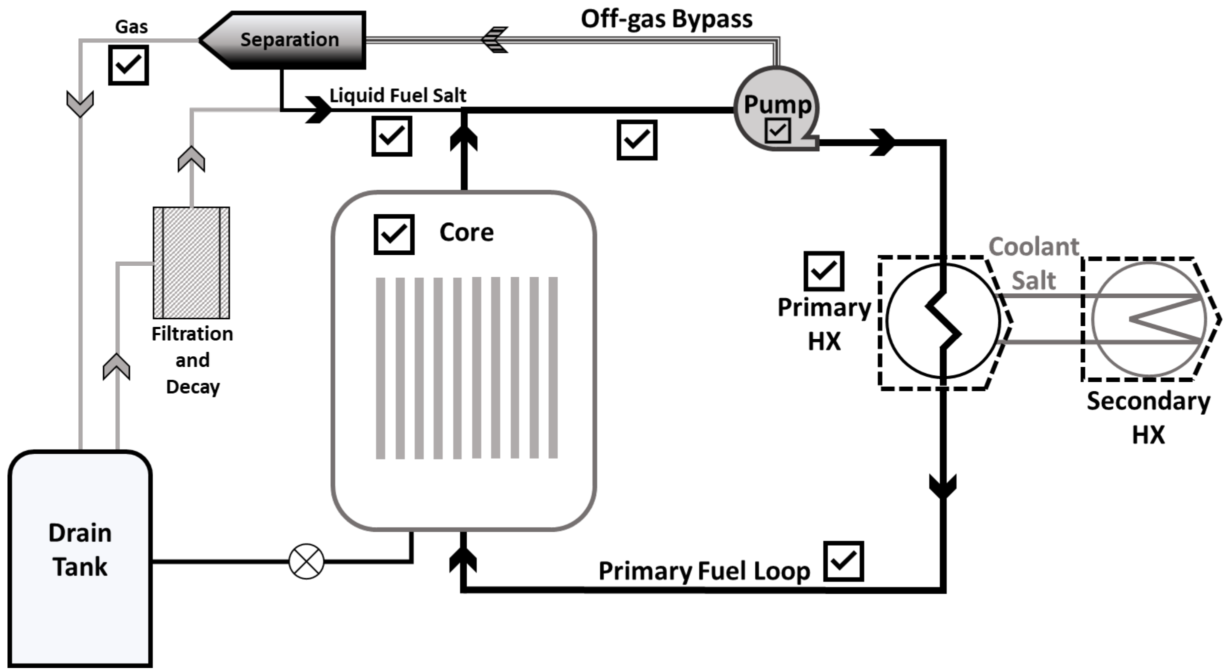

Figure 4. Simplified schematic of the MSDR TRANSFORM model. Radionuclide and fuel salt masses are available at the checkbox locations.

The modified TRANSFORM MSDR model is a simulation using the U/Pu fuel cycle. The fissile component is 5 wt % ^{235}U (LEU) in a LiF-BeF_2 67-33 mol % (FLiBe) salt with a constant density of 3.353 g/cm^3. The fuel is UF_4 with a 1 mol % composition in the fuel salt. The simulation was run for five years with one day resolution. The fuel system was closed, meaning there was no fuel addition or removal, but the off-gas system was included in the model which removes insoluble fission products (noble gases and metals and their progeny) and some fuel salt. The off-gas system used a helium carrier gas to sparge insoluble fission products from the primary fuel loop. Fuel salt is also removed in the off-gas system at a 2:1 salt-to-gas volume ratio. The fuel salt and gas mixture removed from the primary fuel loop was separated and the fuel salt is returned to the primary fuel loop. The gases are moved to the drain tank for decay and then filtered before returning to the pump. The model was defined such that the liquid and gas components are available as separate data variables.

4.1 ACTINIDE TRENDS AND PROPERTIES OF IRRADIATED SALT

As discussed in Section 2, the type and amount of SNM impacts the MC&A requirements to safeguard the material. The modified TRANSFORM MSDR model provides insight into Pu production and depletion of the U fuel which can be used to categorize the material and develop an appropriate MC&A approach. For discussion, the actinides in the fuel cell and the off-gas liquid stream are considered. This section considers the SNM within the fuel salt after entering the reactor containment. Future work should consider the diversion of SNM in fresh and makeup fuel salt at MSR facilities.

Because of the long half-lives of the actinides in the model, there is no significant difference in their behavior at the various locations accessible in the model. However, the total mass of each actinide at these locations is distinct. In addition, the actinide to salt concentration is not altered by any chemical processes including the off-gas system since it only separates the gaseous fission products (e.g., Xe and Kr) from the liquid salt (actinides and soluble fission products are contained in the liquid). The actinide to salt concentration changes over time due to transmutation and depletion based on various neutron interactions.

4.1.1 U and Pu Trends

The change in total mass of the U and Pu isotopes in the reactor fuel cell (core) is shown in Figure 5. The trends demonstrate (expected) production of Pu isotopes (e.g., ^{239}Pu) and a reduction in the total ^{235}U content as expected for a closed fuel cycle (i.e., no makeup fuel salt addition or salt removal). The isotopes of Pu result from various neutron capture reactions and radioactive decay starting from the U isotopes of the fuel. The mass of ^{238}U in Figure 5 appears unchanged over time because of the log scale axis. However, there is a total reduction of about 8% total mass of ^{238}U over the 5-y simulation.

The total amounts of Pu and ^{235}U in the reactor's fuel cell are 58 kg and 186 kg, respectively. The Pu reaches a maximum value at the end of cycle while, the ^{235}U is at a maximum at the beginning of cycle. These total amounts are high enough to be considered SSNM in a traditional understanding of facility categorization; however, there are other factors that should be considered when evaluating the type of SNM at MSRs.

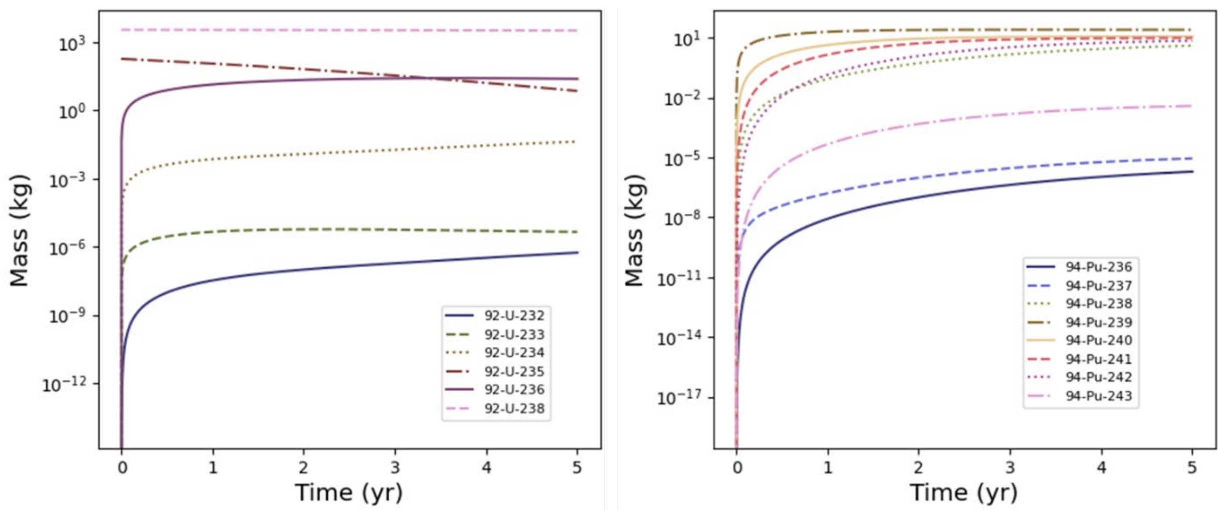

Figure 5. The evolution of U and Pu isotopes in the reactor fuel cell of the modified TRANSFORM MSDR model. For reference, the pipe contains ≈51,620 kg of molten salt. The Pu content builds rapidly after startup reaching 1 kg in two weeks. For comparison, there is 3600 kg of ^{238}U and 189 kg of ^{235}U in the fuel cell at startup.

4.1.2 Concentrations of SNM in Irradiated Fuel Salt

While the total amount of SNM contained in the MSDR reactor model is over the limits to be characterized as SSNM, the SNM is dissolved in the irradiated liquid salt. Therefore, it is only practical to consider removal of unseparated fuel salt as a credible diversion pathway. Significant infrastructure would be needed to chemically separate either U or Pu from the irradiated fuel salt matrix and is not further considered for domestic safeguards purposes. Figure 6 is a plot of the mass concentrations (SNM to total fuel salt mass in µg/g) of ^{235}U, $^{239,\,240}$Pu, and total Pu for the 5-y simulation. In addition, the ^{235}U enrichment (wt %) is provided (red curve corresponding to the secondary y-axis). As noted, because the model does not include refueling, the ^{235}U mass fraction steadily decreases.

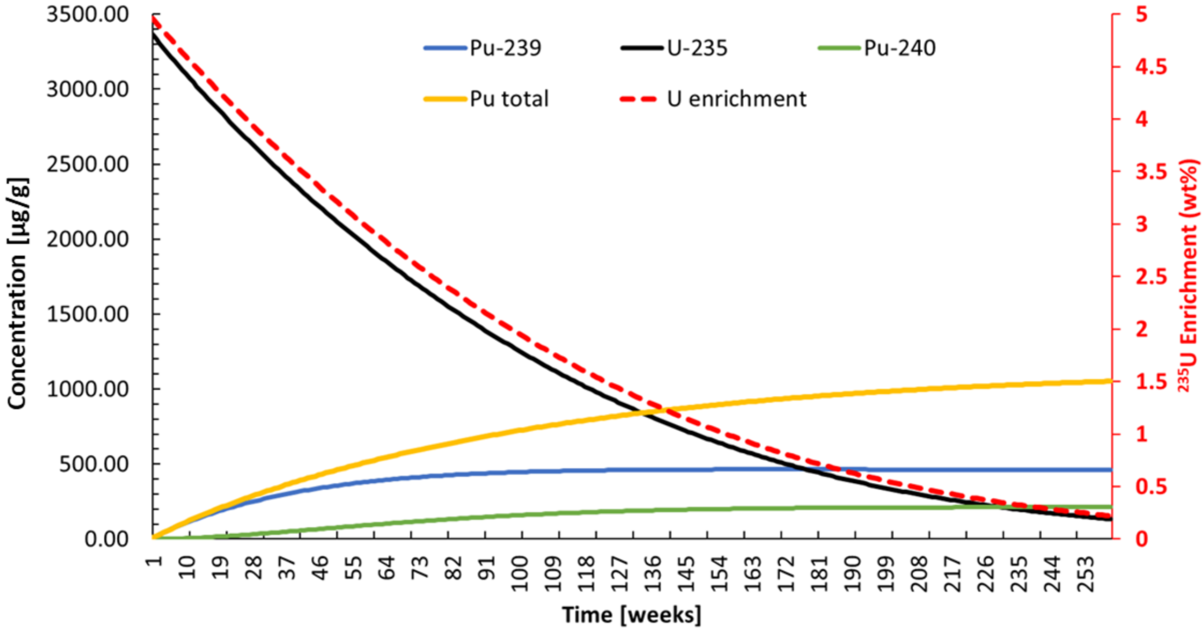

Figure 6. Dynamic concentration (µg/g) of SNM in the liquid fuel salt in the fuel cell of the TRANSFORM MSDR model. The enrichment of ^{235}U steadily decreases because of the closed fuel cycle.

At the end of cycle, the Pu content has reached a maximum concentration of 1,052 µg/g. If the categorization of the facility is Category II, then the diversion and detection of Pu are needed at or above 200 g. Using the concentrations given in Figure 6, the amount of fuel salt that would need to be diverted or stolen to amount to a total removal of 200 g of Pu is 190 kg. The liquid FLiBe salt density defined in the model was 3.353 g/cm^3 that results in 57 L of fuel salt removed that contains 200 g of Pu. These are large amounts of fuel salt that, as shown below, are extremely radioactive.

Alternatively, consider the SNM contained if a nominal volume of fuel salt was removed. For example, how much SNM is contained if 5 L of unseparated salt was diverted? Table 2 shows the quantity of SNM at startup, middle of cycle, and end of cycle. Using the formula quantity for SNM given in Section 2, the total amount of SNM at beginning, middle and end of cycle in 5 L is 57.4, 49.5, and 46.2 g, respectively. The SNM in the fuel salt is in dilute concentrations that should be considered in the development of elements of an MC&A approach.

Table 2. SNM quantities in 5 L of MSDR fuel salt at beginning, middle, and end of simulation cycle. *Note that the ^{233}U concentration is non-zero, but the quantities are less than 1 mg per 5 L of fuel salt.

Time (weeks)	^{235}U (g)	^{233}U (g)*	Pu, total (g)	Formula Quantity (g)
0	57.37	trace	0.00	57.37
130	14.69	trace	13.91	49.46
260	2.22	trace	17.59	46.19

4.1.3 Radioactivity of Irradiated Fuel Salt and Accessibility Considerations

Physical access and hazards like radiation dose and elevated temperatures can provide a means of deterring the potential theft or diversion of SNM and are considered by the NRC for domestic safeguards. The fuel salt and SNM in reactor containment is not directly accessible because it is inside the reactor containment boundary. The environment inside this boundary will have extreme radiation hazards and elevated ambient temperatures. For example, during operation of the Molten Salt Reactor Experiment at ORNL, gamma radiation exposure in the reactor cell was on the order of 40,000 – 70,000 R/hr at 8 MW_{th}.

Radiation dose rate estimates were calculated from the isotopic concentrations of the modified MSDR model. GADRAS was used to calculate dose rate contours based on a modeled geometry of a pipe of the primary fuel loop and of a drain tank. A drain tank is used as a separation mechanism for the gas and liquid components of the off-gas system. The dose rate was calculated at the beginning, the midpoint, and the end of the 5-y cycle. The steel pipe was open ended with a 30.5 cm diameter and a 2.54 cm wall thickness. The pipe section holds 1,492 kg of fuel salt. The drain tank defined in the MSDR model consists of several layers of wall material with air gap separations and tubing intruding into the core of the drain tank. This geometry was simply represented as a cylinder with an outer diameter of 335 cm (11') and height of 457 cm (15'). The main tank wall was made of Hastelloy-N with a thickness of 11.4 cm, with an air gap between the layers occupying 13 cm. The pipes running into the center of the drain tank were modeled as a block of Hastelloy-N. The liquid fuel salt component of the off-gas containing the soluble fission products fills the remaining volume equal to 4,197 kg of liquid salt. The labeled geometries and dose rate contours are given in Figure 7. The dose rate calculation was found using only the photon flux. In future efforts, the neutron dose term can be added.

On contact, there is a dose rate at the pipe of the primary fuel loop of $\approx 4\times10^5$ rem/h falling off to $\approx 1\times10^5$ rem/h at 1 m standoff. The maximum dose rate for the pipe of the primary loop changes from $\approx 4\times10^5$ rem/h to $\approx 7\times10^5$ rem/h comparing the beginning to the midpoint or end of the cycle. There is no significant difference comparing the midpoint to the end of cycle. The drain tank dose rate is significantly less than the primary fuel loop. This is likely due to the increased wall thickness and the block of Hastelloy-N inserted into the geometry to represent the tubing because the radionuclide concentrations (minus the noble gases) are essentially the same as the primary fuel loop especially after equilibrium is reached. The dose rates on contact for the drain tank drop to 5 – 7 rem/h. A lethal dose defined by the NRC is 400 – 450 rem. In the reactor containment boundary, this could be approached in short periods of time even at significant standoff from the primary fuel loop area. The drain tank dose rates are lower but still significant. These calculations will be revisited in future work to accurately model the tubing instead of approximation by a solid metal block to ensure the dose rates are more representative of the model geometry.

If the results from the calculation in Section 4.1.2 are revisited, the dose rates calculated here can be used to estimate the dose rate of the 5 L volume. The 5 L volume is equivalent to about 17 kg of fuel salt (containing about 50 g of SNM), or about two orders of magnitude less fuel salt. It is assumed that the 5 L is diverted in a steel cylinder with a 2.54 cm wall thickness, and the cylinder has the same relative proportions as the pipe. A conservative estimate of the dose rate on contact is ≈1,000 rem/h. This dose rate is significant, and the sample would require special manipulation and handling. Radioactive decay is not considered in these calculations but could significantly reduce the dose rate over time. Future work will perform dose rate calculations for various samples considering storage and decay time to understand the reduction to the dose rate of diverted samples. Regardless, the intense radioactivity of irradiated fuel salt will be an impediment in diverting unseparated fuel salt, will require special handling equipment, and should be considered in the material attractiveness of SNM contained in the primary fuel loop of an MSR.

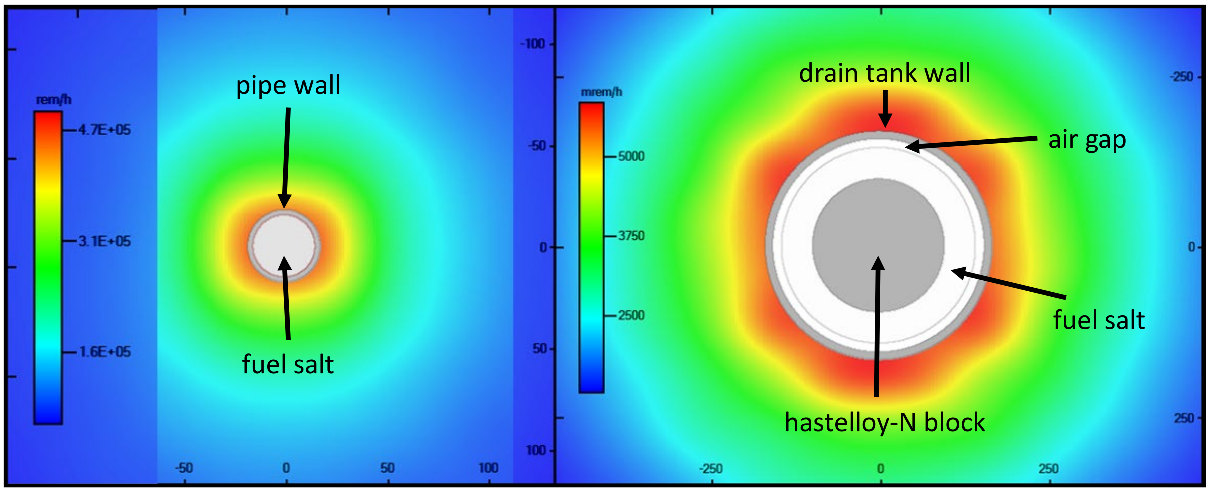

Figure 7. (Left) Dose rate contour of a pipe in the primary fuel loop of the MSDR at the beginning of the 5-y modeled cycle. (Right) Dose rate contour of the drain tank at the beginning of the 5-y cycle. The fuel salt is the liquid separated stream from the off-gas system. Axes units are cm.

5. A PROSPECTIVE MC&A PLAN FOR A SALT-FUELED MSR

MC&A is a regulatory framework that prescribes methods to "control and account" (i.e., safeguard) for SNM at a licensed nuclear facility. Specifically, the developed methods need to control access, account for SNM during periodic inventories, and detect theft or diversion of specific quantities (as defined in 10 CFR Part 74 based on the strategic significance category of the material) of SNM. The requirements for licensing and the categorization of facilities licensed by the NRC were discussed in Section 2. MC&A plans (typically in the form of Fundamental Nuclear Material Control plans) describe the facility-specific methods to control and account for SNM to meet NRC requirements for licensing. Some of the elements defined for MC&A by the NRC are [29]:

- Measurements – e.g., destructive assay (DA) and nondestructive assay (NDA)
- Measurement control
- Physical inventory
- Item control and monitoring – e.g., tamper indicating device (TID)
- Detection and resolution of indicators
- Process monitoring
- Independent assessment
- Recordkeeping

5.1 GENERIC COMPONENTS OF AN MC&A PLAN FOR A SALT-FUELED MSR

As described in Section 1, salt-fueled MSRs potentially include wide variations of safeguards-relevant design features. However, there are many common aspects of their operation such that a process flow and generalized MC&A approach for salt-fueled MSRs can be developed. For example, most, if not all, salt-fueled MSR designs include the following:

- Receipt of fresh fuel salt/makeup[6] salt (or - if synthesized onsite - fresh fuel salt components – e.g., SNM, salt compounds) at the facility.
- Storage of fresh fuel/makeup salt at the facility.
- Heating of fuel salt to enter reactor containment via piping.
- Irradiation of fuel salt within reactor containment.
    - Depletion of SNM and transmutation of fertile material to SNM (e.g., $^{238}U \rightarrow ^{239}Pu$).
- Draining of fuel salt from the core to drain tanks to shut down the reactor (e.g., for safety reasons, testing, or for scheduled maintenance/component replacement).
- Maintaining reactor components or periodic replacement throughout operational lifetime; equipment will likely be replaced using remote handling and specialized equipment.
- Following replacement of components, storage of irradiated/waste equipment onsite in containers and/or in a shielded environment.
- At the end of the reactor lifetime, irradiated "spent" fuel salt will be transferred into containers and stored onsite and/or shipped offsite.
    - Irradiated fuel could be used as fuel at other facilities.

A generic process flow of a salt-fueled MSR includes three potential control areas: 1) fresh fuel salt storage, 2) fuel salt in reactor containment, and 3) irradiated fuel salt and waste storage. Developing a safeguards approach with MC&A elements in each control area (which may be item control areas) would allow for mass balances in those areas. Figure 8 defines the proposed control areas (e.g., Item Control Areas) against the generic process flow (indicated by arrows).

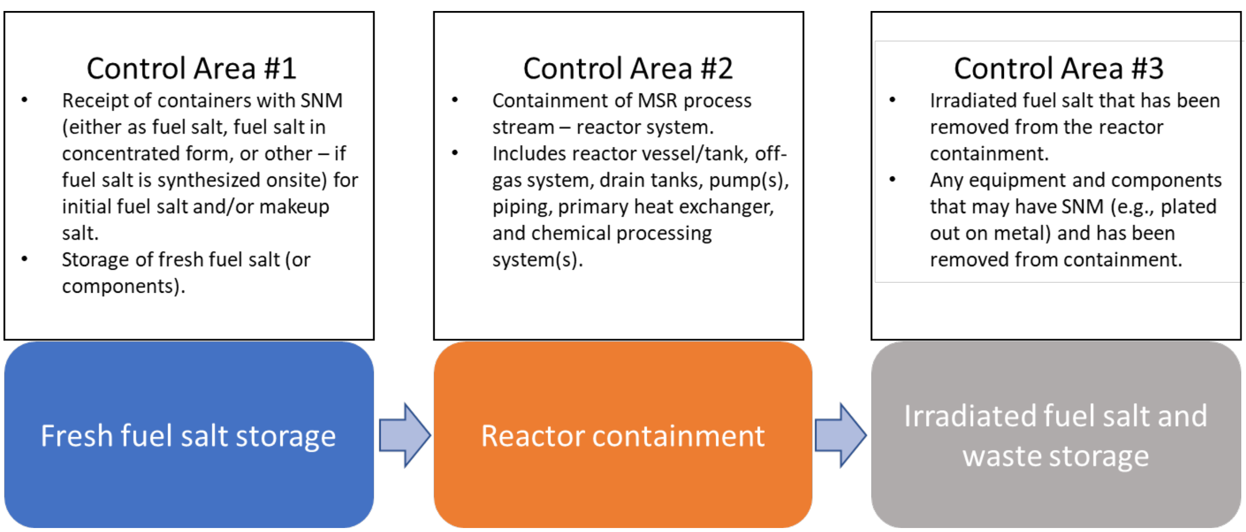

Figure 8. Proposed mass balance control areas and process flow for a salt-fueled MSR.

All MSRs will require delivery of fresh fuel salt in approved containers from a qualified fuel production facility. The fresh fuel will arrive in a metal or metal alloy container along with documentation of measurements that confirm the purity and molarity of the salt and SNM content. The majority of SNM (in

[6] The terminology used here as "fresh fuel salt" is fuel salt used upon initial startup while "makeup salt" is fuel salt added during operation to maintain reactor power.

irradiated fuel salt, flush salt, and/or as holdup in equipment) removed from reactor containment will be stored onsite. Some designs propose shipping the irradiated fuel salt to other MSR facilities or transferring to other MSR modules at the same site (potentially through piping) for reuse. It is not known what types of waste containers will be used for irradiated/spent fuel, or components taken out of service. Some spent reactor components may fall below radiation dose limits after a period of radioactive decay if most of the radioactivity is from shorter-lived fission and activation products. Regardless, containers for the irradiated fuel salt and reactor components will be distinct from fresh fuel containers due to the elevated radioactivity. The storage, shipping, frequency of online fueling and draining of irradiated fuel, and component replacement are design-specific metrics that will need to be considered and defined in developing an MC&A approach.

5.2 PROPOSED MC&A APPROACH FOR THE MSDR MODEL

A high-level MC&A approach was developed for the TRANSFORM modified MSDR model described in Section 3. The MSDR model only contains the components defined in reactor containment (Control Area #2). The receipt of fresh fuel or storage of fresh fuel (Control Area #1), or the removal of system components and irradiated fuel salt from the reactor containment for storage (Control Area #3) is not included in the computational model. In addition, online refueling and removal of irradiated fuel salt is not included. Therefore, assumptions have been made based on the authors' experience and knowledge of current practices for LWRs, gas centrifuge enrichment plants, fuel fabrication facilities, research reactors, and hot cells.

The MC&A objectives, MC&A elements, the type of SNM in the control area, the physical form of the SNM, and the measurement environment are summarized in Table 3. The MSDR is generally representative of several salt-fueled designs regarding MC&A-relevant features. Since Control Areas #1 and #3 are not included in the MSDR model, it assumes that fresh fuel is delivered to the site in the same fissile-to-salt concentration and in the same salt molarity as was defined in the primary fuel system (see Section 4). It is further assumed that there is no onsite synthesis or further processing (besides heating) of fresh fuel received at the facility.

This MC&A approach is generally applicable to salt-fueled MSRs without onsite synthesis of fuel, with continuous (or periodic) addition of makeup fuel salt, and periodic draining of the primary fuel loop for replacement of components. This generalized approach captures components that will be necessary to account for locations and quantities of SNM within the facility. Other MSR designs with features like online chemical separation of fissile material (i.e., in a breeder design), would likely require different or additional MC&A components.

Table 3. MC&A objectives, measurement conditions and MC&A elements to meet objectives for the TRANSFORM MSDR model.

<table>
<tr><th></th><th>MC&A Objectives</th><th>SNM in Area</th><th>Physical Form of SNM</th><th>Measurement Environment</th><th>MC&A Elements</th></tr>
<tr><td rowspan="6">Control Area #1: Fresh fuel salt storage</td><td rowspan="2">Quantify SNM upon receipt of material.</td><td rowspan="6">LEU (≤5 wt%)</td><td rowspan="3">SNM is within salt in a solid form.</td><td rowspan="3">Fuel salt contained within metal (or metal alloy) containers.</td><td>Unique identifiers (serialized tags) on each container for tracking.</td></tr>
<tr><td>Assay material being added to containment from containers; NDA and DA: gamma spectroscopy, mass spectrometry, hybrid k-edge densitometer, etc.</td></tr>
<tr><td rowspan="2">Ensure no SNM is removed from containers stored onsite.</td><td>Confirm integrity of TIDs on containers upon receipt and during inventories at facility.</td></tr>
<tr><td rowspan="3">Fuel salt will be heated to a liquid prior to entering reactor containment.</td><td rowspan="3">LEU has weak radioactive emissions.</td><td>Camera surveillance used to identify serial number of container(s) added to reactor containment.</td></tr>
<tr><td rowspan="2">Quantify SNM entering reactor containment.</td><td>Weigh fuel salt containers upon receipt, during inventories, and after salt is removed and added to reactor containment.</td></tr>
<tr><td>TIDs on any valves that could be manually opened to allow access to the system.</td></tr>
<tr><td rowspan="4">Control Area #2: Reactor containment</td><td rowspan="4">Ensure no SNM is removed from reactor containment</td><td rowspan="2">LEU (≤5 wt%)</td><td rowspan="2">SNM is liquid in salt matrix.</td><td rowspan="2">Extremely high radiation due to fission products.</td><td>TIDs on all pipes/hatches that could be used to access the containment.</td></tr>
<tr><td>Measure actinide concentrations in fuel salt using NDA and/or DA, and total salt volume.</td></tr>
<tr><td rowspan="2">Dynamic concentrations as ^{235}U is depleted and Pu is produced from neutron irradiation.</td><td rowspan="2">Molten salt is not physically accessible.</td><td rowspan="2">Elevated ambient temperature.</td><td>Camera surveillance on all access points to detect removal of SNM.</td></tr>
<tr><td>In-situ NDA techniques (gamma spectroscopy), processing monitoring for indications of diversion.</td></tr>
</table>

<table>
<thead>
<tr><th></th><th>MC&A Objectives</th><th>SNM in Area</th><th>Physical Form of SNM</th><th>Measurement Environment</th><th>MC&A Elements</th></tr>
</thead>
<tbody>
<tr><td rowspan="8">Control Area #3: Irradiated fuel salt and waste storage</td><td rowspan="2">Quantify SNM in salt leaving reactor containment.</td><td rowspan="4">LEU (≤5 wt%)</td><td rowspan="8">Assumed SNM is in salt matrix either as a liquid or solid.</td><td rowspan="4">Irradiated fuel salt is in shielded containers.</td><td>TIDs placed on doors of storage area.</td></tr>
<tr><td>TIDs placed on storage containers once filled with irradiated fuel salt or reactor components.</td></tr>
<tr><td rowspan="3">Ensure no SNM is removed from containers or equipment stored onsite.</td><td>Assay material being removed from containment to containers; NDA and DA: gamma spectroscopy, mass spectrometry, hybrid k-edge densitometer, etc.</td></tr>
<tr><td>TIDs placed on equipment entry points (e.g., waste reactor vessel closure).</td></tr>
<tr><td rowspan="4">Plutonium and other actinides in irradiated fuel salt.</td><td rowspan="4">Irradiated fuel salt is very radioactive because of fission products.</td><td>NDA holdup measurements and imaging techniques to locate material.</td></tr>
<tr><td rowspan="3">Estimate holdup of SNM in equipment removed from reactor containment.</td><td>Camera surveillance on entry/exit points to storage area(s).</td></tr>
<tr><td>Weigh empty and filled containers during inventories or upon changes to content.</td></tr>
<tr><td>In-situ gamma-ray spectroscopy, process monitoring on piping and/or containers as SNM is moved out of reactor containment and into containers.</td></tr>
</tbody>
</table>

†There will be minimal changes in radionuclide inventories over time due to radioactive decay (e.g., $^{241}Pu \rightarrow ^{241}Am$).

6. CONCLUSIONS AND FUTURE WORK

This report summarized the research progress for fiscal year 2021 of the MC&A for MSRs project. The project is developing a domestic MC&A approach through interactions and discussions with the MC&A group at the NRC, and by performing dynamic system level modeling to understand the SNM variability that may impact the MC&A approach. The objective is to provide guidance and documentation to potential vendors to assist in the licensing process. A critical aspect of defining an MC&A approach for salt-fueled MSRs is defining the regulations that such an approach would be evaluated against. To advance this topic, the authors published a report that presented MC&A considerations for salt-fueled MSRs [6]. Discussions and virtual meetings have been ongoing with the MC&A group from the NRC. The NRC reviewed the report, provided positive feedback, and indicated that many of the challenges that MSRs face were covered in the content. It has been very informative for the ARS team to engage with the NRC. The dialogue has fostered a deeper understanding of the licensing steps and the status of NRC regulations of how salt-fueled MSR facilities might fit into the current regulations. Currently, new MSR facilities will have to function within existing regulations and provide justifications for any exemptions. The current regulations were written to ensure the safety and security of existing facilities – including LWRs and fuel fabrication facilities, and some of the language included is insufficient for MSR design characteristics or not (entirely) applicable.

The authors also performed a signature study to simulate gamma-ray spectra using isotope concentrations from the TRANSFORM MSDR model [12]. Direct, online radiometric measurements may not be sensitive enough to detect small amounts of diverted material, but they can be used to monitor reactor power and detect reactivity changes inflicted during refueling or other operational conditions. There are fission product signatures that can be used to understand the fuel composition due to a combination of a high fission yield, photon branching ratio, and specific activity. The modeling and signature study demonstrated the ability to simulate and quantify isotopic populations at key reactor system components. Continued research will investigate how gamma-ray spectra measurements and analysis could be used from an MC&A perspective.

It is important to understand the current licensing requirements established by the NRC. A summary of some of the domestic licensing requirements for nuclear power plants and facilities has been provided. The focus was placed on the regulations particular to MC&A and SNM. The definition and category of SNM is a major consideration in the licensing requirements, especially for MC&A of the SNM at the facility. One research goal of this project is to develop an MC&A approach for salt-fueled MSRs by considering the existing requirements and using the TRANSFORM MSDR model as an exemplar commercial facility. Generalized components for MC&A of a salt-fueled MSR were given in Section 5.1. The proposed general areas follow the fuel (SNM) material as receipt and storage of fresh fuel, into the reactor containment and irradiation, and finally, once removed from reactor containment for storage or as waste. Alternatively, the defined concept of three control areas could be three Item Control Areas within one mass balance area. If that definition was imposed, the function, technology, and MC&A elements needed would remain very similar to what was proposed here.

The generalized MC&A approach was applied to the TRANSFORM MSDR model. The greatest challenges of a successful MC&A approach are found in the transition of the SNM from one area to the next because it will be harder to quantify changes in SNM content during these transient conditions. In all other boundaries, when the SNM is fixed in storage containers or in reactor containment, MC&A elements have been identified that should sufficiently confirm no loss or diversion of SNM. However, during refueling and draining, quantitative measurements must be made to identify the isotopic makeup and amount of SNM (fuel salt) added or removed to (or from) the system. This may require multiple measurement points on the pipe transfer system or whatever mechanism will be used to fuel or drain the MSR.

Currently, only modeling can provide radionuclide inventories of salt-fueled MSR systems. In order to identify potential MC&A detection technologies and frequency of measurements, a system-level model has the potential to inform a proposed MC&A approach. Therefore, significant progress has been made to TRANSFORM and the coding tools to process and parse the output files. Nuclear data and matrix calculations from the SCALE/ORIGEN software have been integrated into TRANSFORM. The calculations of isotope depletion and other nuclear reactions have been verified comparing the modified TRANSFORM tool to the SCALE results [16]. The tools developed to process the TRANSFORM output data have flexibility to produce plots of the data or output the data in formats for further analysis. To demonstrate the capabilities of the modified TRANSFORM package, simulations were run to quantify the SNM present in the model MSDR system over a five-year operational cycle. The concentration of the SNM in the fuel salt showed a maximum of 1,000 μg/g total Pu at the end of the operational cycle. Coupled with calculations of very high radiation dose and physical access to the fuel salt while in reactor containment, the low concentration presents a challenge for diversion or theft of irradiated fuel salt. This would require significant capabilities to safely remove large amounts of fuel salt. Therefore, these inherent features of the fuel salt should be considered in an MC&A approach.

The future work scope of this project is to support the development of a safeguards approach to assist MSR vendors during the licensing phase. The proposed MC&A approach will likely have components that are design specific and will require coordination with the NRC to develop this approach that meets regulations. The tasking for FY22 is organized to support these goals.

- Through modeling and analysis, MSR dynamic inventories will be evaluated for available designs to inform an MC&A approach. The MSDR model will be modified to integrate vendor specific operation or components to understand the influence on the fuel salt and SNM concentrations.

- Collaborations with the NRC will continue to discuss and identify the challenges involved in an MC&A plan for MSRs. The key components or areas of an MSR facility that may require further description for licensing (i.e., design-specific components) will be categorized.

- ORNL will assist LANL's NDA campaign with measurements of molten salt-like samples that will occur in FY22 at ORNL.

Enabled through the project's research in FY 2021, a report documented the future infrastructure needed to fully support evaluations of MC&A for MSRs [11]. The report included funding needs in the following research areas:

1) Irradiated salt experiments and accessing representative nuclear material forms/types,

2) Developing online or in situ monitoring capabilities, and

3) Expanding system components testing and increased modeling capabilities to consider nuclear material accounting.

There are other areas of research that require investigation to support an MC&A approach for salt-fueled MSRs. These include the holdup of fissile material in reactor system components or core materials (e.g., graphite used for neutron moderation) which has not been fully considered and NDA systems capable of dealing with high-rate environments and/or molten salt samples. Therefore, technology or detector systems capable of identifying and quantifying holdup in the reactor components will be needed. Other systems capable of determining SNM concentrations in the molten fuel salt would be a valuable addition if coupled with techniques to determine bulk salt quantities. The commercial Hybrid K-Edge Densitometry system has been evaluated against surrogate fuel salt matrices but will require modifications to achieve high precision and accuracy in the presence of large amounts of fission products. Potential changes to the system could be achieved utilizing the radionuclide inventories found using TRANSFORM and coupled with high fidelity simulations of the detector system. In addition, other NDA systems capable of handling high gamma exposure rates but remaining sensitive to all portions of the energy spectrum (higher peak to Compton ratios) could provide useful information when combined with the excellent low energy response of the SOFIA microcalorimeter being developed at LANL. Coincident gamma spectroscopy systems are developed specifically for this method. The resulting anticoincident spectrum can significantly reduce the Compton continuum and provide enhanced sensitivity to lower energy photons that would be useful in an online measurement system especially during transient reactor conditions.

7. REFERENCES

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APPENDIX A. PROJECT OVERVIEW: MOLTEN SALT REACTORS – MATERIAL CONTROL AND ACCOUNTABILITY (MC&A)

APPENDIX A. PROJECT OVERVIEW: MOLTEN SALT REACTORS – MATERIAL CONTROL AND ACCOUNTABILITY (MC&A)

This research project, Molten Salt Reactors – Material Control and Accountability (MC&A), was funded under the ARS program in the US Department of Energy's Office of Nuclear Energy. The program is supports advanced reactor technology currently funded under the Office of Nuclear Energy's Advanced Reactor Demonstration Program. The overall goal of ARS is to identify, and address challenges these new (reactor) technologies face for MC&A and physical protection systems.
The goal of the ORNL research project is to perform an assessment of MC&A for salt-fueled MSRs. The effort provided early insight into safeguards and security by design and has identified challenges to assist the MSR vendor(s) with licensing requirements. The project was divided into the following tasks:

1. *Nuclear Regulatory Commission (NRC) Interactions:* This task will engage with the NRC MC&A group to discuss their view of licensing requirements for MSRs and potential vendor interactions. This work will leverage the framework defined by prior ORNL work on MC&A approaches for pebble bed reactors. The proposed deliverable is a memo report outlining a list of technical questions and considerations for vendor interactions with the NRC when starting the path to licensing their MSR technology.
    a. **Progress and Status:** Staff from ORNL hosted two virtual meetings with the NRC's MC&A group in calendar year 2021. The interactions and discussion with the NRC contributed to an increased understanding of NRC concerns for salt-fueled MSRs and served as a mechanism to propose our ideas and prospective approaches to the problem. The dialogue and discussions from the two meetings are summarized in this report (see Section 2.1). Another key outcome from these discussions was the report titled "Domestic Safeguards Material Control and Accountancy Considerations for Molten Salt Reactors [13]." This effort will continue into fiscal year 2022 to document NRC activities and priorities for ARS projects related to MSR MC&A and gather feedback on the development of a prospective MC&A approach for salt-fueled MSRs.
2. *Development of a Monitoring Approach:* The objective of this task is to continue to develop safeguards and security by design concepts for MSRs using the Molten Salt Demonstration Reactor (MSDR) as the target design. Findings from the MSDR reactor physics model and understanding key similarities and differences across current MSR vendor designs is critical for evaluating safeguards challenges. The goal is to develop a MC&A plan for salt-fueled MSRs.
    a. **Progress and Status:** The MSDR Transient Simulation Framework Modules (TRANSFORM) model has continued to be refined and enhanced. The code has been supplemented with SCALE/ORIGEN nuclear data and matrix calculations which provide confidence in the dynamic isotopic concentrations. Scripting tools were developed to easily manage the output of the MSDR model and generate gamma-ray spectra of defined geometries. The status of the modeling effort and how it is being analyzed for developing an MC&A approach is discussed in this report. In addition to the improved MSDR TRANSFORM model, the authors produced a mid-year report titled "Signature Analysis Utilizing a Dynamic Molten Salt Reactor Model for MC&A" under this task [12].
3. *Infrastructure Assessment:* This task reviewed and summarized gaps in safeguards technology and US infrastructure to support the evaluation of domestic safeguards approaches for MSRs.
    a. **Progress and Status:** The deliverable of this task was a report, "Infrastructure and Testing Needs for Molten Salt Reactor Safeguards," that documented the findings [11]. This effort was completed in FY21.

In addition, the ORNL team supported Los Alamos National Laboratory's project *Experimental Validation of Nondestructive Assay Capabilities for MSR Safeguards*. A measurement campaign was planned and performed from Sept. 20–24 2021 at the Irradiated Fuels Examination Laboratory at ORNL.

The objective of the measurements was to evaluate and validate three neutron detector's performance in realistic operational environments at ORNL. The neutron instrumentation was delivered to ORNL and inserted through a collimator port into the hot cell. Segmented samples of spent LWR fuel were placed in repeatable proximity to the detectors for neutron and gamma flux measurements. The measurement campaign resulted in several key findings as summarized in LANL's report [17]. This collaboration will continue in FY22 and additional measurement campaigns at ORNL are being planned.