

Safeguarding Commercial Breed and Burn Sodium-Cooled Fast Reactors

**Nuclear Science and Engineering Division
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Executive Summary

A new class of reactors is under development in the United States. These new reactor designs use fast neutron spectra and rely on fuel technology advancements to reach fleet-wide operating conditions where neither enrichment nor reprocessing fuel are required to sustain or grow the fleet. Achieving this level of operation requires nuclear fuel that can remain in the reactor environment for much longer than current limits. The transfer of irradiated fuel is also required as the initial fissile source for new and replacement reactors. This class of reactors is known as breed and burn reactors and can be deployed as a fleet whereby the transfer of fuel is used to power subsequent generations of reactors. The use of breed and burn reactors reduces, and potentially eliminates, the need for enrichment and reprocessing to sustain a reactor fleet. Currently, enrichment and reprocessing plants require extremely complicated safeguards implementation and offer States diversion pathways for indirect use material. By eliminating the need for these facilities, a nuclear power fleet based on the BREED AND BURN reactor concept has the potential to indirectly reduce the burden on IAEA safeguards. However, this complex dynamic system does pose new safeguards challenges that are explored further in this report.

Initial deployment of a breed and burn fleet will require a substantial inventory of low enriched uranium (LEU) including enrichment above 10%. Over time, a system of reactors will evolve to have substantial in-core inventories of plutonium (Pu) at concentrations near 10%. After the initial inventory, no additional fissile material is required to sustain operations because the required fissile material is bred in-situ, referred to as “breed and burn.” For future reactors, the initial fissile material will come from partially irradiated used nuclear fuel (UNF) that will continue to see use until the end of its practical operational lifetime.

This study evaluates safeguards-relevant aspects of a designed system including but not limited to inventory and isotopic evolution of UNF, the amount of special nuclear material in the core at the end of the reactor life, and the unique challenge of transferring unprocessed/unconditioned special nuclear material in the form of UNF to a new reactor. The overall goals of this study are therefore to identify and quantify the complex dynamic inventories for an example system that will need to be under international safeguards, evaluate how international safeguards measures can be applied to operation of a fleet of breed and burn reactors, and identify challenges that exist for the application of safeguards to this unique system.

Given the wide range of potential designs for this initial assessment¹, an example reactor design was selected which is deployed using LEU for its initial core and transitions to equilibrium breed and burn operations. The example system consists of two generations of solid-fueled, sodium-cooled fast reactor designs. The first-generation reactor is initially fueled with LEU fuel up to 14.15% enriched. The second-generation reactor is initially fueled with a mix of UNF discharged from a first-generation reactor and low enriched uranium fuel.² After the first cycle, both the first- and second-generation reactors require only depleted uranium for all refueling. Plutonium is bred in the depleted uranium fuel introduced to the

¹ Currently, U.S. vendors TerraPower and General Atomics are developing breed and burn reactors concepts. However, if this concept of operations is demonstrated, it could be applicable to a wider range of fast reactor designs.

² Some deployments do not require additional enriched uranium fuel in second generation reactors.

reactor, allowing it to sustain criticality. The concentration of Pu in the fuel changes with each cycle, eventually reaching an equilibrium concentration at discharge after several cycles of operation.

The modeling approach used in this study provides representative material inventories and flows for the nearer-term deployment of breed and burn reactor technology. A key aspect of this technology is the interconnection of reactors that results from the utilization of UNF in one reactor that was previously irradiated in one or potentially multiple other reactors. To inform on the behavior of the material inventories and flows within a fleet of reactors, system dynamics modeling tools were used to evaluate different potential scenarios and perform sensitivity studies on key parameters to inform the safeguards assessment.

This study focuses on the unique or substantially different technologies beyond those used for current safeguards approaches for conventional light-water reactors (LWRs) operating on a once-through cycle or sodium-cooled fast reactors (SFRs) operating on a closed-cycle. The following differences were identified:

- Utilization of UNF as part or all the initial fissile inventory for newly deployed SFRs requires mechanisms to verify Pu content of fuel received from a separate reactor.
- There is a greater evolution in reactor *core* nuclear material inventories from initial deployment to equilibrium conditions.
 - SFRs deployed with a core inventory will see significant increase in significant quantities (SQ) of Pu. A SFR of approximately 150 SQ of LEU evolves to approximately 1300 SQ of Pu at the end of an equilibrium cycle.
 - UNF can be discharged from a first cycle with about 0.5 SQ of LEU and 0.5 SQ of Pu per assembly evolve to equilibrium spent fuel discharged with about 5 SQ of Pu per assembly.
- Differing from LWRs, *assembly* compositions exhibit a wider range between freshly depleted uranium fuel to peak Pu concentrations greater than 10% Pu over the long lifetime of a fuel assembly.
 - Greater swings in special nuclear materials (SNM) content from core loading to discharge places a higher emphasis on accurate results of fuel isotopic evolution calculation for material accounting.
- There is potential for complex, dynamic material flows. For example, an assembly may flow from the fabrication plant, be irradiated in one reactor, transferred to another reactor, and possibly additional reactors before ultimately being sent to a repository.

At the time of this study, the most relevant safeguards techniques for breed and burn class of reactors were found to be a robust Continuity of Knowledge (CoK) regime and reliance on Containment and Surveillance (C/S) for Nuclear Material Accounting and Control (NMAC). These findings were based on observing the safeguards regime in place during the Monju fast reactor as well as existing practices for managing spent fuel inventories for power reactors. Given the reliance on CoK, further considerations for the breed and burn fleet should be made to include methods of reinforcing existing CoK capabilities and reverification if CoK were ever lost. Maintaining CoK requires the collection and maintenance of detailed historical data for UNF in the fleet. As such, detailed reporting of operational history is useful for both safeguards implementation and reactor operations.

Technology for the direct measurement of Pu content in UNF discharged from breed and burn reactors is not currently available. Consequently, quantifying nuclear material for reporting will require the use of

core depletion calculations. This is standard practice for many reactors today, however the high burnup, correspondingly high nuclear material concentrations, and large changes in those concentrations of breed and burn reactor UNF increases the importance of accurate and reliable calculations.

The system of breed and burn fleet reactors is still under development. To date, non-traditional reactors such as those applicable in a fleet are few and safeguards approaches are developed on specific case-by-case basis. Full deployment of the technology is likely decades away; however, the use of core spawning could have benefits in any fast reactor system to reduce initial fissile inventory requirements. Understanding these safeguards challenges will allow for development of appropriate methods, techniques, and controls in advance to ensure a robust safeguards regime is in place for this promising technology.

Further studies regarding the applicability of safeguards to breed and burn fleets will evolve as the concept continues development. Considerations must be made for increasing fleet size, the implication of multi-country participation, and fundamental operations of an international fleet, among other unique challenges. Future studies that may arise from this work include:

- Considerations for other technologies, specifically molten salt fast reactors with liquid-fuel.
- Collaborative research with specific vendors utilizing their specific designs and technology.
- Evaluation of the relevant safeguards techniques identified through demonstration at the facility level to verify applicability.
- Using comprehensive State-specific models, including details such as Additional Protocols, overall energy consumption, resource limitations, etc., to develop detailed analysis for a breed and burn fleet deployment.
- Quantitative analysis of the uncertainties associated with special nuclear material within a realistic deployment of a breed and burn reactor fleet scenario.

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1. Introduction

Key Takeaways for International Safeguards

- This study considers a model breed and burn fleet with a generic solid-fuel, sodium fast reactor concept based on research at Argonne National Laboratory and elsewhere. The study estimates nuclear material inventories and compositions for facility-level safeguards implementation and expands the discussion to encompass a fleet deployment scenario of this reactor type.
- Reactors operating on the breed and burn concept utilize discharged used nuclear fuel from one reactor as the startup fuel for a subsequent generation of reactors. This presents a novel challenge for the international safeguards regime: verifying the composition of used nuclear fuel received at a reactor facility.
- The key aspects of breed and burn reactors are a fast neutron spectrum, very long fuel residence time, a high discharge burnup relative to current light water reactor operations, and a breeding ratio slightly above 1.0. These aspects result in used nuclear fuel with high plutonium concentrations, relative to light water reactors.
- Reactors operating on the breed and burn concept have the potential to reduce the need for uranium enrichment during refueling. This potentially alleviates pathways available to a State to obtain weapons usable nuclear material.

A once-through nuclear fuel cycle supporting a commercial power sector mainly consisting of light water reactors (LWR) utilizes only a small percentage of the natural uranium feed with approximately 85% of the natural uranium mass becoming depleted uranium (DU) tails (generally considered as waste) and 15% as low enriched uranium (LEU) fuel, of which about 4% is fissioned. Overall fuel utilization, expressed in terms of the input natural uranium, is only ~0.6%. Greenspan estimates that of the 700,000 tons of uranium mined for U.S. power generation, approximately 60,000 tons becomes used nuclear fuel (UNF), and more than 600,000 tons is rejected as waste [Greenspan, 2012]. Fast breeder reactors (FBR) were thus proposed as a possible solution to the low utilization of natural uranium. This was a popular concept early in the nuclear age when resources were thought to be more limiting. By recycling³ actinides, FBRs are capable of fissioning nearly 100% of the original natural uranium and closing the fuel cycle (only high-level waste sent to a repository). However, absolute uranium utilization is not possible under a single irradiation campaign due to neutron-induced damages and other environmental factors that limit the fuel burnup levels. Thus, absolute uranium utilization would require the fuel to be recycled multiple times [Greenspan, 2012].

A special class of fast reactors referred to as “breed and burn” or “travelling wave” reactors can forego fuel reprocessing by breeding plutonium (Pu) in-situ, through fissioning a significant fraction of the DU feed. Breed and burn reactor concepts avoid fuel reprocessing if fuel can be designed to survive in the

³ The terms reprocessing and recycling are both used in this paper. In the context of this paper, recycling refers to all processes to reuse fissile material while reprocessing refers to the irradiated nuclear fuel processing steps that involve separation of the fissile material.

reactor environment for sufficiently long periods of time, allowing for in situ breeding of Pu to sustain critical operations. This requires unique fuel characteristics that allow very high burnups and significantly higher fast neutron fluences that have been demonstrated through current experience.

Breed and burn reactors are a family of reactor concepts including many preexisting concepts in the nuclear industry, which culminate to a system of fast neutron reactors that can be sustained indefinitely without enrichment or, reprocessing and could operate off the large existing inventories of DU for centuries. Breed and burn reactors only require externally supplied fissile material (i.e., enriched uranium) for initial startup feed. A fleet of breed and burn reactors will ultimately reach an equilibrium such that neither enrichment nor fuel reprocessing would be required to sustain or grow the fleet. In comparison, LWRs using a traditional once-through fuel cycle rely on expensive enriched uranium at reactor startup and during subsequent refueling to sustain criticality. Breed and burn reactors require enriched fuel only to start the first generation of reactors, but do not require enriched fuel for refueling or in subsequent reactors. Thus, the concept of utilizing breed and burn reactors is appealing for maximizing fuel utility and reducing the economic burden on the world's uranium capacity.

Various designs, approaches, and configurations have been studied for breed and burn reactors. TerraPower's Sodium Ultimate [Neider, 2021] is one of the more prominent concepts currently under development and Neider includes a well-documented history of its evolution. Other approaches for breed and burn fleets have been considered, such as described by Huang in which a comprehensive history of the design was given [Huang, 2015]. The key aspects of breed and burn reactors are a fast neutron spectrum, very long fuel residence time, a high discharge burnup relative to current LWR operations, and a breeding ratio slightly above 1.0.⁴ To sustain breed and burn operations, a minimum of about 20% of the initial fuel loading (i.e., heavy metal or HM) is required to fission [Greenspan, 2012]. A large advancement in fuel technology is required to ensure assemblies survive in the reactor environment for the required duration. This is due to the larger core size and subsequent reshuffling of fuel to achieve the required breeding ratio while satisfying the necessary safety criteria.

It is important to understand how nuclear safeguards concepts can be applied to a breed and burn reactor fleet such that the relevant techniques may be identified. Current safeguards approaches were predominantly developed for LWRs, and the limited experience implementing safeguards at fast reactors has been historically difficult. Therefore, identifying the challenges to safeguards implementation at breed and burn reactors prior to deployment is vital to understanding and addressing possible technological gaps. The Key Takeaways featured in the sections of this report seeks to summarize the most relevant discussion for safeguards.

There are key differences in safeguards implementation at a reactor operating as part of a breed and burn fleet compared to those of a more traditional fast reactor deployment. Large commercial power reactors that must operate on the fast neutron spectrum for breed and burn operations have a much higher fissile requirement on a per unit power basis (when compared to thermal spectrum reactors) and therefore have a much higher initial fuel cost. To minimize this fuel cost, the UNF discharged from other reactors may still be utilized as the fissile source for subsequent reactor deployments until the practical end-of-life is achieved. The study is therefore focused on those aspects of safeguarding such a system that are unique

⁴ Breeding ratio is defined as the ratio between the average rate of fissile atom production and average rate of fissile atom consumption. Reactors with breeding ratio greater than 1.0 are considered breeder reactors as they produce (breed) more fissile material than they consume.

or significantly different from currently deployed commercial LWRs or sodium-cooled fast reactor (SFRs) operating as part of a closed fuel cycle.

1.1. Primary Assumptions

A model breed and burn fleet with a generic solid fuel based on the SFR research at Argonne National Laboratory and elsewhere was evaluated and estimates of the relevant fleet data was generated. The reactor model is the basis for applying the safeguards assessment. The design is assumed to operate on a typical 1.5-year cycle length. The key assumptions for this study are as follows:⁵

- A. Discharged fuel does not require recladding, conditioning, or processing prior to reuse as starter fuel in a subsequent reactor.
- B. The fuel assembly remains intact for all stages of the fuel cycle from initial fabrication of the unirradiated fuel to direct disposal of the intact spent fuel assemblies.
- C. For the safeguards assessment, the breed and burn fleet consists of two generations of reactor deployment—an initial LEU-fueled core in a first generation (S1) reactor followed by a single second generation (S2) reactor core that uses UNF as part of its startup fuel. (Note that transitions studies considered multiple generations).
- D. Following the initial fueling, all fuel used for refueling is natural or depleted uranium.
- E. To simplify concepts for the safeguards assessment, each generation of deployment involves one fast reactor.
- F. All deployed reactors and fuel transfers are isolated to one country that has at least signed a comprehensive safeguards agreement with the International Atomic Energy Agency (IAEA).

The breed and burn fleet is a complex system involving several mechanisms that would be working in tandem. Therefore, the study focuses on a simple model of two deployments to demonstrate the most relevant safeguards considerations: in particular, the relevant techniques for material accounting and measurements of UNF. A brief discussion on the implications of a real-world deployment scenario is given. Discussion on future work in addition to this study may include more comprehensive considerations, such as multiple deployment stages to the fleet, reactors (and fuel) from several different countries with and without Additional Protocols in force and applied safeguards measures to an ever-increasing complex system. This analysis does not address the complexities associated with Additional Protocol reporting for Article 2 activities.

1.2. Breed and Burn Operations Description

At the time of this study, there are two deployment scenarios under consideration for breed and burn reactor fleets. The first mode proposes deployment using LEU to fuel a first generation of reactors (the first generation reactors are referred to as S1 reactors), followed by subsequent generations that use a mix of partial core discharges from the first generation and LEU fuel to start up. During the early cycles of these S1 reactors, partially irradiated fuel (i.e., assemblies that have not reached lifetime limiting exposure) will be removed and used as startup fuel for the next generation of reactors (S2 reactors). This discharge of low burnup UNF is typical of startup fuel for all multi-batch reactors, including breed and burn reactors. To avoid the need for entire new cores of unirradiated LEU fuel, the UNF is expected to be used to the extent practicable in the deployment of new breed and burn reactors. A version of this

⁵ Section 4 of this study addresses fleet deployment and includes an additional set of assumptions, identified therein.

deployment scenario is described by Neider [Neider, 2021]. The second mode involves the transfer of a full core from any generation reactor at the end of its operational lifetime for use as the first core in its replacement. The two possible modes of operation are described in Table 1.

Table 1. Breed and burn reactor Operating Modes

Operating Mode 1 (Partial Core transfer, Deployment Phase):		
	Initial Core	Refuel
<i>First Generation – S1</i>	LEU	DU or NU
<i>Second Generation – S2</i>	LEU and UNF from S1 (Partial Core)	DU or NU
Operation Mode 2 (Full Core Replacement):		
	Initial Core	Refuel
<i>First Generation – S1</i>	LEU	DU or NU
<i>Second Generation – S2</i>	UNF from S1 (Full Core)	DU or NU

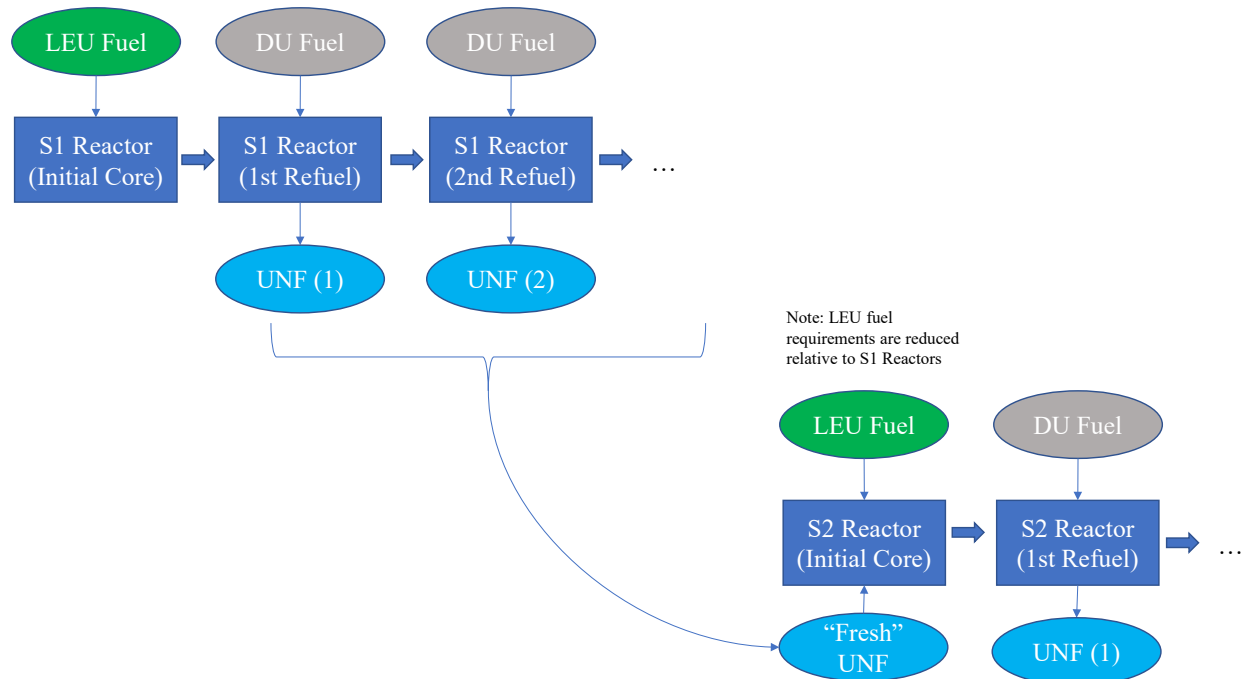


Figure 1. Deployment schematic of first generation (S1) and second generation (S2) reactors in the example fleet.

The example deployment strategy developed for this study implements a variation of Operating Mode 1. Figure 1 shows the schematic of the deployment of S1 and S2 generations of breed and burn reactors where key elements of fuel flow between the two generations can be seen. S1 will utilize only LEU for the initial fissile inventory, and subsequent refueling will rely on DU without additional fissile material to sustain operations. The S2 reactor will receive a fraction of its initial fissile inventory from UNF and part from LEU fuel (the amount is determined by many practical considerations). Following initial startup, the UNF discharged from S2 will contribute to the inventory of UNF in support of fleet expansion operations

(within a model state where no enrichment or fuel reprocessing is needed to sustain the fleet, and the isotopic composition is constant between cycles).

1.3. Definitions

This study uses various terms that are common to the breed and burn reactor literature but are not commonly used in other areas of nuclear engineering. These terms are defined for reference here:

Breed and Burn – A state of reactor operations with a breeding ratio slightly greater than 1, where operations only require unenriched uranium fuel (i.e., source material) as feed to sustain operations. After initial fissile inventory, the reactor breeds sufficient fissile material that no external source of fissile material is required in the form of enriched uranium or reprocessed plutonium.

Breeding Ratio – the ratio between the average rate of fissile atom production and average rate of fissile atom consumption. Reactors with breeding ratio greater than 1.0 are considered breeder reactors as they produce (breed) more fissile material than they consume.

Core Spawning – The use of used nuclear fuel containing fissile material bred in one breed and burn reactor as the startup material for future reactor cores.

Equilibrium Core – A state where no enrichment or reprocessing is needed to sustain the fleet, and the isotopic composition is constant between cycles.

Used Nuclear Fuel (UNF) – In the context of this report, fuel that has been previously irradiated in one or more reactors but has not reached its operational limits and could potentially be reused to supply initial fissile inventory for a newly deployed reactor.

Spent Nuclear Fuel (SNF) – In the context of this report, fuel that has reached its operational limits and cannot be reused and will be directly disposed.

First Generation (S1) – A breed and burn reactor where the initial fissile inventory is supplied solely by low-enriched uranium.

Second Generation (S2) – A breed and burn reactor where the initial fissile inventory is supplied wholly or in part from UNF from S1 reactors.

Third Generation plus (S3+) – A breed and burn reactor where the initial fissile inventory is supplied from any combination of UNF from S1, S2, S3+ and LEU.⁶

Replacement Reactor – A breed and burn reactor deployed specifically to replace a reactor that has reached the end of its operational lifetime. The initial fissile inventory is provided by directly transferring the inventory of UNF from the reactor being replaced. This is required to sustain breed and burn operations indefinitely.

Significant Quantity (SQ) – The IAEA defines a significant quantity as the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded [IAEA,

⁶ This report only analyzes S1 & S2 reactors, but deployment scenarios may include S3+ reactors as the fleet is deployed.

2022]. Using values specified by the IAEA, for this study, one SQ would be 8 kg of Pu in irradiated fuel, 75 kg of U-235 in the LEU or DU (or 10 tonnes of NU or 20 tonnes of DU).

1.4. Report Overview

Section 2 discusses the relevant international safeguards approaches. These are the basis for the safeguards assessment of the breed and burn reactors

Section 3 describes the example breed and burn reactor design that was used in the safeguards analysis. There are many uncertainties in the future design(s) that are important to safeguard, but for this initial assessment a representative example was chosen that includes the key features that need to be considered.

Section 4 provides information on a representative once-through LWR and a representative SFR operated on a closed fuel cycle that recycles all uranium and plutonium and only requires depleted uranium additions to sustain operations. LWRs are representative of a large fraction of current reactors in operation and a traditional SFR is presented for comparison with the example breed and burn SFR.

Section 5 analyzes the deployment of a fleet of breed and burn reactors. The amount of UNF within the system and how it moves is a complex relationship driven by several considerations. This information is presented in the context of safeguards planning at the State level.

Section 6 assesses the safeguards challenges of breed and burn reactors. This is limited to the example reactor which captures most of the key aspects of an eventual breed and burn reactor fleet.

Section 7 provides the conclusions and insights of this report. It also provides the recommendations for research and future work required to address some of these challenges. Any key aspects not addressed in this report are identified, such as limiting this report to solid-fuel (there are liquid-fueled breed and burn alternatives) and no intermediate fuel processing (recladding of the UNF is an alternative if sufficient fuel lifetime cannot be achieved to attain breed and burn).

Appendix A provides the list of acronyms. Appendix B provides a detailed set of inventories for the example breed and burn reactor from the initial core through equilibrium operations. Appendix C provides additional details on the deployment modeling.

2. International Safeguards Approaches

Key Takeaways for International Safeguards

- Basic international safeguards definitions and concepts are introduced for use in the study, with implications for application to breed and burn reactors.
- Breed and burn reactor operations present two distinct challenges to spent fuel data management: (1) the maintenance of reliable assembly-level data following a transfer of UNF between operators and (2) the estimation of radionuclide content at extremely high burnups. These data will form the basis for operator nuclear material inventory declarations to the IAEA.
- The uncertainties associated with assembly level history data and radionuclide estimates will need to be well understood to allow the IAEA to draw conclusions and address any identified anomalies.

The purpose of international safeguards is to ensure the peaceful application of nuclear materials and activities. For non-nuclear weapon states party to the Nonproliferation Treaty (NPT), a Comprehensive Safeguards Agreement (CSA) enables a State to verifiably demonstrate proper declaration of all nuclear material and activities and that such activities are peaceful in nature. The IAEA's safeguards goals under a CSA are to be able to 1) detect diversion of declared nuclear material, 2) detect undeclared production or processing of material at declared facilities, and 3) detect undeclared nuclear material or activities in the state. At the facility level, comprehensive safeguards include the following elements:

- Nuclear material accountancy (NMA) and verification—the State reports information on nuclear material inventories and movements, and the verification of that information through facility inspections conducted by the IAEA;
- Design information verification—the reporting of information on the facility's design and characteristics, and verification of that information by IAEA inspectors; and
- Containment, surveillance, and monitoring—nuclear material control (NMC) measures to maintain a continuity of knowledge (CoK) on nuclear material and activities to facilitate verification efforts.

Application of these measures is determined by subsidiary arrangements between the State and the IAEA. The frequency and intensity of safeguards measures are determined by various factors such as the quantity, throughput, type, and form of material present in the facility. NMA is accomplished by dividing a facility into one or more material balance areas (MBA), where the quantity of material transferred in and out, as well as the overall physical inventory, can be determined. Key measurement points (KMPs) are established where either the flow or inventory of material can be readily measured; these may be at transfer points in between MBAs or within an MBA. Measurements taken by inspectors are designed to accomplish three detection goals: gross defect detection (diversion of whole items or batches), partial defect detection (diversion of a substantial fraction of an item or batch), or bias defect detection (diversion of a small portion of material in a flow stream, over a protracted timeframe).

Safeguarding a breed and burn fleet is a challenge with many complexities, as a single fleet involves multiple reactors and movement of fuel. The safeguards assessment is provided through a model fleet where UNF usage from one reactor (S1) to support the deployment of future reactors (S2), as described in the Assumptions, and further details regarding reactor design shall be described in the following sections. It is noted that transfers of UNF to support subsequent generations following S1 and S2 are conceptually similar and are excluded from evaluation in this report.

The focus of safeguards in a breed and burn fleet was identified to be most relevant at the reactor facility. Safeguards implementation has been studied at fast reactors, such as the Monju and Joyo reactors. Thus, safeguarding a fleet of breed and burn reactors will include techniques drawn from historical experience at these facilities. Data management and inventory reporting is an established practice for power reactors, particularly for special nuclear material (SNM) reporting requirements. A complete review of data management requirements and record maintenance for spent fuel management may be found in IAEA TECDOC-1519 [IAEA, 2006]. Data management for breed and burn reactors will also follow the established practices with emphasis on quality reporting passed from reactor to reactor, for safety, safeguards, security, and operational needs. Facility operators have the responsibility to maintain power history for each fuel assembly, along with the results of routine radionuclide content calculations for individual assemblies [IAEA, 2006]. Breed and burn reactor operations present two distinct challenges to spent fuel data management: the maintenance of reliable assembly-level data following a transfer of UNF between operators and the estimation of radionuclide content at extremely high burnups. The data will form the basis for operator nuclear material inventory declarations to the IAEA. Their associated uncertainties will need to be well understood to allow the IAEA to draw conclusions and address any identified discrepancies.

2.1. Material Balance Areas and Material Accountancy

The IAEA defines an MBA as an area where (a) the quantity of nuclear material in each transfer into or out of the MBA can be determined and (b) the physical inventory of nuclear material can be determined when necessary [IAEA, 2022]. The drawing and assignment of MBAs vary by facility and may differ by number and/or layout to accommodate facility operations, material flow, and storage (in bulk or item form). Defining MBAs provides facilities flexibility in meeting IAEA material accountancy goals. For example, high throughput facilities may have high values of uncertainty which could be made more manageable if separated into smaller components or processes with less throughput [Pillay, 1987]. A single MBA for the entire facility may also be used in conjunction with sufficient KMPs for flow and inventory [IAEA, 2017]. In general, the reactor in-vessel is considered one MBA while material storage areas and fuel handling facilities may be considered separate MBAs. As material is transferred into and out of MBAs, the facility can define locations for the KMPs where material flow measurements can be taken. They may also define KMPs where inventory measurements can be taken, using equipment such as load cells for gravimetric measurements.

2.2. IAEA Timeliness Detection Goals

One of the goals of the IAEA and international safeguards is the timely detection of material diversion and/or facility misuse. The techniques and tools described in this section contribute to high detection probabilities. The IAEA previously identified targeted detection times for detecting material diversion based on material categories, listed below [IAEA, 2001]. While these are no longer part of official IAE guidance, they are useful reference points.

- One month for unirradiated direct use material,
- Three months for irradiated direct use material,
- One year for indirect use material.

Direct use material is defined as, “nuclear material that can be used for the manufacture of nuclear explosive devices without transmutation or further enrichment.” Indirect use material is all material except direct use material [IAEA, 2022]. The detection goals are used for establishing the frequency of inspections and safeguards activities at facilities. The SFR ideal for breed and burn deployment contains inventories of indirect use nuclear material (fresh LEU or DU fuel) and irradiated direct use material (UNF). Therefore, the IAEA would develop a safeguards approach that could detect the diversion of one significant quantity or more of indirect use material within a calendar year and irradiated direct use material within three months.

2.3. Key Measurement Points

KMPs are locations where nuclear material is measured to determine material flow or inventory [IAEA, 2022]. KMPs are vital to safeguarding facilities and allow for NMA without disrupting facility operations. The placement of KMPs (and, in essence, MBA boundary definitions) also relies on the accuracy of chosen measurement systems to ensure minimal uncertainty in the material quantification measurements (as defined by IAEA’s International Target Values [IAEA, 2010]). KMPs placed in between MBAs are referred to as flow KMPs and contribute to reducing overall uncertainty. Inventory measurements, taken where material is not in movement, such as in material storage locations, are known as inventory KMPs.

2.4. Containment and Surveillance

As part of an effective NMC approach (which complements NMA techniques), containment and surveillance (C/S) measures are crucial when defining MBAs and KMPs. C/S supports CoK as material moves through a facility and prevents undetected access to nuclear material or undeclared facility operations. The concept of CoK is defined as a system of data or information regarding an item that is uninterrupted and authentic, which provides the IAEA with adequate insight to draw definitive conclusions that nuclear material is not being diverted from peaceful purposes [Blair, 2014]. Facilities are responsible for the material in their possession, but the IAEA must independently ensure that CoK is maintained. The major restriction applied to C/S measures is that they cannot interfere with or disrupt a facility’s normal operation and material flow [IAEA, 2001].

Containment can be accomplished with locks and seals. Locks prevent unauthorized access, while seals indicate unauthorized access or entry. Locks and seals can be found on IAEA monitoring equipment, stored nuclear material, or even specific facility equipment. Seals, such as a tamper-indicating device (TID), do not provide the same security as locks, but they are designed to indicate if an item or material has been accessed. Thus, a system of locks and seals deny and deter malicious access, respectively. TIDs come in two varieties: passive and active. Passive TIDs require no source of power but cannot indicate the exact time of access. They can, however, provide a window of time corresponding to an inspection schedule. Passive TIDs may include adhesives, cap seals, pull-tight seals, etc. A comprehensive list of relevant C/S equipment can be found in the IAEA Safeguards Techniques and Equipment handbook [IAEA, 2011]. Conversely, active TIDs require a form of power, either external or internal. Active TIDs may be limited in deployment by distance from a power source or battery lifetime. However, active TIDs may precisely detect time of access and could be connected to an alert or alarm system. Regardless of the seal type, they must all be regularly inspected and evaluated for integrity and be uniquely

identifiable. These measures may also be applied to prevent tampering of safeguards equipment used for nuclear material accounting and control (NMAC) purposes, such as load cells for monitoring material quantities, video surveillance equipment, and other measurement devices.

2.5. Non-Destructive Analysis and Destructive Analysis

Material content of traditional LWR fuel may be verified by the facility operators at any time by sampling the fissile content of the fuel assemblies or sampling from a container (in which fuel assemblies or SNM in general is stored) prior to sealing and shipment. The goal is to ensure special fissionable material quantities align with facility declarations and that diversions with or without substitutions did not occur. Random sampling of material informs the detection bias, which may occur if small amounts of material were diverted over time. Thus, applying physical and chemical analysis techniques ensures the best possible accuracy.

Techniques used for non-destructive analysis (NDA) by IAEA inspectors include a wide range of gamma spectrometers, and passive and active neutron counters. Specific equipment is dictated by the fuel characteristics, such as size, geometry, and radionuclide content; the allotted space capacity within the facility; and the available power source [IAEA, 2011]. NDA equipment may have large geometries and be capable of running continuously for active monitoring or smaller for on-site verification. Samples of fuel may also be retained from a container by inspectors for destructive analysis (DA) in a laboratory setting. These samples can be measured for elemental and isotopic composition. Although DA provides a more accurate report, it requires extensive time and resources while NDA may be conducted in-situ at the facility. Detailed descriptions of NDA and DA equipment and techniques may be found in the IAEA Safeguards Techniques and Equipment guidance [IAEA, 2011].

3. Example Breed and Burn Reactor Design

Key Takeaways for International Safeguards
• An example reactor concept was evaluated for this study to produce representative data for safeguards analysis, including nuclear material inventories at the beginning and end of reactor cycles. • Operational design of the example facility is described along with the associated safeguards considerations necessary, per IAEA guidance. • In an equilibrium cycle, the example concept discharges 12 assemblies each containing approximately 5 SQs of nuclear material (predominantly Pu). • The example concept will contain a core inventory of greater than 1200 SQs of nuclear material. As a point of comparison, that represents 0.75% of all Pu contained in irradiated nuclear fuel under IAEA safeguards as of 2020.

This section provides the important high-level characteristics of the example reactor concept based on Argonne’s experience with breed and burn SFR designs. Additional detailed results are included in Appendix B. In general, the core physics will drive the general evolution of fuel isotopic compositions. Practical engineering limitations, primarily the achievable power density that can be safely operated, will impact material inventories and mass flow of the fuel. Inventory per gigawatt-electric (GWe), individual assembly size, and average annual fuel discharged will be the most uncertain parameters. The concentration of the major fissile isotopes (i.e., U-235 and Pu-239) accumulated per unit of initial heavy metal⁷ will have lower uncertainties because of physics constraints. These constraints include the need for the reactor to remain critical, the need to operate at the necessary breeding ratio, and the necessary lifetime to achieve equilibrium breed and burn operation.

The example reactor core concept is a 3000 megawatt-thermal (MWth) / 1200 megawatt electric (MWe) SFR using 90 wt% U, 10 wt% Zr metallic fuel (U-10Zr) with fuel density of 15.6 g/cm³ described by Kim [Kim, 2010]. The core configuration can be seen in Figure 2, which consists of 408 hexagonal fuel assemblies each with 20.13 cm modeled flat-to-flat distance. The assembly volume fractions of fuel, initial gap, structure, and coolant are 41%, 19%, 15%, and 25%, respectively. The inner, middle, and outer enrichment zones each contain 108, 126, and 126 fuel assemblies, and their fresh startup fuels use 9.1%, 11.13%, and 14.15% enriched uranium in the driver (enriched region) axial region, respectively. Axially, each fuel assembly consists of 130.33 cm driver fuel in the middle axial region, with 43.44 cm upper and lower axial blankets using depleted uranium (0.25% U-235). Radially, the core is surrounded by one ring of 48 blanket fuel assemblies containing depleted uranium (0.25% U-235) along the entire 217 cm length, one ring of HT9 steel reflector assemblies, and finally one ring of B₄C-containing shield assemblies.

⁷ Initial heavy metal denotes the mass of actinides and fission products which is the initial mass of heavy metal atoms prior to start of irradiation.

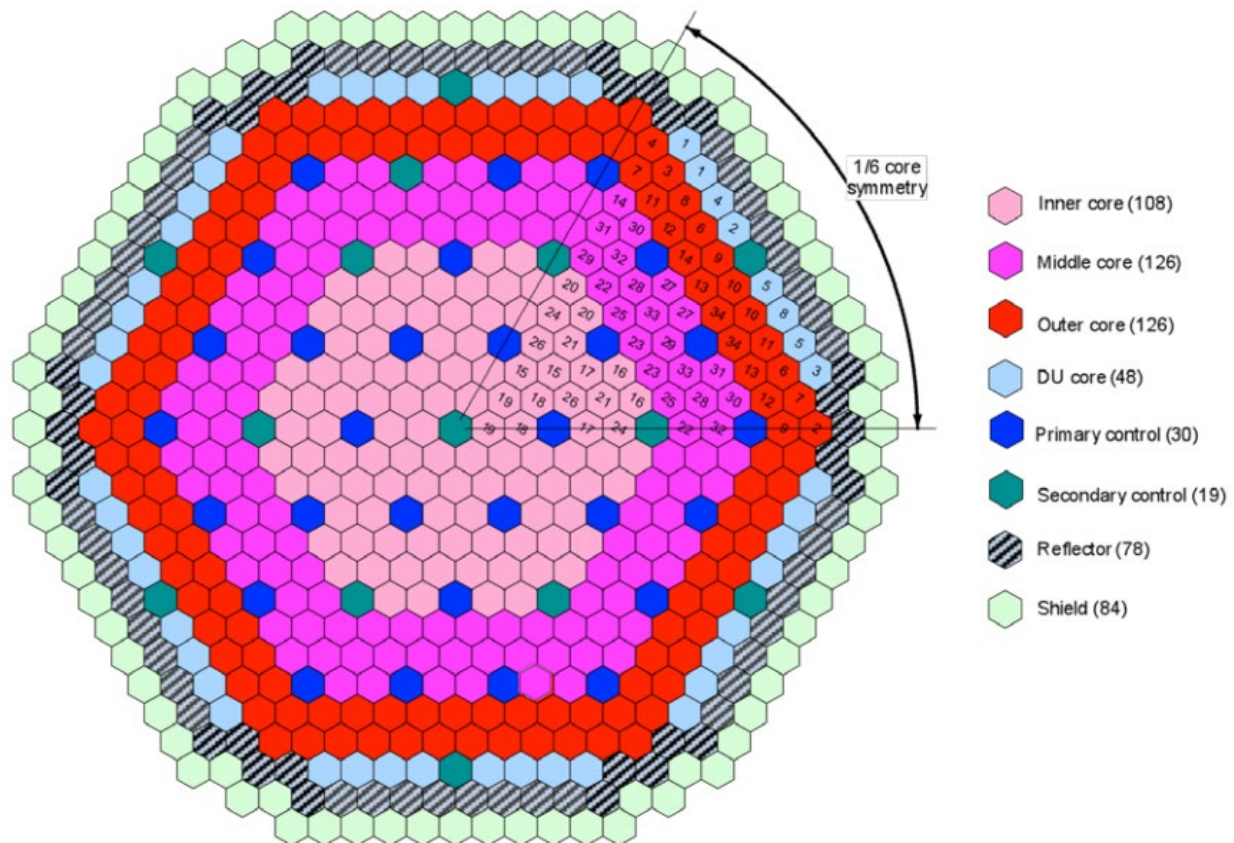


Figure 2. Radial core layout of example breed and burn reactor concept. [Kim, 2010]

The operating cycle length of the reactor is 493 effective full power days with a shutdown time of 54.5 days, corresponding to a 1.5-year cycle length with an average capacity factor of 90%. Between each cycle, 12 UNF assemblies are discharged from the outer active core zone. The remaining partially burned assemblies are shuffled between the radial blanket and the three active core zones, and 12 fresh DU assemblies are added to the radial blanket.

The evolution from the initial enriched uranium core to the equilibrium core (with no isotopic evolution between cycles) results in significant changes in material inventories and in the composition of fuel discharged from the reactor at the end of each cycle. The results of this example core design are given in the following figures and tables. Figure 3 shows the evolution of the neutron multiplication factor or k_{eff} with burnup over 100 cycles. This example core was not optimized for the transition cycles and thus the k_{eff} will be reduced for both safety (to limit excess reactivity for accident scenarios) and economic efficiency (reduce fissile requirements). The reactivity swing of early cycles is expected to be limited to that of the eventual equilibrium operations. This still provides reasonable estimates of inventories with larger sources of uncertainty, such as power density, which will have bigger impact on reactor and system inventories for a given power level.

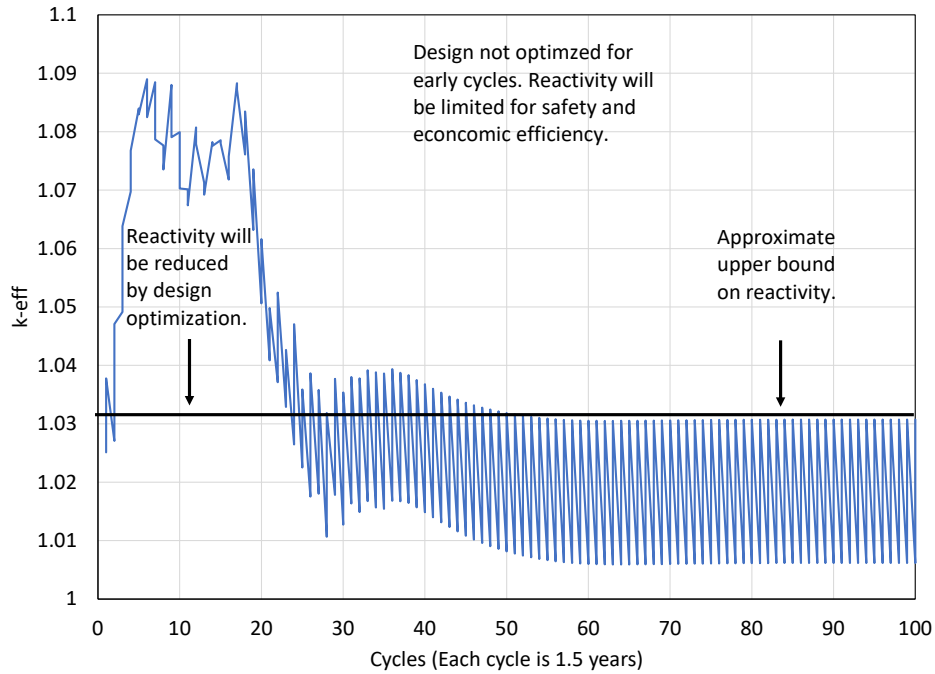


Figure 3. Evolution of k_{eff} for 100 cycles.

As the enriched uranium starter fuel is depleted and gradually discharged, Figure 4 conveys the U-235 inventory decreasing steadily. Simultaneously, Pu-239 is rapidly bred over the first 10 cycles and eventually levels off. The evolution of k_{eff} and Pu-239 inventory indicate that equilibrium operation is achieved after around 75 effective full power years (EFPY), corresponding to 50 operating cycles. During each equilibrium cycle, 370 kg of Pu-239 is bred, resulting in a cycle reactivity swing of 2360 percent mille (pcm)⁸.

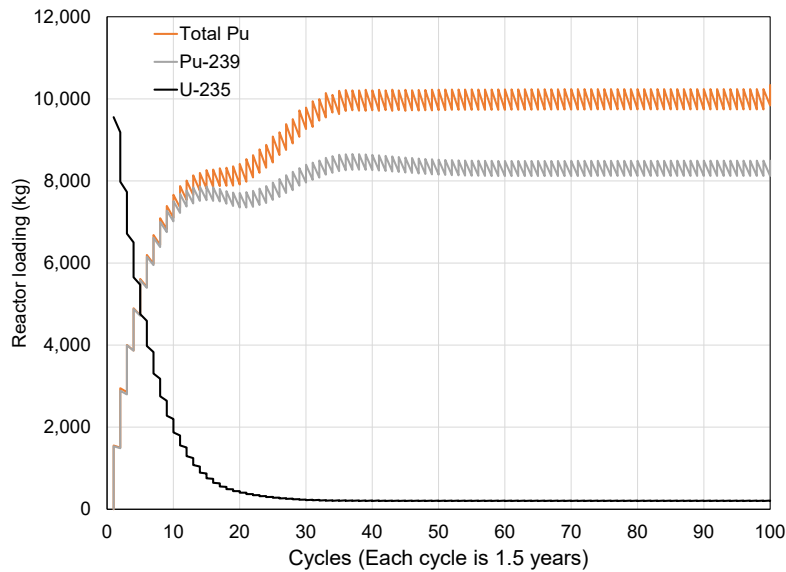


Figure 4. Evolution of fissile isotope core inventories for 100 cycles.

⁸ Percent mille is a commonly used unit of reactivity equal to $10^{-5} \Delta k_{\text{eff}}/k_{\text{eff}}$.

Table 2. Core Inventory for Example Breed and Burn Reactor Concepts

	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Equilibrium Cycle	
	BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC
U-235 (kg)	10,978	9,553	9,185	7,980	7,731	6,717	6,504	5,654	212	200
Total U (kg)	177,641	174,570	174,685	171,711	171,904	169,099	169,387	166,739	149,847	147,814
Pu-239 (kg)	0	1,537	1,487	2,889	2,798	3,995	3,865	4,878	8,123	8,492
Total Pu (kg)	0	1,552	1,502	2,948	2,855	4,123	3,989	5,094	9,840	10,337
Np-237 (kg)	0	13	13	32	31	55	53	78	62	65
Am (kg)	0.000	0.002	0.003	0.029	0	0	0	0	44.3	49.3
Significant Quantities	146	321	310	475	460	605	585	712	1233	1295

Table 2 provides a summary the average properties for the discharged assemblies at the beginning of cycle (BOC) and end of cycle (EOC) for the first four cycles and the equilibrium cycle (estimated at 50 cycles). Figure 5 shows the evolution of the average content of the 12 UNF assemblies discharged at the end of each cycle: mass of fissile material content (including U-235, Pu-239, and total Pu) in kilogram per assembly on the left and in significant quantities per assembly corresponding to the right ordinate. In the first few cycles, the discharged assemblies have low burnup and contain primarily U-235 with information for more cycles provided in Appendix B. As plutonium is bred in the core and equilibrium is approached, the discharged assemblies eventually attain a burnup of 29.1% fissions per initial metal atoms (FIMA) and contain primarily Pu-239. Trace amounts of Neptunium (Np)-237 and Americium (Am) are also discharged at equilibrium. The core Pu inventories in the example reactor are extremely high, compared to past commercial LWR experience. As a point of comparison, an equilibrium core from a single reactor at EOC has an inventory that represents 0.75% of all plutonium contained in irradiated nuclear fuel under IAEA safeguards as of 2020, as generated by 263 power reactors over decades of operation [IAEA, 2021].

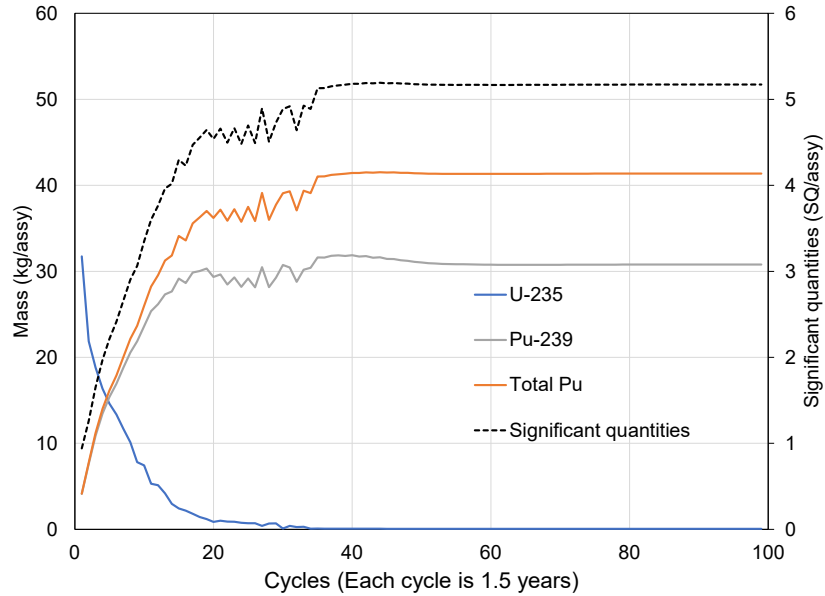


Figure 5. Evolution of average discharged assembly content

Table 3. Average Used Nuclear Fuel Discharge for Example Breed and Burn Reactor Concepts

	End of:				
	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Eq. Cycle
Assemblies Discharged ⁹	12	12	12	12	12
U-235 (kg/assy)	31.7	21.9	18.8	16.4	0.0
Total U (kg/assy)	425.8	419.2	411.4	404.5	266.0
Pu-239 (kg/assy)	4.1	7.6	10.9	13.5	30.8
Total Pu (kg/assy)	4.2	7.8	11.2	14.0	41.4
Np-237 (kg/assy)	0.0	0.1	0.2	0.2	0.3
Am (kg/assy)	0.000	0.000	0.000	0.001	0.495
Significant Quantities (SQ/assy)	0.94	1.26	1.65	1.97	5.18
Fuel Burnup (fraction HM fissioned)	1.2%	1.9%	2.9%	3.8%	29.1%

3.1. Summary of Material Movements

The example breed and burn reactor has a total of 408 fuel assemblies in the active core region of the reactor during operations. During a typical refueling outage, 12 UNF assemblies are removed from the active core region and placed into in-vessel storage (IVS). IVS, which is a low flux region within the reactor vessel used to allow UNF to cool (i.e., decay), provides a less challenging alternative for fuel handling compared to out-of-vessel storage. This is the approach for handling UNF in this study and is based on relevant methods for SFRs. The number of cycles the fuel is stored in-vessel is a design and operational decision depending on the reactor that eventually facilitates the transfer to dry storage. For this example, 3 cycles of in-vessel storage (4.5 years) were adopted, which is appropriate for fuel that may potentially

⁹ Discharged fuel is depleted uranium with 1.1 kg U-235/Assy and 435.2 kg total U per Assembly.

be placed back into sodium for future use in an SFR. There is likely to be significant fuel shuffling of the remaining UNF in the core in addition to the loading of 12 unirradiated DU assemblies during each refueling. Prior to each refueling, the 12 unirradiated DU assemblies would have been received in the previous cycle, inspected, and placed into unirradiated fuel storage to be loaded.

During the 4th and subsequent refueling outages, based on the assumption of 3 cycles in the IVS, the 12 UNF assemblies are transferred from IVS to the ex-vessel UNF storage which is assumed to be a highly shielded hot cell that can store UNF. The UNF assemblies remain in storage until they are sent via heavily shielded spent fuel casks to another breed and burn SFR for continued utilization, to onsite dry storage away from the reactor, or to a repository for disposal.

All fuel movement into and out of the reactor core will be conducted by the fuel transfer system. This will be a system of cranes to load and unload the transfer cask that will be used to move the fuel between storage, into containment, and connection with the fuel transfer port. From there, the in-vessel transfer machine will load or unload the transfer cask and place or remove the assemblies within the reactor. This is described in the ABR1000 Reference Concept Report [ABR, 2007]. The transfer machines were also present at the Monju and Joyo reactors and will be discussed further in the safeguards analysis section.

For the initial core load, the reactor facility must receive all 408 fuel assemblies, perform inspections, and subsequently transfer into the reactor core before initial operations can begin. For a S1 reactor, this loading consists of 360 LEU fuel assemblies and 48 DU fuel assemblies. For a S2 reactor, some mix of unirradiated LEU and DU fuel assemblies is required, as well as UNF assemblies previously irradiated in one (or more) S1-type reactors. For a reactor that is replacing another reactor that has reached the end of its operating life, the initial core would be composed of 396 UNF assemblies from the reactor that it is replacing, as well as 12 unirradiated UNF assemblies. There will be no unirradiated LEU fuel required for the replacement reactor.

Figure 6 shows all the potential fuel flow paths for a breed and burn reactor in the S2 generation. An S1 reactor and a replacement reactor will largely follow a similar concept, except there would be no input UNF flow path and all input UNF would come from the reactor being replaced in the replacement reactor. This figure captures all the fuel assembly flow paths that will be required for any deployment scenario, but it must be noted that a specific reactor may not have an input of LEU fuel assemblies (such as the case where all initial fissile content is from UNF) or an input of UNF assemblies (where all initial content comes from LEU fuel). Fuel essentially exists in four separate areas throughout the operational lifetime at the reactor. The safeguards implications of the fuel storage areas are discussed in the safeguards analysis section.

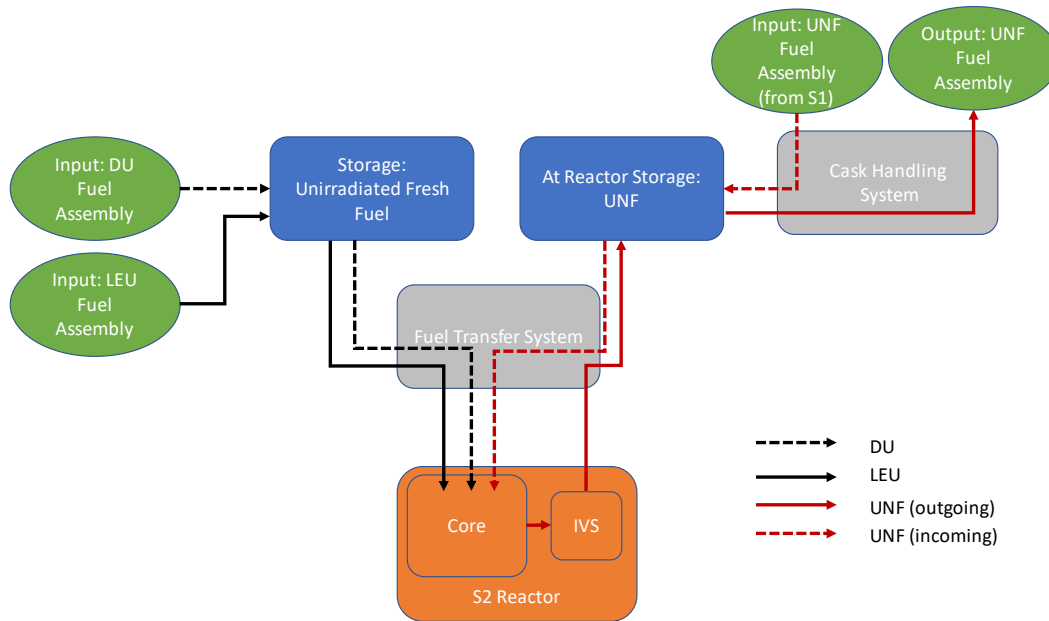


Figure 6. All potential fuel flow paths for a second-generation (S2) example breed and burn reactor.

For an S1 reactor, Table 4 describes the assembly inventory by number of assemblies, type of assemblies, and where those assemblies are during a particular operational step of a particular reactor cycle. Table 5 provides the assembly information for an S2 reactor.

Table 4. Sequential representation of assembly inventory by type, cycle, and operational step for an S1 breed and burn SFR.

Cycle	Step	At-Reactor Storage		Reactor	
		Unirradiated Fuel	UNF	Core	IVS
1	Fresh Fuel Receipt	360 LEU / 48 DU			
	Core Load			360 LEU / 48 DU	
	Operations			408 UNF	
2	Fresh Fuel Receipt	12 DU		408 UNF	
	Core Unload	12 DU		396 UNF	12 UNF
	Core Load			12 DU / 396 UNF	12 UNF
	Operations			408 UNF	12 UNF
3	Fresh Fuel Receipt	12 DU		408 UNF	12 UNF
	Core Unload	12 DU		396 UNF	24 UNF
	Core Load			12 DU / 396 UNF	24 UNF
	Operations			408 UNF	24 UNF
4	Fresh Fuel Receipt	12 DU		408 UNF	24 UNF
	Core Unload	12 DU		396 UNF	36 UNF
	Core Load			12 DU / 396 UNF	36 UNF
	Operations			408 UNF	36 UNF
5	Fresh Fuel Receipt	12 DU		408 UNF	36 UNF
	Core Unload	12 DU	12 UNF	396 UNF	36 UNF
	Core Load		12 UNF	12 DU / 396 UNF	36 UNF
	Operations		12 UNF	408 UNF	36 UNF
6	Fresh Fuel Receipt	12 DU	12 UNF	408 UNF	36 UNF
	Core Unload	12 DU	24 UNF	396 UNF	36 UNF
	Core Load		24 UNF	12 DU / 396 UNF	36 UNF
	Operations		24 UNF	408 UNF	36 UNF

Table 5. Sequential representation of assembly inventory by type, cycle, and operational step for an S2 breed and burn SFR.

Cycle	Step	At-Reactor Storage		Reactor	
		Unirradiated Fuel	UNF	Core	IVS
1	Fresh Fuel Receipt	240 LEU / 48 DU	120 UNF		
	Core Load			240 LEU / 48 DU / 120 UNF	
	Operations			408 UNF	
2	Fresh Fuel Receipt	12 DU		408 UNF	
	Core Unload	12 DU		396 UNF	12 UNF
	Core Load			12 DU / 396 UNF	12 UNF
	Operations			408 UNF	12 UNF
3	Fresh Fuel Receipt	12 DU		408 UNF	12 UNF
	Core Unload	12 DU		396 UNF	24 UNF
	Core Load			12 DU / 396 UNF	24 UNF
	Operations			408 UNF	24 UNF
4	Fresh Fuel Receipt	12 DU		408 UNF	24 UNF
	Core Unload	12 DU		396 UNF	36 UNF
	Core Load			12 DU / 396 UNF	36 UNF
	Operations			408 UNF	36 UNF
5	Fresh Fuel Receipt	12 DU		408 UNF	36 UNF
	Core Unload	12 DU	12 UNF	396 UNF	36 UNF
	Core Load		12 UNF	12 DU / 396 UNF	36 UNF
	Operations		12 UNF	408 UNF	36 UNF
6	Fresh Fuel Receipt	12 DU	12 UNF	408 UNF	36 UNF
	Core Unload	12 DU	24 UNF	396 UNF	36 UNF
	Core Load		24 UNF	12 DU / 396 UNF	36 UNF
	Operations		24 UNF	408 UNF	36 UNF

The assembly information is given in Table 6 for a replacement generation reactor, where two types of assemblies with of various fissile content will exist:

- A DU assembly which is any unirradiated fuel assembly that contains depleted uranium; and
- A UNF assembly is any fuel assembly that has been irradiated for any amount of time, regardless of initial enrichment or composition.

Table 6. Assembly inventory by type, cycle, and operational step for replacement breed and burn SFR.

Cycle	Step	At-Reactor Storage		Reactor	
		Unirradiated Fuel	UNF	Core	IVS
1	Fresh Fuel Receipt	12 DU	396 UNF		
	Core Load			12 DU / 396 UNF	
	Operations			408 UNF	
2	Fresh Fuel Receipt	12 DU		408 UNF	
	Core Unload	12 DU		396 UNF	12 UNF
	Core Load			12 DU / 396 UNF	12 UNF
	Operations			408 UNF	12 UNF
3	Fresh Fuel Receipt	12 DU		408 UNF	12 UNF
	Core Unload	12 DU		396 UNF	24 UNF
	Core Load			12 DU / 396 UNF	24 UNF
	Operations			408 UNF	24 UNF
4	Fresh Fuel Receipt	12 DU		408 UNF	24 UNF
	Core Unload	12 DU		396 UNF	36 UNF
	Core Load			12 DU / 396 UNF	36 UNF
	Operations			408 UNF	36 UNF
5	Fresh Fuel Receipt	12 DU		408 UNF	36 UNF
	Core Unload	12 DU	12 UNF	396 UNF	36 UNF
	Core Load		12 UNF	12 DU / 396 UNF	36 UNF
	Operations		12 UNF	408 UNF	36 UNF
6	Fresh Fuel Receipt	12 DU	12 UNF	408 UNF	36 UNF
	Core Unload	12 DU	24 UNF	396 UNF	36 UNF
	Core Load		24 UNF	12 DU / 396 UNF	36 UNF
	Operations		24 UNF	408 UNF	36 UNF

While a specific example for an S2 reactor fuel is not explicitly shown, it is known that a mix of LEU, DU, and UNF is likely for the initial core load. The exact mixture will remain unique for each S2 reactor. For demonstration purposes, it was assumed 120 UNF assemblies would replace 120 LEU assemblies in the S1 reactor (as shown in Table 4), but realistic operations will see some variance in the minimum number of UNF assemblies to justify the added complexity of utilizing UNF. In an extreme case, the entire fissile inventory could be UNF if available.

These examples have shown the evolution of fuel over 6 cycles. The flow of assemblies will remain the same for the life of the reactor regardless of generation. The total number of UNF assemblies accumulated in onsite storage will depend on operational considerations and when UNF is either transferred to another reactor for reuse or sent offsite for storage or disposal. At the end of the reactor life, the fuel in the last core will be stored for some time in place in the reactor to allow the fuel to decay. The entire core may then be offloaded through onsite storage and the bulk of the UNF (396 of the 408 fuel assemblies in the example reactor) will be transferred to the replacement reactor. The remainder of the UNF will be stored on site until transferred to offsite storage or disposal.

4. Comparison of Once-Through Light Water Reactor and Closed Fuel Cycle Sodium-Cooled Fast Reactor to Breed and Burn Fleet

Key Takeaways for International Safeguards

- SNM inventories in comparable LWR (specifically, PWR) and SFR fuel cycles are analyzed and compared.
- On a power normalized basis, the end of cycle breed and burn SFR used nuclear fuel contains nearly seven times as much SNM as an equivalent PWR, and over 40% more than an equivalent SFR not operating as a breed and burn reactor.
- The SNM content of a breed and burn reactor assembly increases from 0.02 SQ at initial loading (as DU) to over 5 SQ at discharge, representing a greater than 250x increase in SNM over its lifetime. This is considerably higher than SFR blanket fuel (20x increase in SNM content) or LWR fuel (6% increase).
- Very high levels of SNM production in breed and burn reactors will test the limits of inventory tracking based on depletion analysis. Estimating the SNM content in assemblies discharged from a breed and burn reactor in an equilibrium cycle to within 1 SQ will require 1.6% accuracy in core depletion models.

There are many differences in design, operation, and deployment between the example breed and burn SFR and typical once-through light water reactors or traditional closed-cycle SFRs. These areas of difference may raise new issues and challenges that have not previously been considered, or where there are large differences, prompt questions on the application of international safeguards to breed and burn SFRs. This study thus identifies areas where there are notable differences and discuss the challenges relevant to the unique features of SFRs in the breed and burn fleet. One such difference and subsequent challenge concerns the use of liquid sodium and the implication on relevant techniques.

Two representative reactors are provided for comparison purposes. The equilibrium system data is provided from the Nuclear Fuel Cycle Evaluation and Screening, a comprehensive study of nuclear fuel cycle alternatives [Wigeland, 2014]. The first describes the behavior of a pressurized-water reactor (PWR) fueled with LEU operated on an open fuel cycle. The spent nuclear fuel (SNF)¹⁰ is discharged from the reactor, stored for some time, and then directly disposed as intact SNF. Figure 7 shows the mass flows for this cycle referred to as EG01 by Wigeland. This represents technology currently under safeguards worldwide. For initial deployment, a full core of LEU with a somewhat lower average enrichment than the equilibrium refueling is utilized for new reactor capacity.

¹⁰ Irradiated reactor fuel is referred to as both SNF and UNF in this paper. As stated in Subsection 1.3, SNF refers to irradiated fuel intended for disposal and UNF refers to irradiated fuel intended for reuse or recycling.

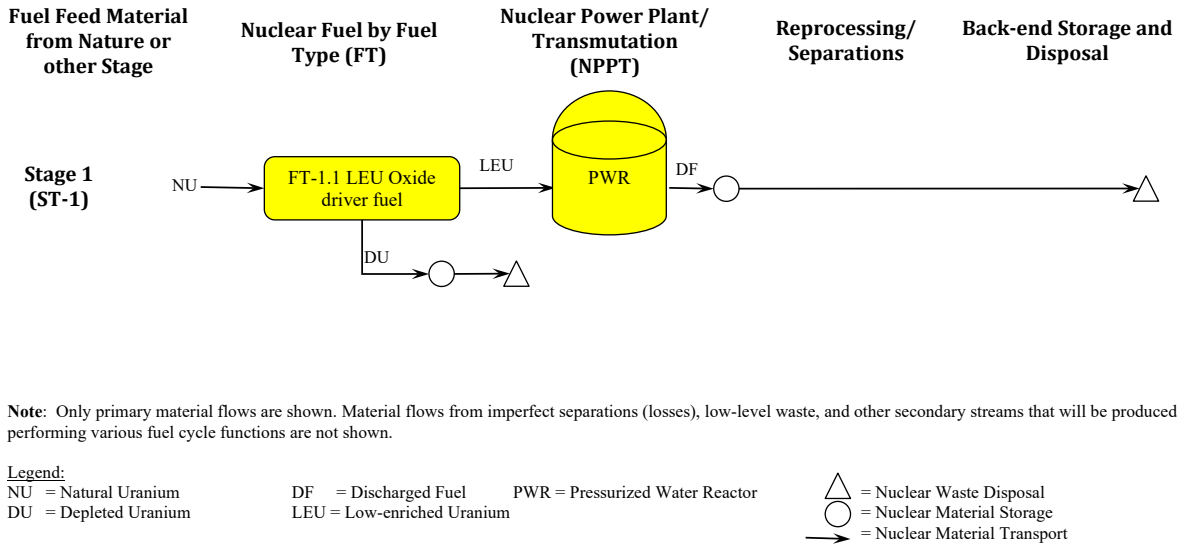


Figure 7. Mass flow for PWR LEU base case (EG01) [Wigeland, 2014].

The second representative reactor is an SFR operating on a closed fuel cycle with the reactor breeding ratio designed to “breakeven,” or breed just enough excess Pu to compensate for losses during reprocessing and fuel fabrication. All uranium and plutonium in the UNF are recycled and blended with unenriched uranium, most likely DU, to replace the heavy metal fissioned or lost during reprocessing and fuel fabrication. Figure 8 shows the mass flows for this cycle referred to as EG23 by Wigeland. This represents a common approach for deploying a fleet of SFRs and the fuel cycle that would be utilized.

Similar to the two scenarios identified in Section 1.2 for breed and burn deployment, there are two deployment options for the breakeven SFR. The initial fissile source for the first core must be either LEU or Pu recycled from an external source (e.g., LWR UNF). Additional fissile material is required for the first few refueling cycles until the cycle can be closed and the Pu can be obtained from recycled UNF. For direct comparison to the example breed and burn SFR deployment scenarios, LEU fuel was assumed for the first cycle.

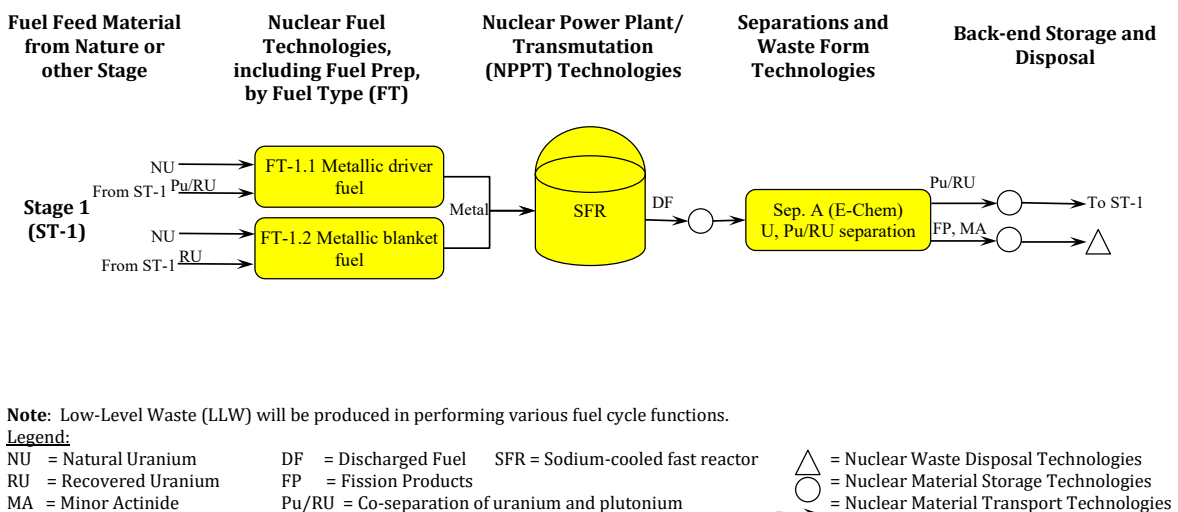
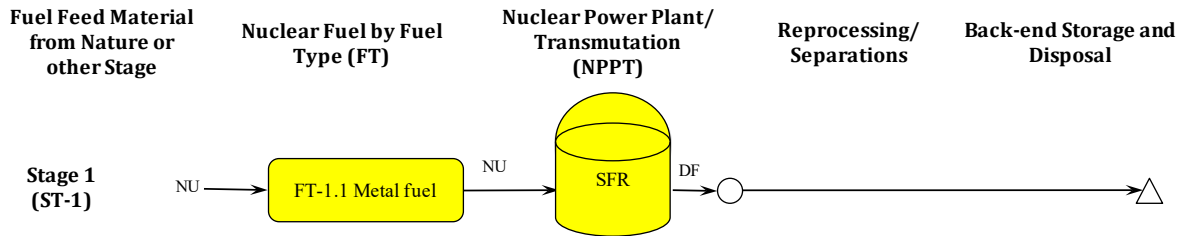


Figure 8. Mass flow for continuous Pu recycle in SFR (Metallic Fuel) (EG23) [Wigeland, 2014].

Figure 9 shows the mass flows for the breed and burn SFR referred to as EG04 by Wigeland. Thus, key differences can be seen regarding the true equilibrium state system. Likely, the natural uranium would only be used in the future after all DU has been consumed. Most importantly, the initial deployment of the system will require an external fissile source which is LEU for the example system.



Note: Only primary material flows are shown. Material flows from imperfect separations (losses), low-level waste, and other secondary streams that will be produced in performing various fuel cycle functions are not shown.

Legend:

NU = Natural Uranium

DF = Discharged Fuel

SFR = Sodium-cooled Fast Reactor

△ = Nuclear Waste Disposal
 ○ = Nuclear Material Storage
 → = Nuclear Material Transport

Figure 9. Mass flow for breed and burn SFR without separation (EG04) [Wigeland, 2014].

To illustrate the relevant differences, Table 7 summarizes the representative core inventories for the PWR, SFR, and breed and burn SFR operating under equilibrium conditions. Table 8 provides the same information normalized by reactor power. It can be seen that the SNM content of the SFR core, even at a lower total power, is considerably higher than the PWR. The total end of cycle SFR SNM content, expressed in significant quantities, is 94% higher than a PWR. On a power normalized basis, the end of cycle SFR SNM content is 380% higher than the PWR. This is primarily a result of the need for higher enrichments in a fast spectrum. The difference between the breed and burn SFR and the PWR is even more pronounced. On a power normalized basis, the end of cycle breed and burn SFR SNM content is 590% higher than the PWR and 43% higher than the SFR not operating as a breed and burn. Within a single cycle, the SNM inventory of the breed and burn reactor increases by ~5%, compared to 2% for a PWR and less than 0.5% for an SFR. For a breed and burn reactor, this represents a swing of 62 SQs in a single equilibrium cycle.

Table 7. Core Inventory for example PWR and SFR concepts.

	PWR (990 MWe)		SFR (400 MWe)		Breed and Burn SFR (at Equilibrium) (1,200 MWe)	
	BOC	EOC	BOC	EOC	BOC	EOC
U-235 (kg)	2,709	1,683	24	21	212	200
Total U (kg)	86,828	84,898	16,563	16,208	149,847	147,814
Pu-239 (kg)	180	360	1,637	1,643	8,123	8,492
Total Pu (kg)	354	709	2,395	2,407	9,840	10,337
Np-237 (kg)	19	38	3	5	62	65
Am (kg)	8	15	5	8	44.3	49.3
Significant Quantities	153	156	300	302	1233	1295

Table 8. Power-normalized core inventory for example PWR and SFR concepts.

	PWR (990 MWe)		SFR (400 MWe)		Breed and Burn SFR (at Equilibrium) (1,200 MWe)	
	BOC	EOC	BOC	EOC	BOC	EOC
U-235 (kg/GWe)	2,736	1,700	59	53	177	167
Total U (kg/GWe)	87,705	85,756	41,408	40,520	124,873	123,178
Pu-239 (kg/GWe)	182	364	4,092	4,109	6,769	7,077
Total Pu (kg/GWe)	358	716	5,988	6,017	8,200	8,614
Np-237 (kg/GWe)	19	38	7	12	52	54
Am (kg/GWe)	8	16	13	21	37	41
Significant Quantities/GWe	154	157	751	754	1,028	1,079

Table 9 shows information of the assemblies including average number of assemblies cycled through each reactor in an average year and the composition of those assemblies. The PWR assemblies contain more heavy metal in terms of mass but utilizes lower enrichment thus resulting in approximately half the SQs of material per assembly than for the Pu-bearing SFR driver fuel (and about the same as the discharged blanket fuel). All fuel assemblies will have physical dimension of similar size with an overall length in range of about 4 m for the PWR to around 5-6 m for the SFR assembly. SFR assemblies are ducted assemblies which are much more complex design which makes disassembly and pin removal more challenging [Dubberley, 2000].

Table 9. Assembly information for example PWR and SFR concepts.

	PWR (990 MWe)		SFR (400 MWe)				Breed and Burn SFR (at Equilibrium) (1,200 MWe)	
			Driver		Blanket			
	Charge	Discharge	Charge	Discharge	Charge	Discharge	Charge	Discharge
Average # assemblies per year	35	35	38	38	5	5	12	12
U-235 (kg/assembly)	23.8	4.2	0.0	0.0	1.0	0.5	1.5	0.1
Total U (kg/assembly)	565.3	528.5	86.7	78.6	138.4	127.4	434.7	266.0
Pu-239 (kg/assembly)	⁻¹¹	3.4	10.6	9.9	-	6.9	-	30.8
Total Pu (kg/assembly)	-	6.8	15.6	14.9	-	7.4	-	41.4
Np-237 (kg/assembly)	-	0.4	-	0.1	-	0.0	-	0.3
Am (kg/assembly)	-	0.1	-	0.1	-	0.0	-	0.495
Significant Quantities (SQ/assembly)	0.95	1.01	1.95	1.87	0.04	0.94	0.02	5.18
Approximate assembly dimensions – pitch ¹² x length (m)	0.2 x 4		0.2 x 5				0.2-0.3 x 5-6	

This analysis thus demonstrates the disparity in SNM content between the breed and burn SFR compared to the PWR and the SFR. The SNM increases from 0.02 SQ at initial loading (as DU) to over 5 SQ at discharge, representing a greater than 250x increase in SNM over its lifetime. This is considerably higher than SFR blanket fuel (a 20x increase in SNM content) or LWR fuel (a 6% increase). The very high levels of SNM production in breed and burn reactors will test the limits of inventory tracking based on depletion analysis alone. Breed and burn fleet operators must subsequently estimate the Pu content to within 1.6% to remain within 1 SQ as required for discharged fuel assemblies. This is a reasonable goal, but will require extensive nuclear data validation and rigorous management of the operational history of individual assemblies.

¹¹ “-” Denotes none. “0.0” indicates less than 0.05. All values per assembly.

¹² Pitch is the distance from flat to flat for square PWR assemblies and hexagonal SFR assemblies.

5. Example Deployment of a Fleet of Breed and Burn Reactors

Key Takeaways for International Safeguards
- Fuel cycle simulation code, DYMOND, is used to estimate nuclear material production rates, inventories, and rates of nuclear material transfers in a breed and burn fleet deployment. This analysis can be used for safeguards planning purposes at the State-level. - The SNM inventory in the system per unit of power produced is strongly correlated with the minimum time before a UNF assembly is reused; however, the equilibrium mass of UNF required for reactor startup decreases with longer cooling times. - The rate of SNM shipments between reactors is strongly correlated with the rate of reactor deployment. - UNF assemblies may be transferred between as many as 8 different reactors in aggressive deployment scenarios.

The deployment of fleets of breed and burn reactors presents unique safeguards challenges given the transport of UNF between reactor sites, and the large inventories of relatively high-quality direct use nuclear material built up. The manner of the deployment for the breed and burn reactors (and the fuel cycle) must consider the impact the quantities of materials will present in a reactor fleet, the complexity of the fuel burn histories, and the frequency of transport and receipts. Prior sections of this report focused on safeguards implementation at the facility level. This section shall evaluate the dynamics of a full-scale breed and burn reactor fleet deployment, including different deployment scenarios that may alter safeguards planning assumptions at the State-level and consider how deployment scenario may impact the uncertainties associated with nuclear material quantities.

The IAEA develops a State-level safeguards approach in which the plausible diversion paths by which nuclear material suitable for use in a nuclear weapon or nuclear explosive device can be acquired are analyzed. As the example reactor core analyses have shown, the breed and burn reactors present extremely high quantities of Pu. Deployment of a breed and burn reactor fleet has the potential to complicate IAEA safeguards planning efforts. Minor variations in how a reactor fleet is deployed can have major implications for the total quantity of nuclear material present in the State. While safeguards-by-design is traditionally limited to facility-level design decisions, there also exists an opportunity to design a breed and burn fleet deployment capable of reducing the overall nuclear material available in the State and thus reduce the frequency of UNF transfers. Projections using the methods and tools described in this section can help identify (at the onset of breed and burn reactor deployment) strategies that forestall safeguards implementation challenges.

During transition, many tons of UNF with compositions more like those of the assemblies discharged in the first few cycles (e.g., higher U concentration, lower Pu concentration than equilibrium fuel) will accumulate. To better understand these challenges and the impact of the transition characteristics, two fleet-level case studies were conducted. The first scenario is analogous to a “real-world” deployment. This scenario describes deployment of a fleet of SFRs to replace an existing LWR nuclear capacity at the end of deployment lifetime. The existing LWR fleet in the scenario has the same characteristics (i.e., number of reactors, installed capacity, timelines to retirement) as the existing U.S. nuclear fleet in the scenario. The

second concept is a parametric study of the two parameters that most directly affect the reuse of UNF assemblies: the deployment rate of new reactors and the minimum time before an assembly can be reused. In modeling the deployment of a fleet of breed and burn reactors and their associated fuel cycle, the following assumptions must be made:

- All breed and burn SFRs deployed have the same reactor design and operational characteristics as described in Section 3, except for the fuel composition which will change between units.
- The minimum time before a UNF assembly is reused (hereafter referred to as cooling time) is independent of burnup, and constant across the fleet for a given scenario.
- There is no maximum time that a UNF assembly can cool before it is reused.
- The fuel composition of UNF assemblies reused is the average from the stock of those available.
- Fresh LEU assemblies are fabricated with material at the needed enrichment.
- Post-S1 reactors cannot be fueled by material discharged from an S1 reactor within the preceding year, effectively setting a minimum cooling time of 1 year.
- S1 reactor core assemblies are selected and transported over the course of the 4 years prior to a subsequent (i.e., S2) reactor coming online.
- Natural uranium fuel is used for refueling.

These case studies were modeled using the DYMOND nuclear fuel cycle simulation (NFCS) code [Richards, 2021]. DYMOND is an Argonne National Laboratory-developed and maintained NFCS code. The DYMOND code models a State-level nuclear fuel cycle including material inventories and transport, reactor irradiation with depletion physics, and front-end and back-end processes. Further details on the DYMOND modeling method are provided in Appendix C.

5.1. Real-World Deployment Scenario

The real-world deployment scenario of a fleet of breed and burn SFRs (Figure 10) was calculated from replacing the retiring nuclear capacity (of legacy LWRs) with breed and burn SFRs, in a State with equivalent nuclear generation as the United States. The pace of deployment in this scenario is based on the current license expiration dates of U.S. commercial nuclear plants, shown in Figure 11 [NRC, 2022]. This case of deployment was chosen to best investigate the effects of a non-fixed deployment rate informed by real-world data. The scenario also demonstrates a plausible deployment profile that does not rely on any assumptions regarding the growth of nuclear power or other future economic or political factors. Given the large nuclear generating capacity in the U.S., this scenario is most representative of a deployment experience in a nuclear weapon state. The timeline begins in 2020 assuming the technology necessary for commercial deployment of the SFRs is ready for deployment commencement in 2035. It is noted that the initial deployment date to complete breed and burn reliance appears aggressive in this scenario, the purpose of which is to demonstrate the characteristics of a very vigorous and successful deployment.

Results from the model can be seen as follows. Figure 10 and Figure 11 are graphical representations of the transition to a breed and burn fuel cycle over the course of 21 years, during which 61 reactors (73.2 GWe capacity) are deployed, with the last being deployed in 2056. The S2 and subsequent generation reactor cores are fueled prioritizing available UNF and any shortfalls in driver assemblies being made up

with new LEU assemblies. The combinations of driver assemblies and natural uranium assemblies in the first cores were calculated based on first core criticality requirements and approximations for k_{inf} using ORIGEN2.

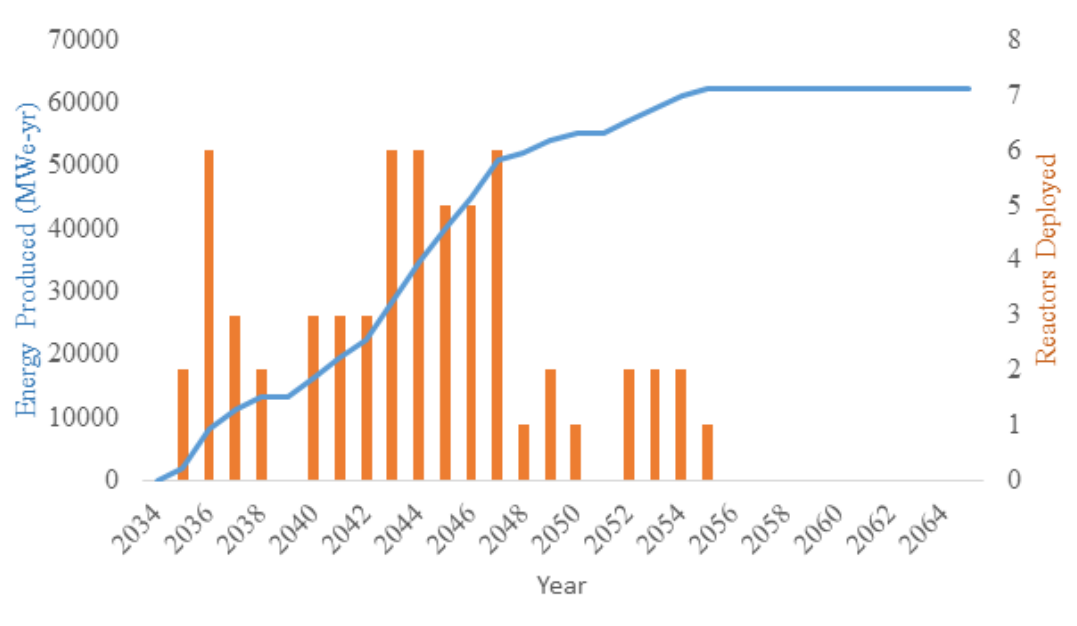


Figure 10. Deployment schedule and annual energy generation from breed and burn SFRs.

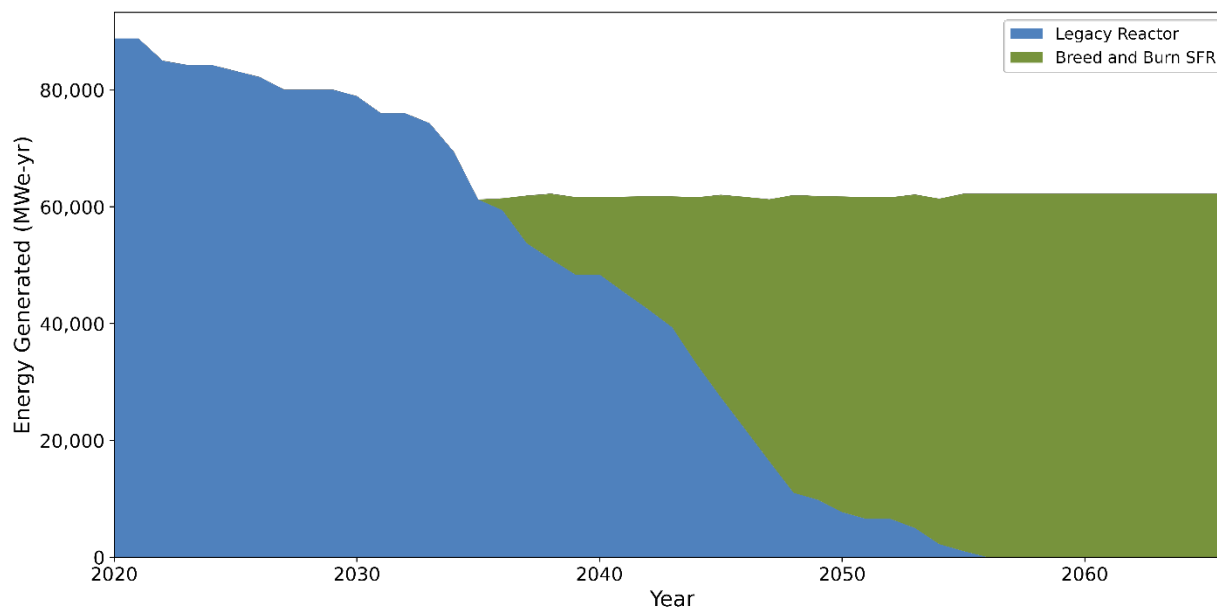


Figure 11. Energy generation during "real-world" scenario from a country of similar classification to the U.S.

S2 reactors serve as a bridge between the S1 reactors that are fueled solely with fresh fuel and the S3+ reactors that are fueled solely with UNF driver fuel and natural or depleted uranium used for refueling. As described in the previous models, S2 reactor fuel generally includes a more complicated burn history compared to other reactor generations which will contribute to the uncertainties. This in turn brings

unique concerns for nuclear material accounting at this generation. The time necessary for a fleet to reach self-sufficiency¹³ and the transition duration¹⁴ collectively influence the number of S2 reactors deployed.

Table 10. Characteristics of transition from S1 to S3+ SFR first cores in the real-world scenario measured in units of metric tonnes (MT).

Minimum Cooling Time (years)	Transition Duration (years)	Year of Self-Sufficiency (S3)	S2 Generation Size (# reactors)	Peak UNF Transfer (MT Heavy Metal/Assemblies/MT Pu)
3	6	2046	38	42.25 / 102 / 3.26
5	5	2047	34	32.09 / 78 / 2.54
7	3	2047	25	29.57 / 72 / 2.27

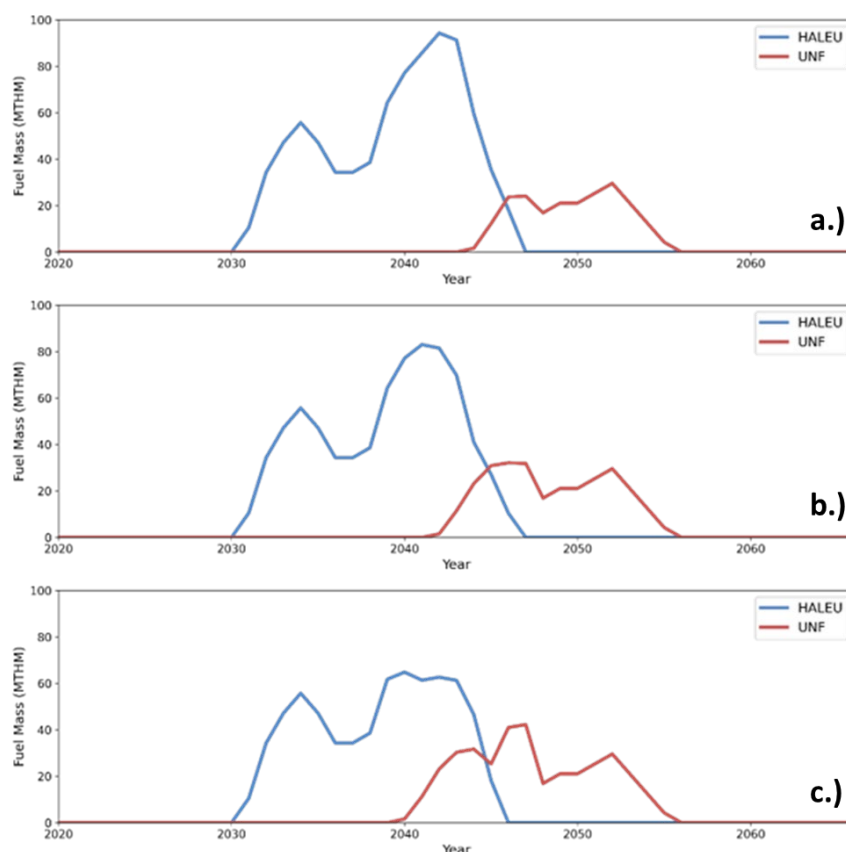


Figure 12. Mass of nuclear material of interest being charged each year by material classification for a 7-year (top, a.), 5-year (middle, b.), and 3-year assumed minimum cooling time (bottom, c.).

The deployment model also considered separate cooling times and the impact to the overall scenario, with UNF assemblies at three different (assumed) minimum cooling times before reuse: 3 years, 5 years, and 7 years. The 7-year minimum cooling time value is consistent with past studies on the breed and burn concept and was therefore considered as the base case scenario. The results are presented in Table 10

¹³ The point where the fleet is large enough to generate sufficient UNF to fuel subsequent reactor deployments.

¹⁴ The time between initial deployment of the first S2 reactor and deployment of the last S2 reactor.

and Figure 12. A full breed and burn fleet deployed on the described timeline is capable of reaching self-sufficiency between 2046 and 2047, depending on the assumed minimum cooling time, as illustrated in Figure 12. In contrast, a deployment scenario with a 3-year minimum cooling time achieves self-sufficiency only nominally faster than the scenario that assumes a 7-year minimum cooling time. The transition time to UNF-only fueled cores occurs more rapidly for the 7-year cooling time scenario (taking only three years) when compared to five years for the 5-year cooling time case and six years for the 3-year cooling time case. Due to the assumptions that fuel cannot be transferred until the following reactor year and that transfers occur over a four-year period, fuel transfers are affected three years beyond the transition period. This is reflected in the S2 generation size reported in Table 10.

The amount of UNF that will be available for use in subsequently deployed reactors is relative to the cumulative number of reactors that have been deployed prior. A front-loaded deployment schedule, as assumed in this scenario, allows for a larger amount of UNF to become available towards the beginning of the S2 generation than if the deployment rate was constant. A spike of deployments in 2044-2049, followed by a slower deployment rate, limit when the S3 generation can begin. S3 requires a large amount of material over years when the fleet is still relatively small and then decreasing that demand after the fleet has grown. This results in the first year of the S3 generation deployment only varying by one year despite requiring four years of less cooling time.

The deployment schedule also impacts the maximum amount of material that is transported between reactors per year as it is limited by material availability rather than demand in this scenario. This is again reflected in the greater than 40% higher peak UNF and Pu transfer rates for the 3-year minimum cooling time case compared to the 7-year minimum cooling time. The shorter minimum cooling times result in a longer S2 generation time and a larger peak mass of UNF that must be transported between reactors. These effects are the result of several features of the deployment profile, and the intrinsic relationship of these characteristics to the cooling time and deployment rate are detailed in Section 4.2.

Another factor impacted by the minimum cooling time is the number of different reactors an assembly can be expected to involve during its lifetime. The longer minimum cooling time in the base case scenario limits the maximum number of different reactors in an assembly's lifetime to 3, given there is a minimum of 8.5 years between loadings. This is in comparison to the 4.5 years between loadings for the shortest minimum cooling time scenario, which corresponds to a maximum of 5 reactors. While the number of reactors in which an assembly is used does not directly impact the quantity or quality of nuclear material that must be accounted for, it does increase the uncertainty in the composition of the material in the assembly. The importance of this uncertainty is emphasized due to the lack of quantitative assay options for UNF. A quantitative evaluation of the uncertainty is beyond the scope of this analysis.

The total quantity of material available in the fuel cycle is proportional to both time and the total number of reactors deployed. Figure 13 illustrates the exponential inventory growth while the fleet is growing, and the linear growth once the equilibrium fleet size is reached. However, shorter minimum cooling times can reduce the overall mass of SNM in the system as the quantity of LEU input is reduced as well as the S1 generation time. This effect is shown in the total discharged U-235 mass, presented in Figure 14.

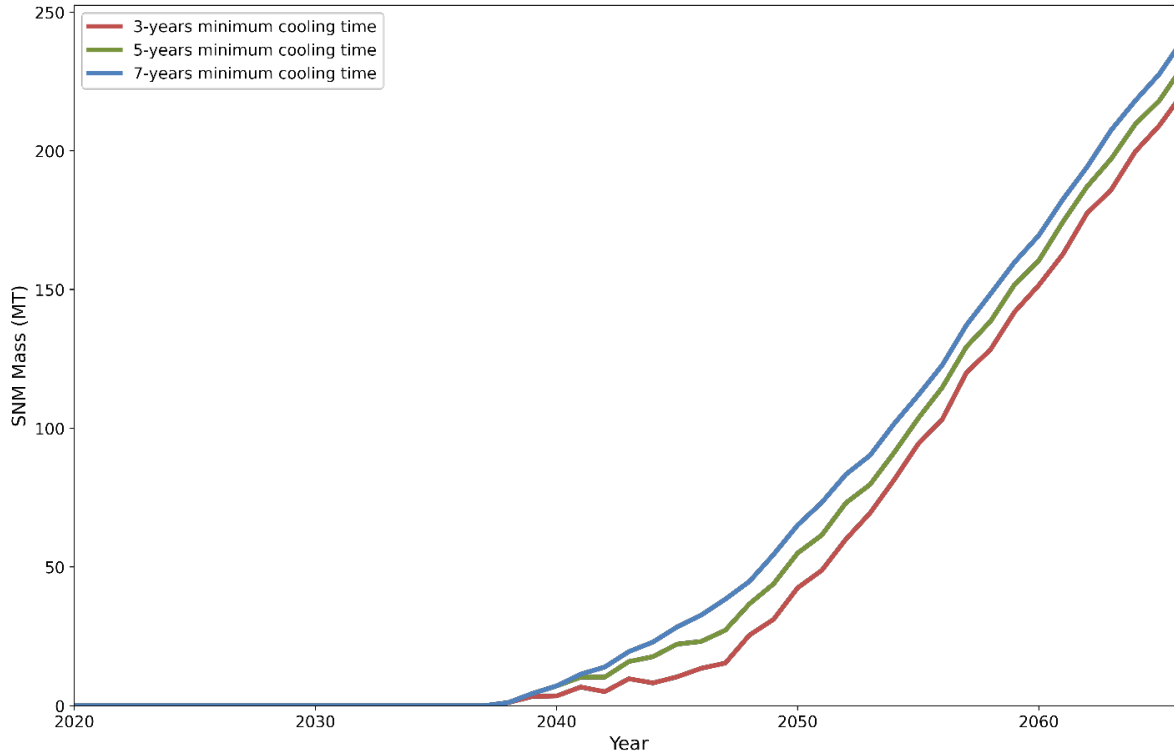


Figure 13. Mass of special nuclear material in the backend of the fuel cycle.

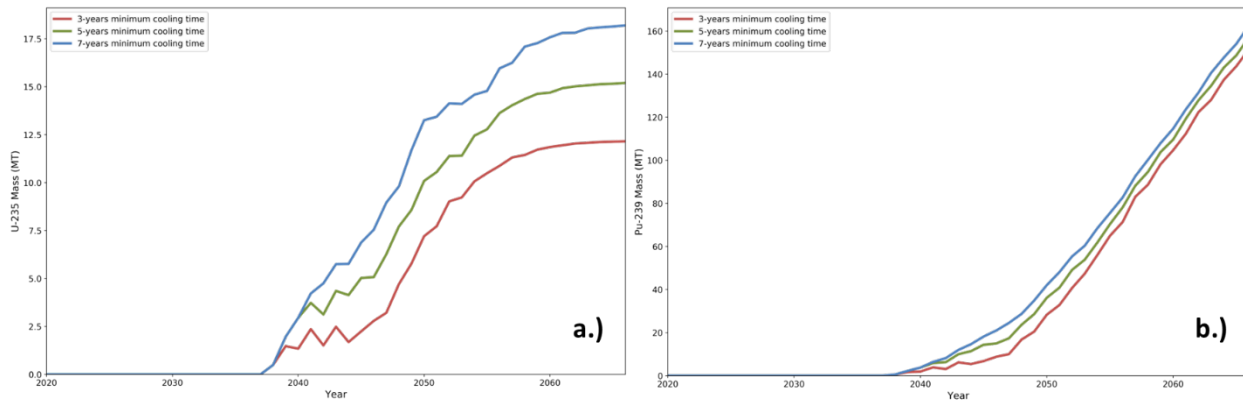


Figure 14. U-235 (left, a.) and Plutonium quantity (right, b.) in the backend of the fuel cycle,

While the scenario does not negate any requirements for IAEA safeguards implementation, it is noted that a more rapid transition to the S2 generation also accelerates the decrease in the material attractiveness. The special nuclear material present in the initially discharged fuel will be mostly U-235, which began as LEU and only achieved a very low burnup. However, as demonstrated in representative reactor comparison, the quality of the Pu is very high (>92% Pu-239) and the average quality of the Pu in the backend of fuel cycle decreases more slowly than the increase in fraction of material that is Pu, seen in Figure 15. This trend continues until UNF can be irradiated again, as the low-burnup high-quality assemblies leave the backend of the fuel cycle and the newly discharged assemblies have a much higher burnup. Both the fraction of used material that is Pu and the decrease in the overall Pu quality are reached as the equilibrium cycle approaches. This trend is observed regardless of minimum cooling time and is represented in Figure 15 for the 7-year cooling time as the model scenario.

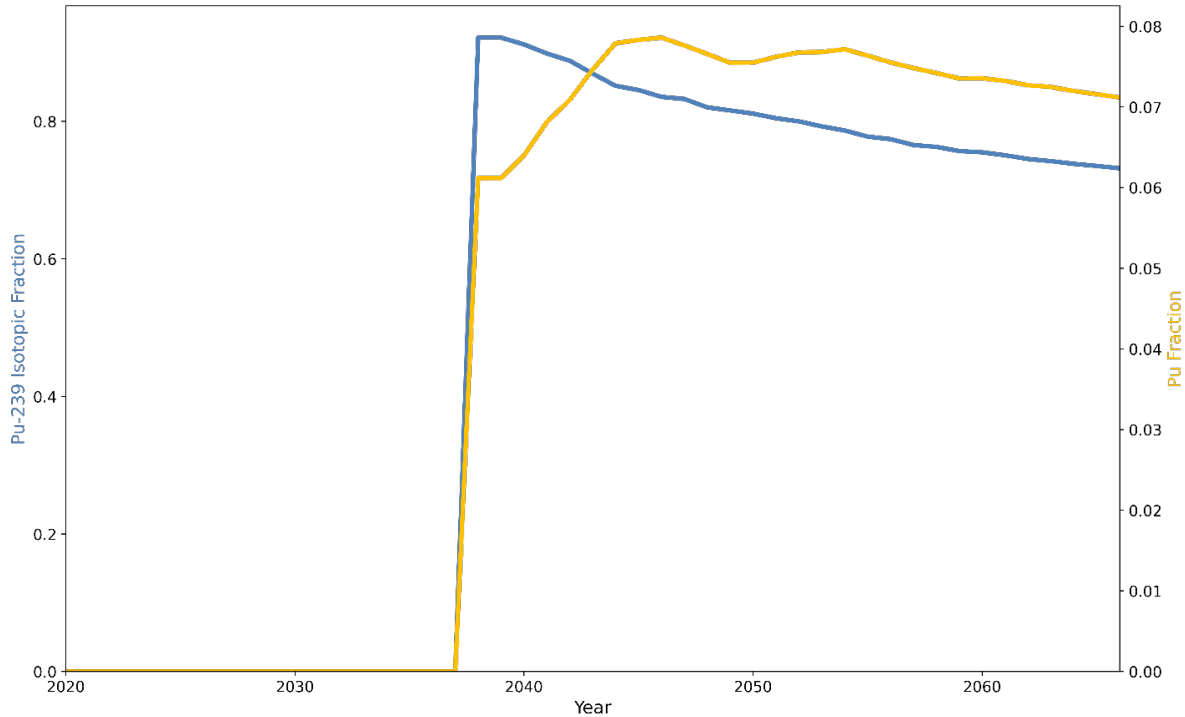


Figure 15. Plutonium fraction and quality in the backend of the fuel cycle.

5.2 Breed and burn Deployment Parameter Study

The second study is the parametric study of how the deployment rate and minimum cooling time affects the material composition in the backend. The second study made use of hypothetical deployment scenarios, rather than real world scenarios. The direct and synergistic effects of the two most impactful parameters for determining the quantity and composition of material were examined. Namely, these are fuel storage outside the reactors and transport between reactors, as well as associated cooling times and deployment rates for both. The results show how fleet deployment parameters may influence the quantity of nuclear material in the overall system which subsequently affects the rate of nuclear material required to be moved between reactors, and the overall range of results that are possible in deploying a fleet of breed and burn reactors. These results may find more applicability than the results presented in Section 5.1, as this analysis may be used to simulate the decision-making process a State may adopt when pursuing a breed and burn nuclear fuel cycle. Likewise, the results may be used to forecast the overall difficulty of implementing international safeguards in the State.

For a given scenario in this study, the deployment is designed to be linear in the number of reactors deployed each year for 25 years. The deployment rates varied from 0 to 5 reactors per year and the minimum cooling time varied between 2 months and 10 years. In the case of a deployment rate of 0, a single reactor was deployed in the first year with no subsequent deployments. Other than these variables, all other assumptions, reactor characteristics, and fuel cycle characteristics were the same in this study as described in the real-world scenario presented in Section 5.1. These two parameters were varied simultaneously using a Latin Hypercube sampling method with 2000 total scenarios sampled to uniformly sample the entire parameter space (see Appendix C.3).

The results from this analysis show nearly identical trends for all relative measures of the material in the backend of the fuel cycle (i.e., Pu fraction, Pu quality, U quantity) and closely resemble the results from

the real-world deployment study. This is to be expected as there is no strong correlation between changes in the discharged isotopics due to the cumulative effect of UNF discharged and the smooth transition of fuel types from LEU to UNF during the S2 generation. The composition of the average discharged UNF per fleet will likely weight more heavily towards earlier generation (S1) of deployed reactors due to the cumulative effect. However, outliers were seen in the scenarios where very few total reactors were deployed. This is because a single discharged batch of UNF represents a large fraction of the total material in these scenarios and thus the average burnup of assemblies varies rapidly with assembly charge and discharge.

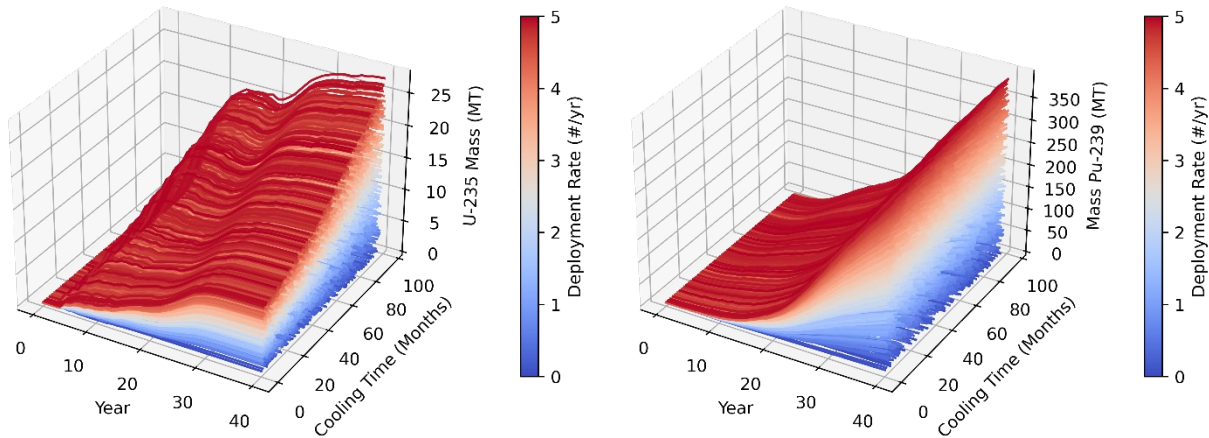


Figure 16. Time dependent mass of U-235 and Pu-239 in the backend of the fuel cycle for all scenarios

Overall, the absolute measures such as total SNM mass, show a strong correlation across the varied parameters. In Figure 16, the strong positive correlation can be seen between the mass of U-235 and both the minimum cooling time and the deployment rate, as this value is driven by the total quantity of HALEU necessary for fleet deployment. However, the mass of Pu-239 is not correlated to the cooling time and simply increases proportional to the fleet size. These correlations can be seen in Figure 16 as the mass of U-235 initially rises during S1 (local maximum between years 10 and 20) then plateaus or, in the case of fast deployment rates, even decreases during S2. This trend continues until the cumulative effect of higher burnup UNF being discharged is greater than the effect of UNF being loaded in the system. An increase in deployment rate, thus the total number of reactors deployed, clearly results in a corresponding increase in the SNM in the system and likewise the mass transported; however, when normalized to the annual power generation, there is still a strong correlation between the quantity of SNM in the backend of the fuel cycle and the minimum cooling time as shown in Figure 17.

In addition to the total quantity of material trend, the effect of multiple reuses of an assembly is also observable in the masses of U-235 throughout the backend of the fuel cycle, particularly in the scenarios with minimal cooling time. While the scenarios with long cooling times have very few assemblies being reused more than once or twice, those representing a rapid deployment with low minimum cooling times have assemblies that are reused in up to 8 different reactors. Furthermore, a rapid increase in the quantity of U-235 in the backend of the fuel cycle occurs in all scenarios as new reactors cease to be deployed. This is the result of HALEU assembly build up in the backend of the fuel cycle that do not see use for further fleet growth.

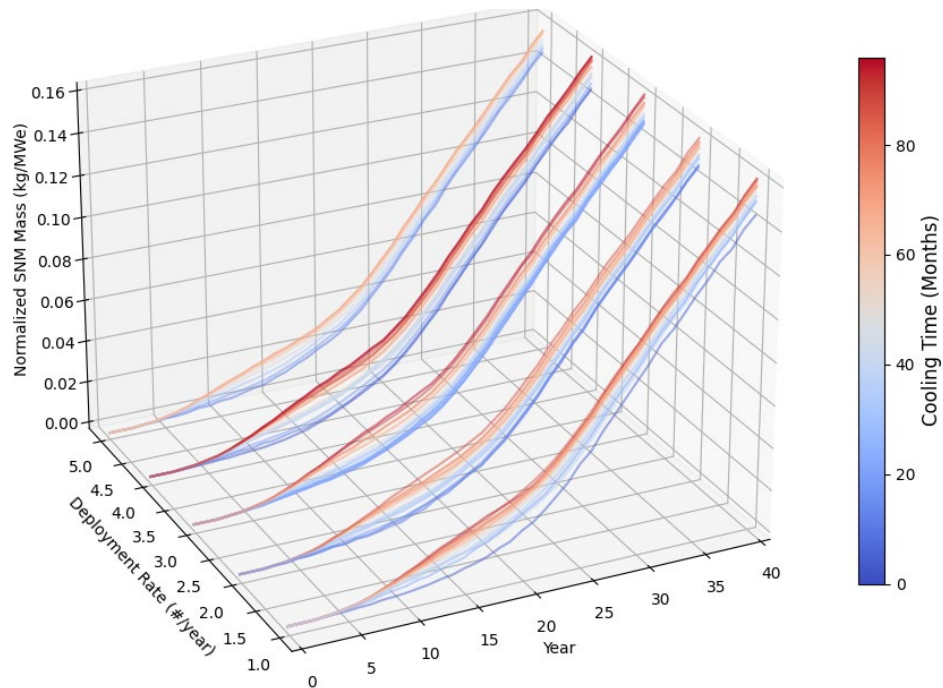


Figure 17. Deployment rate cross sections of the normalized SNM mass in the backend of the fuel cycle

The results of this parametric study can be further expanded upon through a global sensitivity analysis with Sobol indices. A detailed explanation of the indices is given in Appendix C.1, this sensitivity analysis can be summarized as a means for determining how much parameter variance affect a metric. The time dependent behavior of the deployment was chosen for demonstrative purpose and are represented by two key response metrics:

- The total mass of excess SNM generated during the scenario normalized to the power generated and
- The annual mass of SNM transported to support self-sufficient reactor deployment (i.e., deployment during the S3 generation).

The first metric provides a measure of the material generated in each scenario and (as can be seen in Figure 17) the time at which the measure ceases to be significant for relative comparisons. The second metric provides a measure of the peak material transfer that will require safeguarding during a scenario. The results of these metrics provide insights that demonstrate the challenges associated with a breed and burn reactor deployment. Until the time where safeguards techniques are applied to a breed and burn fleet, the appropriate methods must be considered and conceptualized. Results from metric studies such as those demonstrated here can provide the quantitative information necessary to inform safeguards decision making prior to deployment.

The indices, and the observable trends in the sampling data (available in Appendix C.2), indicate that both response metrics are dominated by a single parameter. The normalized SNM mass that is largely dependent on the cooling time, and SNM shipment at equilibrium is largely dependent on the deployment rate. The metrics are positively correlated with the parameters in both cases, thus if the parameter

increases, the metric will too. These relationships resemble the observations made on the real-world scenario model (the causes were discussed previously). However, there are also synergistic effects of the minimum cooling times on the SNM mass shipment at equilibrium. The synergistic effect is such that the longer cooling times reduce equilibrium shipment mass while increasing deployment rate. This effect is a result of increasing overall UNF worth (in scenarios with longer cooling times) becoming more impactful the more reactors that are deployed. The increased reactive worth of UNF arises from UNF originating as LEU and more UNF available for reuse due to the much larger S1 generation and are therefore more alike in isotopic composition to the first few cycles. This increased worth results in fewer assemblies necessary for initial startup of new reactors in the fleet, and thus less material is transferred.

6. Safeguarding the Breed and Burn Reactor Fleet

Key Takeaways for International Safeguards

- The movement of irradiated fuel between facilities, the quantity of Pu increasing exponentially with increased deployments and irradiation cycles, and the longer lifetime of the fuel all present unique new safeguards challenges inherent to breed and burn reactors.
- Computational error in depletion calculations, uncertainties in calculations, and undocumented variations in the operational history of a given assembly may result in high overall uncertainties in nuclear material quantities reporting. This could complicate efforts by an operator or State-level safeguards authority to meet timeliness goals, or complicate efforts to re-verify nuclear material quantities if CoK is lost.
- Deployment of a fleet of breed and burn reactors will require routine shipment of irradiated nuclear fuel between facilities. While robust C/S measures will be implemented during shipment, the large quantities of nuclear material involved, and complicated burn history may necessitate additional quantitative measures to accurately determine fissile content of assemblies.

6.1. Reactor Facility Safeguards

Safeguarding the breed and burn fleet is driven by the design of SFR facilities and the need to meet timeliness goal for irradiated direct use material. Specifically, safeguards for SFRs have been previously defined by a reliance on C/S and maintaining CoK approach [Beddingfield, 2007]. Likewise, the relevant safeguards to a breed and burn fleet will involve many of the approaches already considered for SFRs and other used fuel facilities. The storage of direct and indirect use material will also determine the actions that must be taken to meet the appropriate timeliness goals for the facilities. Therefore, discussion on safeguarding the breed and burn fleet may be considered in two parts: safeguarding the fuel at the reactor and safeguarding the movement of fuel between reactors. The movement of fuel between facilities, the quantity of Pu increasing exponentially with increased deployments and irradiation cycles, and the longer lifetime of the fuel, all present unique new safeguards challenges.

Some considerations have been made for safeguarding non-light-water-cooled reactors such as SFRs, and conclusions on best approaches were drawn from the experience of safeguarding the Monju and Joyo reactors. At the time of this study, fast reactor safeguards have only been applied to a limited number of facilities in CSA States: Joyo and Monju in Japan and Aktau in Kazakhstan [Williams, 2009]. In particular, the measures implemented at the Monju reactor and the lessons learned all help shape the approaches for safeguarding future fast reactors, such as those in a breed and burn fleet.

SFRs pose a variety of challenges for reactor safeguards approaches because continuous monitoring of the fuel is difficult given a variety of reasons; visuals on the fuel is difficult when submerged in opaque liquid sodium, the chemical reactivity of sodium is too high for many viewing devices and detectors, and remote handling is necessary for irradiated UNF [Beddingfield, 2007]. Thus, fuel assemblies are treated as individual items with unique serial identifiers when received by the facility, whereby a visual inspection can be conducted for verification prior to loading in reactor. Visual inspection of the fuel assemblies may be limited but possible when the fuel is removed from the transfer cask prior to entering the reactor and when the irradiated fuel is removed from the core (prior to encapsulation in

the next transfer cask). Under these circumstances, NMAC may be best achieved through rigorous book-keeping and reviewing to maintain CoK, core verification, and UNF reverification. C/S will likewise involve continuous monitoring of fuel moving equipment with cameras (visual and ultrasonic) and detectors, and seals on transfer casks. A comprehensive list of C/S equipment may be found in the Safeguards Techniques and Equipment publication [IAEA, 2011].

Once the fuel enters the reactor system, the assembly will not be accessible for visual inspection until the fuel is removed. Monitoring the core for movement of the assemblies may be possible with under-sodium viewing devices. In the example breed and burn fleet, UNF is transferred to dry storage in a large hot cell before shipment to the next reactor, traditional techniques such as Cerenkov viewing in a spent fuel pond is therefore not applicable. Therefore, emphasis should be placed on other adequate material accounting techniques for safeguarding fissile content within the breed and burn fleet reactors for meeting timeliness requirements.

In support of material accountancy, depletion model calculations are necessary in conjunction with stringent monitoring techniques. Modeling calculations in lieu of direct nuclear material measurement of UNF is a standard practice in traditional LWR reactors. Fuel assemblies in a breed and burn fleet are a particular challenge for accountancy given the extremely high burnups achieved after multiple irradiations. Four separate stores of fuel exist at the reactor site, all of which are initially empty before any fuel has been received, each with specific international safeguards implications depending on the material characteristics and nuclear material content:

- The unirradiated fuel storage area, which can only receive and handle unirradiated fuel. Applying safeguards to this area consists of considering the material category as indirect use material (depleted to low enriched uranium) and the corresponding timeliness detection goal of twelve months.
- The UNF storage area, which is designed for UNF, but can also receive unirradiated fuel. An MBA capable of holding both irradiated direct use material (UNF) and indirect use material (unirradiated fuel) will require safeguard measures that assist in meeting the 3-month timeliness detection goal of the irradiated direct use material.
- The reactor core region where the power is generated. Though material herein is ultimately irradiated direct use material, the operations of the reactor inhibit access between inspection campaigns hence requiring robust NMAC measures to be applied to the reactor vessel itself (as is done for conventional light water reactors at existing nuclear power plants).
- The IVS for storage of UNF allows for cooling in the primary sodium coolant prior to transfer to the UNF storage area. Like the reactor core itself, the irradiated direct use material herein must rely heavily on a robust NMC approach to the containment vessel itself.

These considerations must be made for inventory storage and flow given the variety of SNM types. Where that irradiated direct use material and indirect use material may be stored within the same storage area, the level of safeguards applied to the storage area must pertain to the material of more concern: the irradiated direct use material. Thus, the timeliness goal will be 3 months (compared to the 12-month goal of just indirect use material) and subsequently dictate the inspection goals, physical inventory frequencies, and C/S measures for all SNM in the storage facility. Unirradiated fuel does not require a UNF shipping cask and will be received in a dock facility for inspection and transfer to storage. UNF received or shipped will be via shipping cask which will be loaded and unloaded in a specialized cask handling facility

that can withstand the large weight of the shipping cask and the high radiation levels of the UNF. The maximum number of assemblies per shipping cask was not evaluated but is likely to be a small number limited by criticality given the high fissile content of the fast reactor UNF.

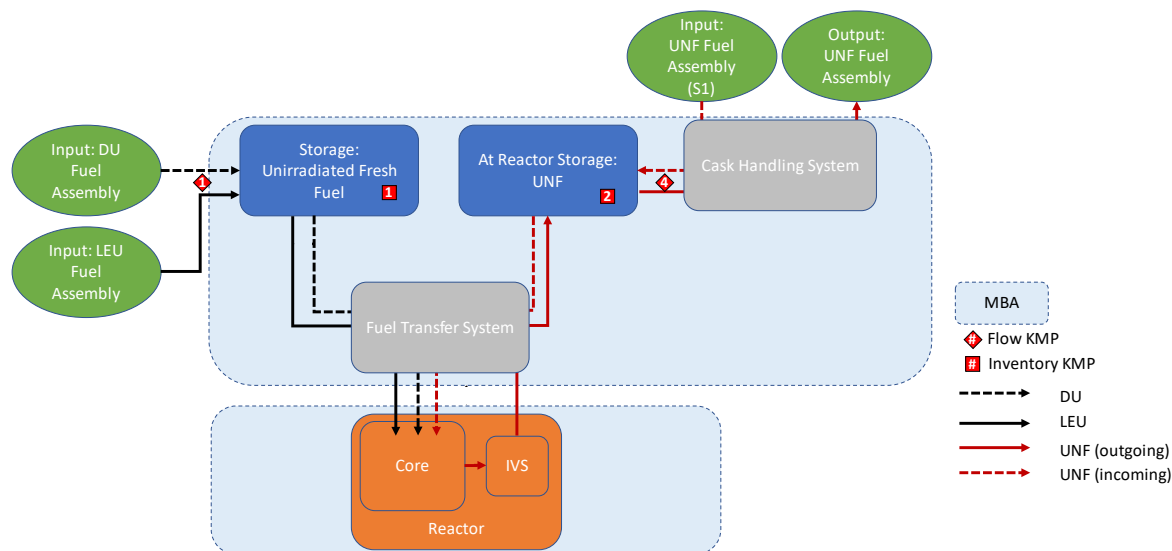


Figure 18. Example MBA and KMP for S1 or S2 generation fast reactor in the breed and burn fleet.

In the example S1 and S2 reactors, the entire reactor facility may be considered two MBAs, one for the reactor and in-vessel storage and one for fuel storage. An example of the relevant MBAs for the S1 and S2 reactors can be seen in Figure 18. Given the operational procedure of the breed and burn fleet, UNF and LEU may occupy the same storage area at the initial startup of S2 reactors only because of the large number of assemblies received prior to initial loading of the reactor, whereby both fuels will be loaded into the core and the storage emptied. Subsequent refueling will not involve UNF and LEU in the same storage area. The fuel storage MBA will contain both direct use and indirect use material. Of the two material types, safeguarding the irradiated direct use material will take precedence and thus inform the timeliness goals of the facility. Inspections in the area will require a mechanism to differentiate between the two fuel types. When material leaves the storage MBA, separate KMPs will be established for UNF and (fresh) LEU fuel from storage areas.

NMAC techniques can be applied during the movement of the fuel by an ex-vessel transfer machine. The transfer machine also serve as a KMP for the fuel. This concept of monitoring fuel movement was demonstrated in Monju, and an illustration of the ex-vessel transfer machine can be seen in Figure 19. The transfer machine moves only in a bilateral direction from the fuel storage to the reactor vessel (and back) and is equipped with monitors for C/S and radiation detection. Perimeter radiation monitors are present while the transfer machine moves between reactor containment and storage area.

Concerns for fuel diversion include the removal or replacement of whole fuel assemblies, unlike traditional LWR fuel where diversion of fuel rods is most concerning. Thus, NMAC goals will be to verify whole fuel assembly movement at the reactor facilities. Verification of fuel flow (through attribute sensors at the KMPs), integrity of seals, material presence, content, and quantities through NDA, and any applicable sampling techniques will be conducted at this stage prior to core loading when the fuel is moved between MBAs. A similar procedure will apply when the fuel leaves the reactor, using the same

ex-vessel transfer machines to move fuel from the reactor vessel MBA to the fuel storage MBA where the used fuel is prepared for shipment to the next generation reactor or storage location.

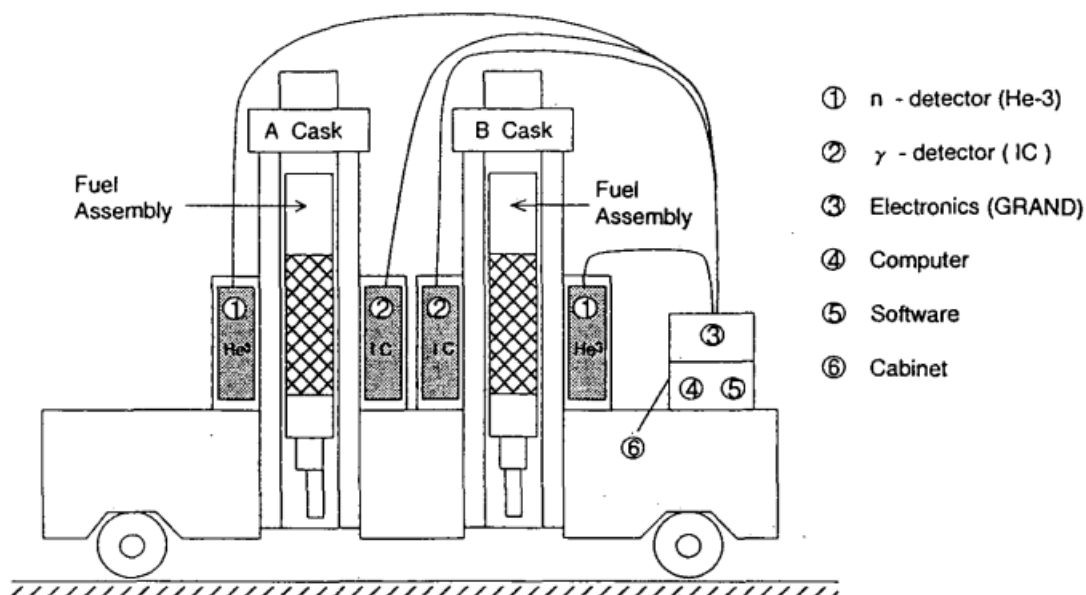


Figure 19. Ex-vessel transfer machine used for loading and unloading of nuclear fuel at the Monju reactor

Safeguarding S2 and subsequent generations of reactors in the fleet will largely involve the techniques discussed but implemented in varying frequencies for each reactor's unique operational history to meet timeliness goals. While inventory data management is an established practice in current safeguards of power reactors, the breed and burn reactors are highly reliant on accurate fuel data and maintaining CoK. Given the enhanced importance of CoK, an operational upset that results in a loss of CoK could severely impact the ability of IAEA to draw a conclusion for a B&B facility. Therefore, developing a method for recovering CoK may be prudent to ensure each facility is within compliance throughout its entire operational lifetime. This may require the industry to pursue additional methods in conjunction with this safeguards approach to ensure reliability and develop methods to recover CoK by reverification if necessary.

6.2. Fuel Transfer Safeguards

In nuclear safeguards applications, NDA and DA techniques can be employed to ensure the quantity of SNM is as declared by the facility. However, as NDA measurements for irradiated fuel generally rely on active and passive gamma-ray and neutron measurements, the isotopic composition of UNF may be difficult to determine with traditional NDA methods. A complete description of the available NDA methods for used fuel may be found in IAEA report STUK-YTO-TR 133 [Tarvainen, 1997]. The intensity of the emitted radiation from UNF assemblies can render spectrometry ineffective and the measurements taken may be outside the range of acceptable error. DA techniques require direct samples taken from UNF for further evaluation in a laboratory. This is extremely difficult given the SFR fuel form, the time and cost of laboratory analysis, and potential need for hot cell facilities. It would be a large burden on the facility operator and may be impractical for use in quantifying material inventory of the breed and burn fleet.

Additional research into alternative methods for NDA of UNF could be a great benefit to breed and burn safeguards efforts and allow for more accurate accounting in-situ as well as provide additional support to the CoK methods previously discussed. Spent fuel inventory database management is vital to the safety and operational needs of a breed and burn fleet and may utilize the same reporting practices as current power reactors. The IAEA may also conduct independent verification of the reporting based on model calculations and reporting history from the facility. This may require the development of new skills within the IAEA inspector and analyst workforce. It should be noted that verification of the fuel inventory must be an inherent feature in the breed and burn fleet regime, as accurate reporting is integral to fleet operations. The IAEA may rely on the accuracy and reporting history of the breed and burn reactors knowing this fact and ease inspector burdens.

Historically, the IAEA has relied heavily on material accountancy measures and maintaining CoK for verification and safeguards in nontraditional reactors. In comparison, safeguarding traditional LWR fuel may be less complex given several factors: the LWR facilities manage UNF on-site indefinitely with little to no fuel movement, the quantity of Pu in LWR is significantly less, UNF is visible while in storage pools. The same emphasis on material accountancy for traditional reactors will also apply to the breed and burn fleet given that deployment operations must rely heavily on fuel CoK between reactors. Following irradiation in the S1 reactor, the assemblies are relocated to a different section of the core (i.e., in-vessel storage) to cool prior to packaging in transfer casks and shipment to the S2 reactor. Typically, the Pu content of SNF is reported at the time of final discharge from the reactor [IAEA, 1980]. In the breed and burn fleet, this would still occur following final discharge from the core.

Operators of all reactor types rely on depletion modeling to support NMA reporting. In instances where the reactor fuel is a uniform composition, with a simple burn history, and a low average burnup, this is a straightforward exercise and results in low uncertainty when reporting nuclear material quantities. However, for the S2 generation of reactors (and onward), the fuel in the core may be a mix of fresh fuel and fuel with a complicated burn history involving multiple cycles in multiple reactors irradiated to high burnups. Computational error in depletion calculations, uncertainties in calculations, and undocumented variations in the operational history of a given assembly may result in high overall uncertainties in reporting. This could complicate efforts by an operator or State-level safeguards authority to meet timeliness goals, or complicate efforts to re-verify nuclear material quantities if CoK is lost. However, maintaining accurate reporting of the assemblies will likely be a fundamental component to the functionality of the breed and burn fleet, as the receiver of the UNF must know the operational history associated with each assembly to successfully integrate the UNF into the reactor. The emphasis on proper reporting and maintaining CoK is therefore an inherent feature of breed and burn fleet deployment and efforts should again be made to strengthen existing methods and ensure the suitability of operator data for use in safeguards reporting.

Deployment of a fleet of breed and burn reactors will require routine shipment of irradiated nuclear fuel between facilities. While robust C/S measures will be implemented during shipment, the large quantities of nuclear material involved, and complicated burn history may necessitate additional quantitative measures to accurately determine fissile content of assemblies. As discussed, the reliance on CoK is necessary to address the unique challenge of SNF as well as preserving operational history. In the existing nuclear fuel cycle, reprocessing facilities and long-term used fuel repositories are the only facilities that routinely accept offsite shipments of irradiated nuclear fuel. In the case of reprocessing facilities, the nuclear fuel is typically dissolved in an accountancy tank after initial receipt, allowing for an

accurate quantitative measurement of contained nuclear material. This is not possible at a geological repository or reactor facility. It is typical for a geological repository to perform gross defect verification or partial defect verification on fuel assemblies prior to encapsulation or placement [Finch, 2016]. Such techniques would also be applicable to receipt operations at a reactor facility receiving UNF as a supplement to containment and surveillance measures. However, this presents two limitations. First, since breed and burn UNF assemblies will likely have as much as 5 times the SNM content as LWR assemblies, reliance on gross and partial defect verification alone may not be sufficient to meet the quantity component of IAEA inspection goals. This is mitigated by the difficulty associated with removing individual pins from an SFR assembly. Second, if CoK is lost during shipping operations, it is difficult to independently confirm the provenance of breed and burn UNF, due to the lack of established verification techniques, and recovery of lost information would be even more troublesome for the operators of each generation of the fleet.

A more advanced non-destructive verification technique, neutron fingerprinting, has been suggested for similar applications but has not yet been deployed [Evans, 2011]. This would allow a facility operator to have more confidence in UNF declarations from the shipping facility and mitigate the process of reverification after loss of CoK. Initial work on this technique demonstrated its application to fast reactor fuel in dry storage [Santi, 2010]. While it has not been demonstrated for this specific use case, this is a promising technique to precisely verify received fuel and could be considered in an approach for breed and burn reactors where more stringent verification mechanisms may be warranted.

The techniques mentioned above for receipt verification can identify defects or confirm a given assembly that is received matches the assembly discharged, but they are not capable of performing nuclear material quantification. It is unclear whether calculation methods alone will be sufficient for an operator to meet their reporting requirements or whether calculations would need to be supplemented with nuclear material measurements to verify facility inventory. Future work derived from this study may consider options for nuclear material quantification and identify situations where it may be required.

6.3. Safeguarding the Full Fleet Deployment

Considerations for safeguarding the breed and burn fleet were also made for the real-world deployment scenario described in Section 5. Specifically, the efforts made in this study highlight the complexity of a fully realized reactor fleet and to call attention to the inherent safeguards challenges. The following timeline is a summary of the real-world deployment scenario described in Section 5.

- Year 0: SFR construction begins
- Year 15: Commercial SFRs are complete and ready for deployment
- Year 27: Breed and burn fleet reaches self-sustainment
- Year 35: Full breed and burn fleet is deployed with 61 reactors

While a comprehensive safeguards regime cannot be detailed in this study for the real-world model, considerations can still be made for the applicability of safeguards techniques. The verification techniques discussed previously are applicable to the real-world deployment scenario, with further emphasis on accurate reporting histories and maintaining CoK during shipment. Aside from the increased reactor count, the real-world scenario also considers another feature that was not discussed in the S1 and S2 model: reactors (and subsequently UNF) from multiple locations. One complexity is

maintaining a relatively low uncertainty. Reactor operators and the IAEA have relied on low uncertainties and calculation errors in the reporting to detect discrepancies or possible signs of diversions. Verification and NMA in a system such as the real-world scenario must account for the increase in uncertainties that will accumulate as UNF moves from reactor to reactor and State to State.

Given the uniqueness of the fast reactor system, this study concludes that a real-world deployment scenario must still rely heavily on a rigorous CoK system maintained by each reactor operator. This can be supplemented by nontraditional techniques such as fingerprinting or other identification methods for cask tracking and transfers in addition to shipper/receiver differences. The IAEA may still consider each facility individually instead of as an overall fleet, given the task of maintaining both fuel inventory data and a continuous record of fuel movement across multiple reactors for decades will become a heavy burden on inspectors. An accurate reporting system must be in place to facilitate the use of UNF from multiple sites. This reporting system should be standardized across a State to allow the IAEA to rely on the accuracy of operator reporting and thus ease the burden to inspectors.

A fleet deployment with many reactors has the potential to rapidly increase the total global amount of Pu under IAEA safeguards. Additionally, the potential for such a fleet to produce very high quantities of nuclear material in a single State may require the IAEA to implement more stringent inspection frequencies as part of a State-level approach.

Another complexity illustrated by the real-world deployment scenario involves the policy implication with multiple States participating in a breed and burn fleet. Per the discussion herein, as part of normal operations, a State may import irradiated reactor fuel containing upwards of 60 SQ of nuclear material per reactor, per refueling cycle. While unirradiated nuclear material is routinely shipped between States, irradiated fuel will lack directly traceable nuclear material quantity measurements (discussed in section 6.2). This will necessitate a protocol for transferring large quantities of nuclear material from one State to another, with incomplete and uncertain information on the quantity of nuclear material involved. This will only be complicated by transfers of nuclear material between States that have implemented different types of safeguards agreements. The foundation of a real-world breed and burn fleet may rely on State-to-State agreements, but the IAEA must consider the oversight required in moving multiple SQs of Pu between States.

7. Conclusion and Recommendations

The concept of breed and burn reactors and the use of core spawning are attractive design features for improvement of uranium utilization without reprocessing and potential reduction of nuclear fuel cycle costs. If successful, the eventual system would have a high utilization of uranium resources (approximately 20 to 30 times more energy generated per ton of ore compared to the current LWR fleet) and would not require enrichment nor reprocessing to sustain operations. While the concept of a fast reactor is well known, a system of fast reactors that constitute a breed and burn fleet presents new safeguards challenges that have not been previously considered. Some of the unique challenges in this study sought to inform upon include core spawning (the transfer of used nuclear fuel irradiated in one reactor for use in another reactor), increased Pu content in UNF, and long-term fuel management.

An example breed and burn reactor was evaluated to collect the necessary material inventories for performing a safeguards evaluation. This information was assessed to identify the areas where traditional safeguards techniques may be applicable. At the time of this study, the most relevant safeguards techniques were found to be a robust C/S regime reliant on CoK for NMAC. These findings were based on observing the IAEA safeguards regime in place during the Monju reactor as well as existing practices for managing spent fuel inventories for power reactors. Further considerations for the breed and burn fleet should be made, however, to include methods of reinforcing existing C/S capabilities and nuclear material quantities reverification if CoK is ever lost. Maintaining CoK requires the collection and maintenance of detailed history data for UNF in the fleet. Appropriate, detailed reporting of operational history is useful for both safeguards implementation and reactor operations.

Technology for the direct measurement of Pu content in UNF discharged from breed and burn reactors is not currently available. Consequently, determination of nuclear material quantities for reporting will require the use of core depletion calculations. While this is standard practice for many reactors today, the high burnup and correspondingly high SNM concentrations, and large changes in those concentrations, of breed and burn reactor UNF increases the importance of accurate and reliable calculations.

The system of breed and burn fleet reactors is still under development. To date, nontraditional reactors such as those applicable in a fleet are few, and safeguards approaches are developed on a specific case-by-case basis. Full deployment of the technology is likely decades away; however, the use of core spawning could have benefits in any fast reactor system to reduce initial fissile inventory requirements. Therefore, understanding these safeguards challenges allows for development of appropriate methods, techniques, and controls in advance to ensure a robust safeguards regime for this promising technology. Further studies regarding the applicability of safeguards to breed and burn fleets will evolve as the concept continues to develop. Future considerations must be made for increasing fleet size, the implication of multi-State participation, and fundamental operations of an international fleet, among other unique challenges.

This report describes specific concepts that utilize solid-fuel (traditional fuel assemblies in a traditional batch fuel cycle) sodium-cooled fast reactor. The breed and burn system may involve any fast reactor concept for which the only technical requirement is that fuel must remain in the reactor for sufficiently high fluence levels to allow in-situ breeding of sufficient Pu for sustaining criticality. This is beyond current experience. Another concept being actively pursued is the molten salt fast reactor which unlike cladded fuel has no practical fuel integrity limits, but other technical challenges require additional

research and development. TerraPower's molten chloride fast reactor (MCFR) is an example of this technology. There are important differences from a safeguards perspective that need to be addressed. Some likely making safeguards easier, such as the ability to take samples of the fuel and get exact compositions, and others likely more challenging, such as the bulk form fuel is not in itemized assemblies that can be identified and counted. One obvious next step is to look at the safeguards of a breed and burn liquid-fueled reactor. In particular, the core spawning (transfer of irradiated fuel from one reactor to another) will be very different.

This study presented representative reactor concepts which were used to demonstrate initial identification of the safeguards challenges. The next step is to continue analysis on a specific vendor design and develop the techniques and methods to address the identified challenges and close technological gaps. Future work that may be considered from this study includes:

- Developing analysis for other reactor core types, including molten salt fast reactors with liquid-fuel.
- Verify safeguards techniques through application at the facility level.
- Produce comprehensive analysis through the models demonstrated in this study to include additional details such as the implication of Additional Protocols, overall energy consumption, resource limitations, etc., to develop detailed analysis for a breed and burn fleet deployment.
- Determine the uncertainties associated with SNM mass in a realistic deployment scenario through quantitative analysis.

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Appendix A: List of Abbreviations and Acronyms

Assy	Assembly
BOC	Beginning of Cycle
C/S	Containment and Surveillance
CoK	Continuity of Knowledge
CSA	Comprehensive Safeguards Agreement
DA	Destructive Analysis
DU	Depleted Uranium (enrichment tails)
EFPY	Effective Full Power Year
EOC	End of Cycle
FBR	Fast Breeder Reactor
FIMA	Fissions per Initial Metal Atom
GWe	Gigawatt Electric
HM	Heavy Metal
IAEA	International Atomic Energy Agency
IVS	In-Vessel Storage
k_{eff}	k-effective
KMP	Key Measurement Point
LEU	Low-Enriched Uranium
LWR	Light Water Reactor
MBA	Material Balance Area
MT	Metric Tonne
MTHM	Metric Tonne of Initial Heavy Metal
MWth	Megawatt Thermal
NDA	Non-Destructive Analysis

NFCS	Nuclear Fuel Cycle Simulation
NMA	Nuclear Material Accountancy
NMAC	Nuclear Material Accounting and Control
NPT	Nonproliferation Treaty
pcm	percent mille (10^{-5})
SA&UQ	Sensitivity Analysis and Uncertainty Quantification
SFR	Sodium-Cooled Fast Reactor
SNF	Spent Nuclear Fuel
SNM	Special Nuclear Material
SQ	Significant Quantity
TID	Tamper-Indicating Device
UNF	Used Nuclear Fuel

Appendix B: Example Breed and Burn Concept

This Appendix provides additional details of the transition core inventory and discharged assembly content.

Table B.1 shows selected core inventory parameters for the BOC and EOC of cycles 1–35. By cycle 35, the inventories of Pu-239 and significant quantities are converged to the equilibrium values.

Table B.1. Summary Core Inventory: Cycles 1–35

	U-235 (kg)	Total U (kg)	Pu-239 (kg)	Total Pu (kg)	Np-237 (kg)	Am (kg)	Sig. Quantities
BOC 1	10,978	177,641	0	0	0	0	146
EOC 1	9,553	174,570	1,537	1,552	13	0	321
BOC 2	9,185	174,685	1,487	1,502	13	0	310
EOC 2	7,980	171,711	2,889	2,948	32	0	475
BOC 3	7,731	171,904	2,798	2,855	31	0	460
EOC 3	6,717	169,099	3,995	4,123	55	0	605
BOC 4	6,504	169,387	3,865	3,989	53	0	585
EOC 4	5,654	166,739	4,878	5,094	78	0	712
BOC 5	5,471	167,110	4,717	4,926	76	0	689
EOC 5	4,754	164,593	5,581	5,897	102	1	801
BOC 6	4,591	165,030	5,396	5,703	98	1	774
EOC 6	3,977	162,609	6,152	6,577	124	2	875
BOC 7	3,830	163,098	5,949	6,362	120	2	846
EOC 7	3,306	160,758	6,616	7,155	144	3	938
BOC 8	3,179	161,321	6,390	6,914	140	3	907
EOC 8	2,753	159,091	7,000	7,649	162	4	993
BOC 9	2,645	159,730	6,753	7,383	156	4	958
EOC 9	2,279	157,638	7,276	8,029	176	6	1,034
BOC 10	2,198	158,286	7,014	7,745	170	6	997
EOC 10	1,873	156,203	7,506	8,363	189	8	1,070
BOC 11	1,796	157,000	7,222	8,051	181	8	1,030
EOC 11	1,556	155,060	7,681	8,627	196	10	1,099
BOC 12	1,506	155,900	7,376	8,288	189	10	1,056
EOC 12	1,296	154,084	7,767	8,792	202	13	1,116
BOC 13	1,247	155,061	7,452	8,437	193	13	1,071
EOC 13	1,077	153,297	7,841	8,934	204	15	1,131
BOC 14	1,040	154,368	7,513	8,558	194	15	1,084
EOC 14	893	152,686	7,862	9,011	204	18	1,138
BOC 15	870	153,796	7,530	8,628	195	17	1,090
EOC 15	756	152,189	7,863	9,061	203	20	1,143
BOC 16	740	153,384	7,513	8,651	193	19	1,091
EOC 16	644	151,782	7,844	9,081	200	22	1,144
BOC 17	631	153,028	7,500	8,677	190	22	1,093
EOC 17	554	151,500	7,805	9,079	197	25	1,142
BOC 18	545	152,824	7,446	8,651	187	24	1,089
EOC 18	489	151,343	7,749	9,045	192	27	1,137

	U-235 (kg)	Total U (kg)	Pu-239 (kg)	Total Pu (kg)	Np-237 (kg)	Am (kg)	Sig. Quantities
BOC 19	485	152,767	7,389	8,609	181	25	1,083
EOC 19	443	151,246	7,717	9,030	185	28	1,135
BOC 20	441	152,742	7,353	8,586	175	26	1,079
EOC 20	402	151,162	7,705	9,034	179	29	1,135
BOC 21	405	152,789	7,353	8,599	168	27	1,080
EOC 21	370	151,178	7,725	9,067	171	30	1,138
BOC 22	371	152,916	7,370	8,620	158	27	1,082
EOC 22	343	151,239	7,757	9,110	161	30	1,143
BOC 23	345	153,084	7,415	8,679	147	28	1,090
EOC 23	317	151,405	7,814	9,178	150	31	1,151
BOC 24	320	153,258	7,462	8,730	135	28	1,096
EOC 24	297	151,498	7,876	9,254	139	31	1,161
BOC 25	301	153,404	7,538	8,824	124	28	1,107
EOC 25	280	151,612	7,961	9,356	127	31	1,173
BOC 26	284	153,548	7,611	8,906	113	28	1,117
EOC 26	265	151,690	8,045	9,456	116	32	1,186
BOC 27	269	153,632	7,708	9,026	102	29	1,132
EOC 27	251	151,750	8,141	9,577	105	32	1,200
BOC 28	259	153,566	7,775	9,107	96	29	1,142
EOC 28	242	151,611	8,222	9,678	100	32	1,213
BOC 29	247	153,578	7,883	9,245	85	30	1,159
EOC 29	232	151,640	8,311	9,796	89	33	1,228
BOC 30	236	153,594	7,959	9,343	74	30	1,171
EOC 30	221	151,592	8,394	9,907	78	34	1,241
BOC 31	233	153,322	8,025	9,437	74	31	1,183
EOC 31	219	151,330	8,446	9,987	78	35	1,251
BOC 32	227	153,189	8,081	9,515	69	31	1,192
EOC 32	213	151,167	8,499	10,065	73	35	1,261
BOC 33	223	153,146	8,153	9,619	64	32	1,205
EOC 33	209	151,117	8,561	10,160	68	36	1,273
BOC 34	219	153,054	8,199	9,686	59	33	1,214
EOC 34	206	151,000	8,604	10,226	63	37	1,281
BOC 35	218	152,830	8,239	9,756	59	34	1,222
EOC 35	204	150,758	8,638	10,293	63	38	1,289

Table B.2 shows the detailed core inventory of all actinide isotopes, for the BOC and EOC of cycles 1, 2, and the equilibrium cycle (taken as cycle 100).

Table B.2. Detailed core inventory: Cycle 1, Cycle 2, and Equilibrium Cycle (kg)

	Beginning Cycle 1	End Cycle 1	Beginning Cycle 2	End Cycle 2	Beginning Eq. Cycle	End Eq. Cycle
U-234	0.00E+00	5.46E-01	5.19E-01	9.39E-01	2.25E+00	2.53E+00
U-235	1.10E+04	9.55E+03	9.19E+03	7.98E+03	2.12E+02	2.00E+02
U-236	0.00E+00	2.72E+02	2.60E+02	4.83E+02	3.56E+01	3.71E+01
U-238	1.67E+05	1.65E+05	1.65E+05	1.63E+05	1.50E+05	1.48E+05
Np-237	0.00E+00	1.34E+01	1.29E+01	3.24E+01	6.16E+01	6.51E+01
Pu-236	0.00E+00	1.98E-05	1.81E-05	8.22E-05	4.41E-04	4.74E-04
Pu-238	0.00E+00	4.87E-01	4.66E-01	2.29E+00	3.73E+01	4.05E+01
Pu-239	0.00E+00	1.54E+03	1.49E+03	2.89E+03	8.12E+03	8.49E+03
Pu-240	0.00E+00	1.41E+01	1.36E+01	5.56E+01	1.52E+03	1.63E+03
Pu-241	0.00E+00	1.19E-01	1.14E-01	9.46E-01	1.35E+02	1.46E+02
Pu-242	0.00E+00	7.73E-04	7.47E-04	1.36E-02	2.83E+01	3.14E+01
Am-241	0.00E+00	1.89E-03	2.63E-03	2.90E-02	3.92E+01	4.35E+01
Am-242	0.00E+00	7.62E-06	7.35E-06	2.51E-04	1.90E+00	2.16E+00
Am-243	0.00E+00	3.85E-06	3.72E-06	1.47E-04	3.27E+00	3.70E+00
Cm-242	0.00E+00	1.92E-05	1.47E-05	5.19E-04	8.67E-01	1.16E+00
Cm-243	0.00E+00	4.13E-08	3.98E-08	2.55E-06	4.51E-02	4.96E-02
Cm-244	0.00E+00	5.04E-08	4.85E-08	4.18E-06	8.47E-01	9.69E-01
Cm-245	0.00E+00	3.98E-10	3.86E-10	7.03E-08	1.18E-01	1.36E-01
Cm-246	0.00E+00	1.11E-12	1.07E-12	4.29E-10	1.01E-02	1.19E-02
Fission Products	0.00E+00	1.50E+03	1.44E+03	2.94E+03	1.78E+04	1.93E+04
Sum	1.78E+05	1.78E+05	1.78E+05	1.78E+05	1.78E+05	1.78E+05

Table B.3 shows selected average discharged fuel parameters for cycles 1–35. By cycle 35, the Pu-239 mass and significant quantities per discharged assembly have converged to within 5% of the equilibrium values.

Table B.3. Average Used Nuclear Fuel Discharge: Cycles 1–35

Cycle	U-235 (kg/assy)	Total U (kg/assy)	Pu-239 (kg/assy)	Total Pu (kg/assy)	Np-237 (kg/assy)	Am (kg/assy)	Sig. Quant./ assy	Burnup (% FIMA)
1	32	426	4	4	0	0	0.9	0.0%
2	22	419	8	8	0	0	1.3	0.7%
3	19	411	11	11	0	0	1.7	1.7%
4	16	404	13	14	0	0	2.0	2.6%
5	15	399	15	16	0	0	2.2	3.4%
6	13	395	17	18	0	0	2.4	4.0%
7	12	389	19	20	0	0	2.7	4.9%
8	10	382	21	22	0	0	2.9	5.9%
9	8	381	22	24	0	0	3.1	5.7%
10	7	369	24	26	1	0	3.3	8.0%
11	5	365	25	28	1	0	3.6	8.3%
12	5	354	26	30	1	0	3.8	10.6%
13	4	346	27	31	1	0	4.0	12.0%
14	3	343	28	32	1	0	4.0	12.6%
15	2	336	29	34	1	0	4.3	13.8%
16	2	332	29	34	1	0	4.2	14.9%
17	2	325	30	36	1	0	4.5	15.9%
18	1	317	30	36	1	0	4.6	17.6%
19	1	311	30	37	1	0	4.6	18.9%
20	1	300	29	36	1	0	4.5	21.6%
21	1	291	30	37	1	0	4.7	23.4%
22	1	282	28	36	1	0	4.5	25.8%
23	1	281	29	37	1	0	4.7	25.6%
24	1	277	28	36	1	0	4.5	27.0%
25	1	274	29	37	1	0	4.7	27.2%
26	1	273	28	36	1	0	4.5	27.7%
27	0	284	30	39	1	0	4.9	24.6%
28	1	271	28	36	1	0	4.5	28.1%
29	1	273	29	38	1	0	4.7	27.5%
30	0	291	31	39	0	0	4.9	23.0%
31	0	281	30	39	1	0	4.9	25.4%
32	0	271	29	37	1	0	4.6	28.2%
33	0	274	30	39	1	0	4.9	26.9%
34	0	283	30	39	0	0	4.9	25.0%
35	0	286	32	41	0	0	5.1	23.7%

The reactor core has one-sixth core symmetry. As a result of this symmetry, out of the twelve assemblies discharged at the end of each cycle, there are two sets of unique assembly histories. The detailed contents of each discharged assembly with one of these two unique assembly histories, A and B, are given in Table B.4 for selected cycles. At the end of each cycle, six assemblies with the composition of Assembly A and six with the composition of Assembly B will be discharged.

Table B.4. Detailed discharged assembly content: Cycle 1, Cycle 10, and Equilibrium Cycle

	Cycle 1		Cycle 10		Eq. Cycle	
	Assembly A	Assembly B	Assembly A	Assembly B	Assembly A	Assembly B
U-234	2.26E-03	2.25E-03	1.20E-02	1.23E-02	2.77E-02	2.68E-02
U-235	3.17E+01	3.18E+01	7.42E+00	7.46E+00	5.04E-02	5.46E-02
U-236	9.89E-01	9.83E-01	3.46E+00	3.45E+00	1.24E-01	1.27E-01
U-238	3.93E+02	3.93E+02	3.58E+02	3.58E+02	2.63E+02	2.69E+02
Np-237	4.63E-02	4.59E-02	6.23E-01	6.20E-01	2.95E-01	2.97E-01
Pu-236	8.15E-08	8.02E-08	3.10E-06	2.94E-06	1.45E-06	1.38E-06
Pu-238	1.71E-03	1.69E-03	1.62E-01	1.61E-01	2.82E-01	2.73E-01
Pu-239	4.13E+00	4.09E+00	2.36E+01	2.36E+01	3.05E+01	3.10E+01
Pu-240	3.96E-02	3.90E-02	2.11E+00	2.10E+00	9.28E+00	9.13E+00
Pu-241	3.39E-04	3.32E-04	1.12E-01	1.10E-01	8.54E-01	8.24E-01
Pu-242	2.23E-06	2.17E-06	8.01E-03	7.89E-03	2.68E-01	2.47E-01
Am-241	5.38E-06	5.27E-06	2.20E-02	2.28E-02	4.45E-01	4.29E-01
Am-242	2.20E-08	2.14E-08	6.25E-04	6.28E-04	2.30E-02	2.15E-02
Am-243	1.08E-08	1.05E-08	3.73E-04	3.66E-04	3.76E-02	3.34E-02
Cm-242	5.53E-08	5.38E-08	3.47E-04	3.56E-04	6.05E-03	5.47E-03
Cm-243	1.14E-10	1.11E-10	8.63E-06	7.91E-06	3.79E-04	3.43E-04
Cm-244	1.36E-10	1.30E-10	4.00E-05	3.89E-05	1.05E-02	8.96E-03
Cm-245	1.02E-12	9.70E-13	2.47E-06	2.40E-06	1.67E-03	1.39E-03
Cm-246	2.64E-15	2.50E-15	6.46E-08	6.26E-08	1.74E-04	1.37E-04
Fission Products	5.41E+00	5.37E+00	3.98E+01	3.96E+01	1.30E+02	1.24E+02
Sum	4.35E+02	4.35E+02	4.35E+02	4.35E+02	4.35E+02	4.35E+02

Appendix C: Deployment Modeling

C.1. DYMOND Method

DYMOND is an Argonne National Laboratory-developed and maintained NFCS code. The DYMOND code models a State-level nuclear fuel cycle through a hybrid modeling of paradigms, combining system dynamics, discrete event and agent-based modeling in the AnyLogic modeling platform, and depletion physics through coupling with ORIGEN2. The workflow of DYMOND uses these different paradigms to manage the abstraction level of the nuclear fuel cycle.

Material flows, transport, and head-end processes are modeled as stocks and flows within a system with feedback mechanisms dictated by the higher definition agents. Agent models are used for the reactors and reprocessing with discrete event models dictating their construction, operational state, and decommissioning. The preserved component linking the model abstraction levels is the composition of the materials at key locations in the fuel cycle. DYMOND explicitly tracks 25 nuclides—actinides that have the greatest impact on fuel recycling in advanced reactor concepts—and a lumped material composite of all fission products. The transmutation of nuclear fuel in a reactor is calculated using reactor-specific depletion libraries in ORIGEN2 generated from coupled neutronics-depletion simulations. These calculations determine the fuel composition after irradiation and cooling from a reactor based on the reactor power and cycle specifications. The coupling of ORIGEN2 also allows for the determination of fuel composition requirements through a criticality search as an alternative to the commonly used fuel “recipes” or approximate nuclide reactivity worth (i.e., Pu-239 equivalence). This feature results in more accurate fuel requirements for the reuse of UNF assemblies and the S2 cores that are composed of both fresh LEU assemblies and UNF assemblies.

C.2. Global Sensitivity Analysis

To better understand the nuclear fuel cycle models that are being simulated and the result of the parametric uncertainty inherent to them, we do sensitivity analysis and uncertainty quantification (SA&UQ). These approaches reveal how the system performance changes as either a single or group of uncertain parameters change. This relationship is studied through sampling methods and surrogate models to calculate a distribution on system performance measures (i.e., response metrics) and to understand which parameters contribute most to the variance of the response and the synergistic relationships between parameters. To perform this analysis a framework based on DAKOTA (Design Analysis Kit for Optimization and Terascale Application) [Adams 2014] is interfaced with DYMOND. The analysis framework allows us to leverage this reduction in model structural uncertainty to better understand the parametric uncertainty through a type of global sensitivity measure called Sobol indices.

Sobol indices give a measure of the contribution of each parameter to the variance observed in a response metric. In the case of DACE (design and analysis of computer experiments) with deterministic models, the variance is the same as the range of values of the output that are possible given a range of values of the inputs, i.e., the importance of a parameter to a response metric. There are two measures provided by Sobol indices, the main effect index and the total effect index. The main effect index indicates the fraction of the variance that a parameter is solely responsible for. This is measured by the ratio of variance that is observed in the response metric at each point that the parameter is constant, averaged over all other parameters, to the total variance in the response. The total effect index is a measure that is a fraction of the main effect of the parameter and all synergistic effects that parameter has with the other parameters. In this context, a synergistic effect may be a constructive or destructive

interaction of any order between two or more parameters that results in the behavior of the response metric not being purely additive from the main effects of the constituent parameters. The total effect is measured as the ratio of the average variance in the response, when only the parameter being measured is allowed to vary, to the total variance in the response. This measurement is akin to a quantification of the average behavior that would be observed from doing many parametric studies on the parameter being measured at different sets of values of the other parameters.

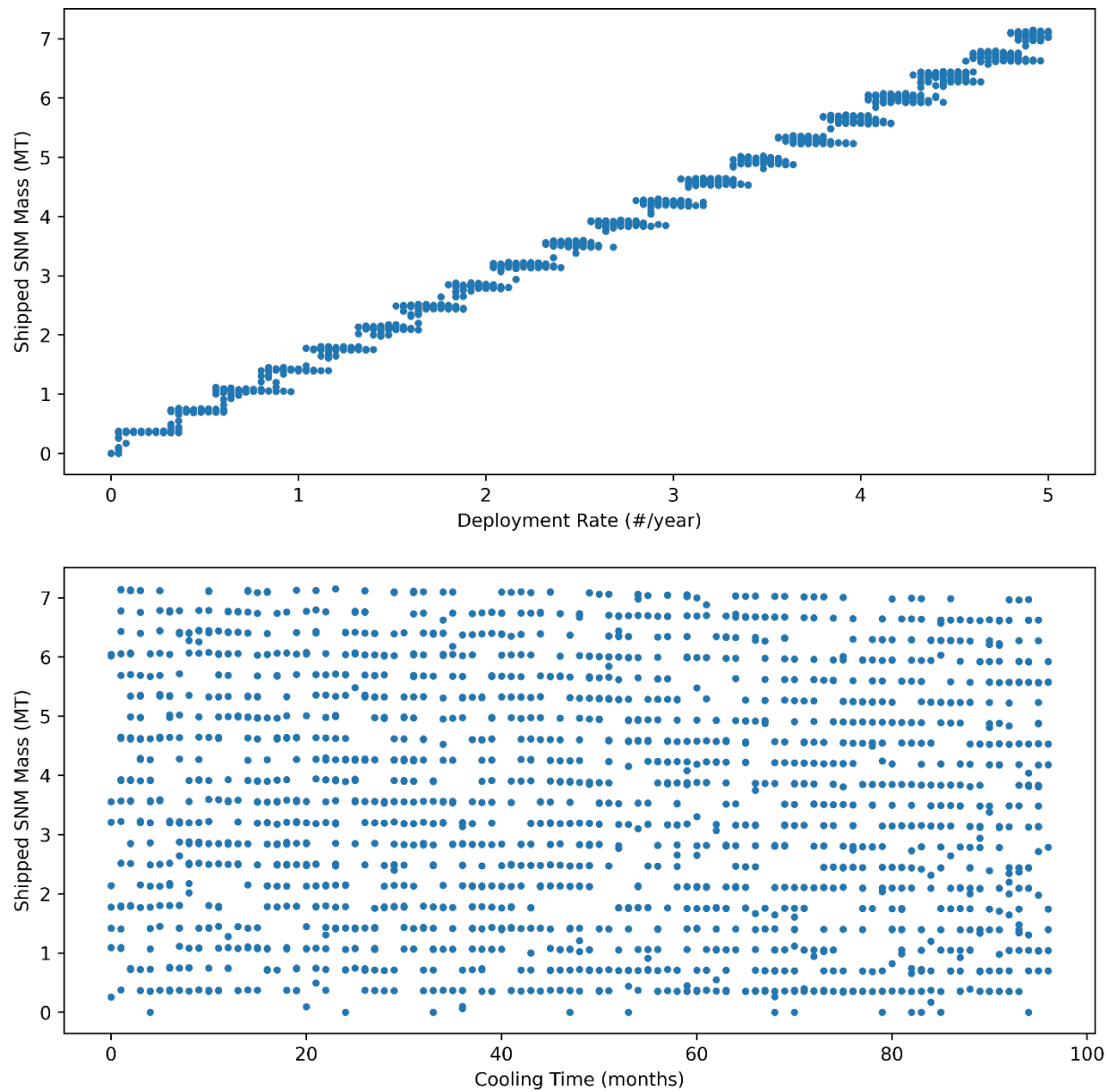
In the case of this global sensitivity analysis in Section 4.2, the Sobol indices are presented in table C.1. The indices reported in this table were calculated using a quadratic regression surrogate model and confirmed with a Gaussian process surrogate model, both of which were trained using the 2,000 direct model samples and then sampled 10,000 times. The quality metrics and sampling results for both models are given in Appendix C.3.

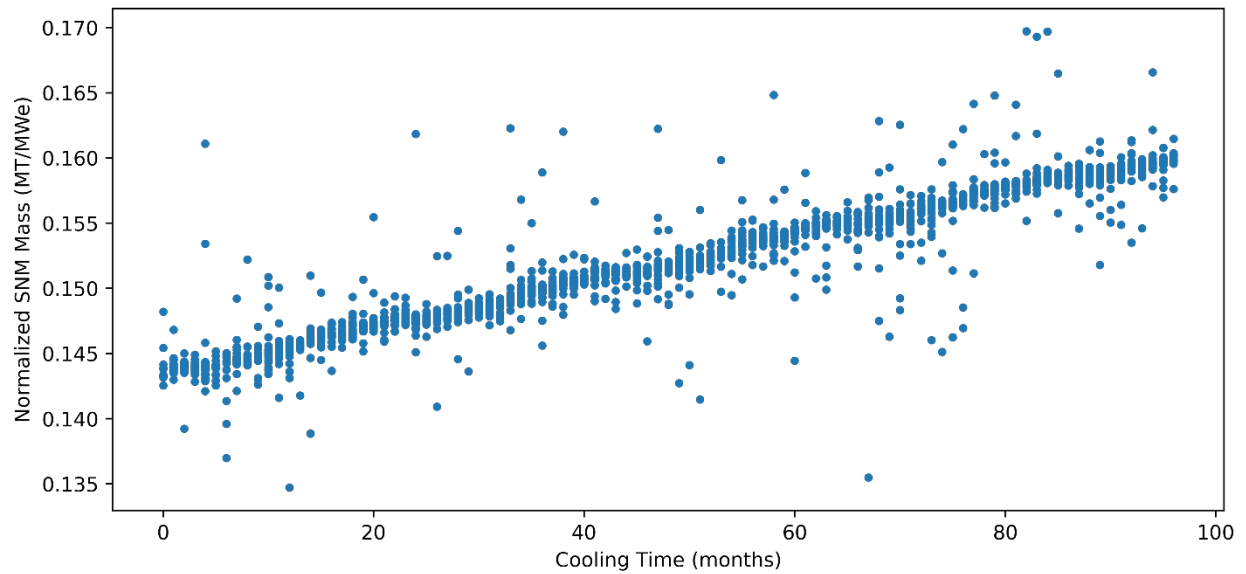
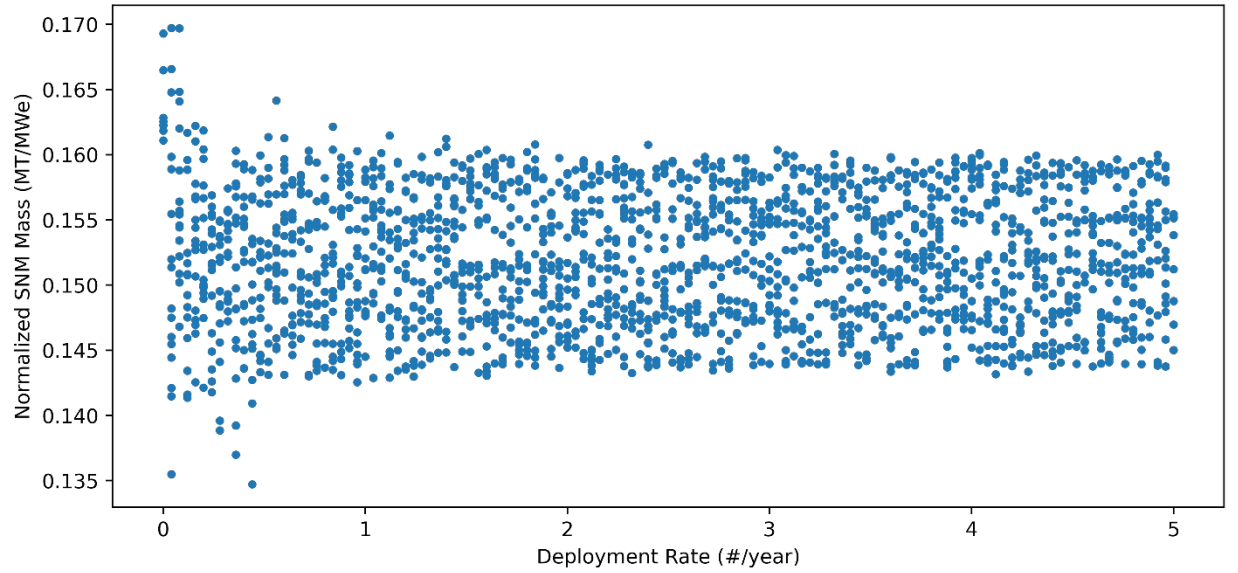
Table C.1. Main and total effect Sobol indices indicating the dominant parameter for each response

Response	Metric	Main Effect Index	Total Effect Index
Normalized Fuel Cycle Backend SNM Mass	Deployment Rate	5.41×10^{-03}	3.67×10^{-02}
	Minimum Cooling Time	9.60×10^{-01}	9.93×10^{-01}
Equilibrium Shipment SNM Mass	Deployment Rate	9.02×10^{-01}	9.95×10^{-01}
	Minimum Cooling Time	4.01×10^{-03}	1.06×10^{-01}

C.3. Model Sampling Results

DYMOND Sampling Results

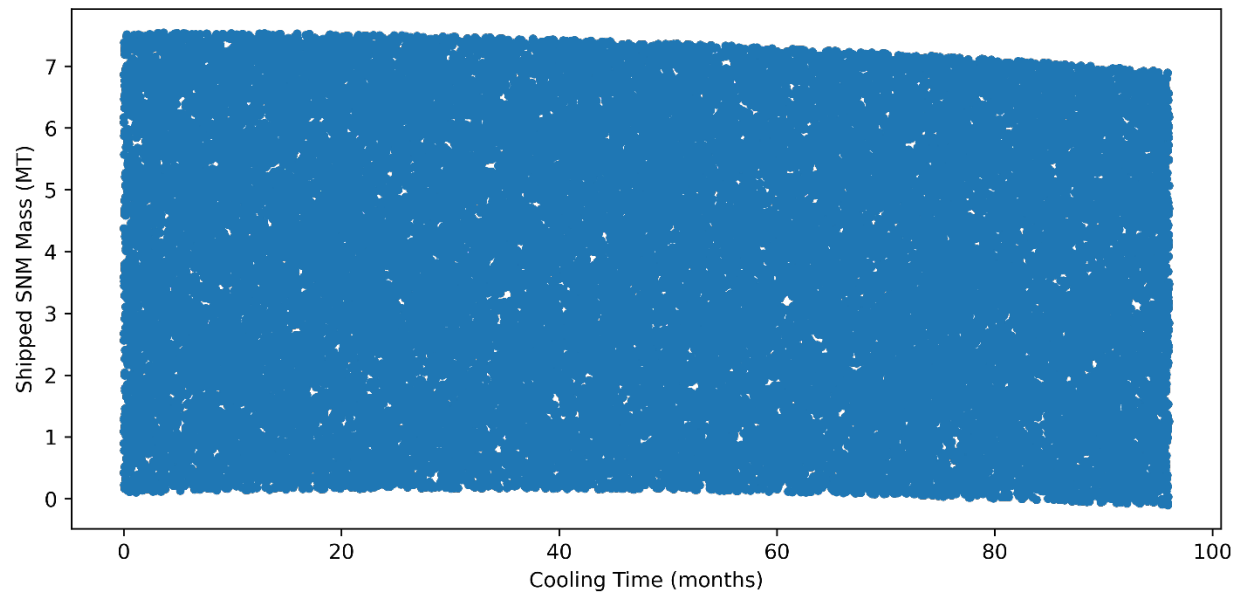
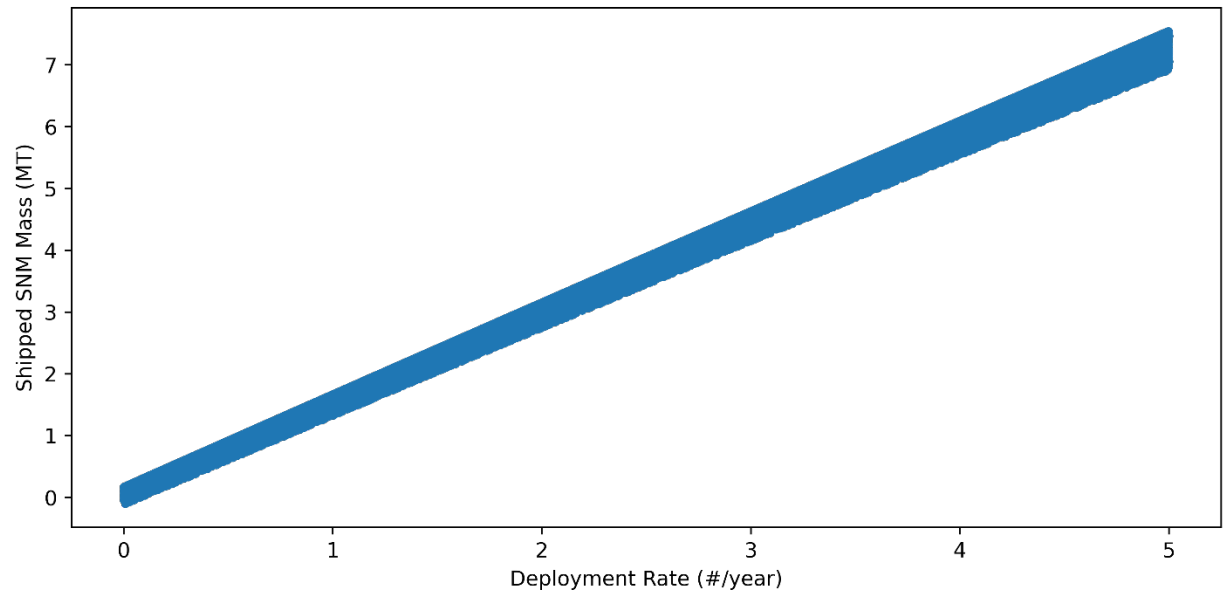


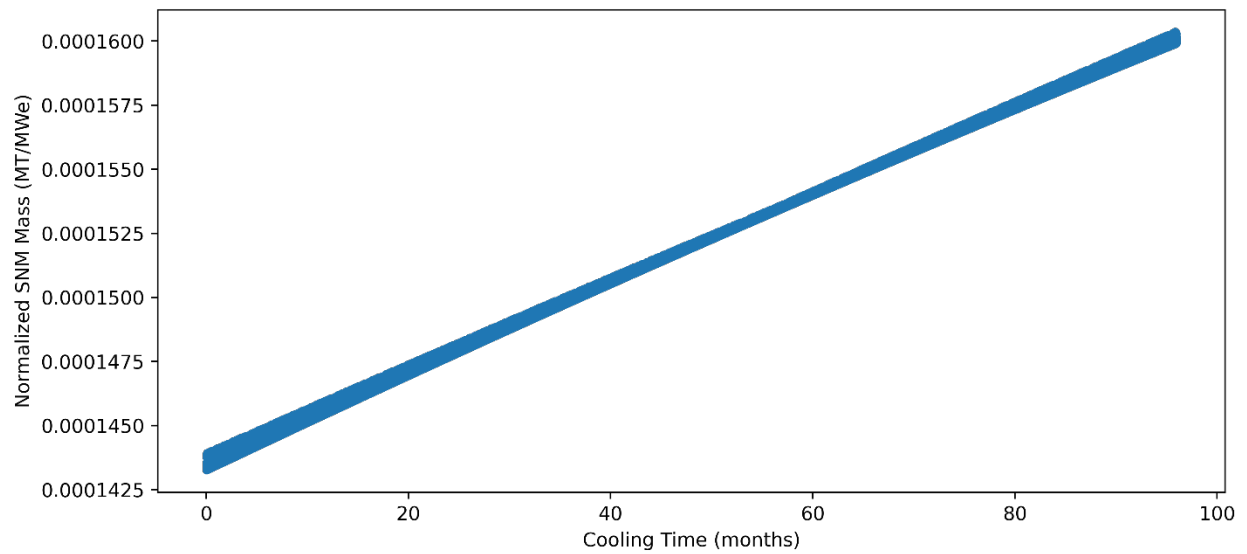
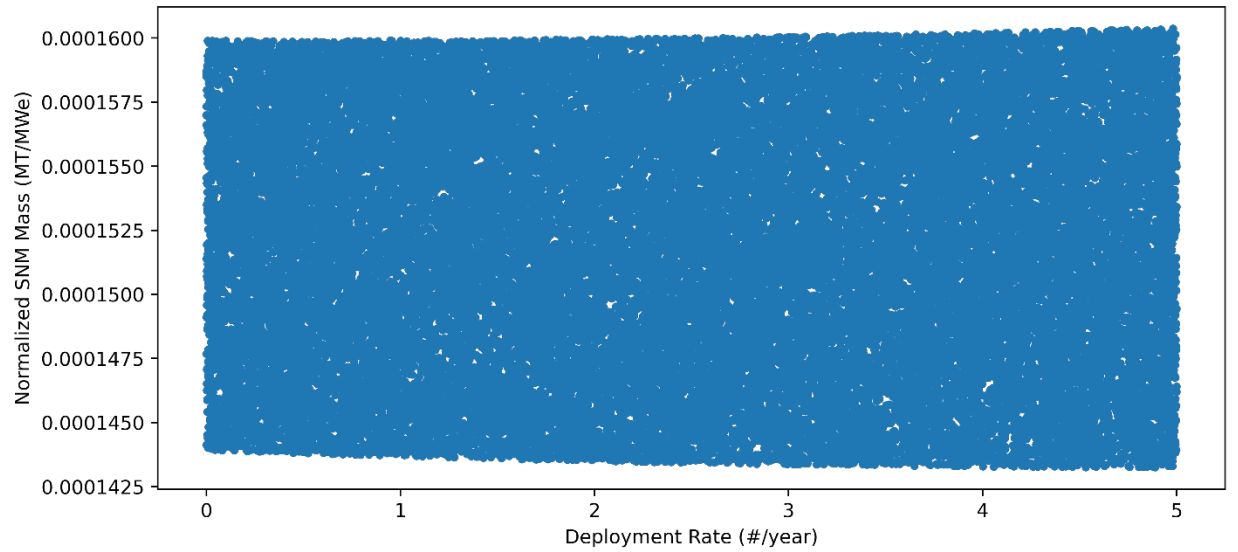


Quadratic Regression Surrogate Model Sampling Results

Table C.2 Model Fit Quality for Quadratic Regression Surrogate Model

Normalized Fuel Cycle Backend SNM Mass	Root Mean Squared	1.737×10^{-6}
	Sum of Absolute Error	1.42×10^{-3}
	R squared	0.883
Equilibrium Shipment SNM Mass	Root Mean Squared	1.216
	Sum of Absolute Error	487.97
	R squared	0.746

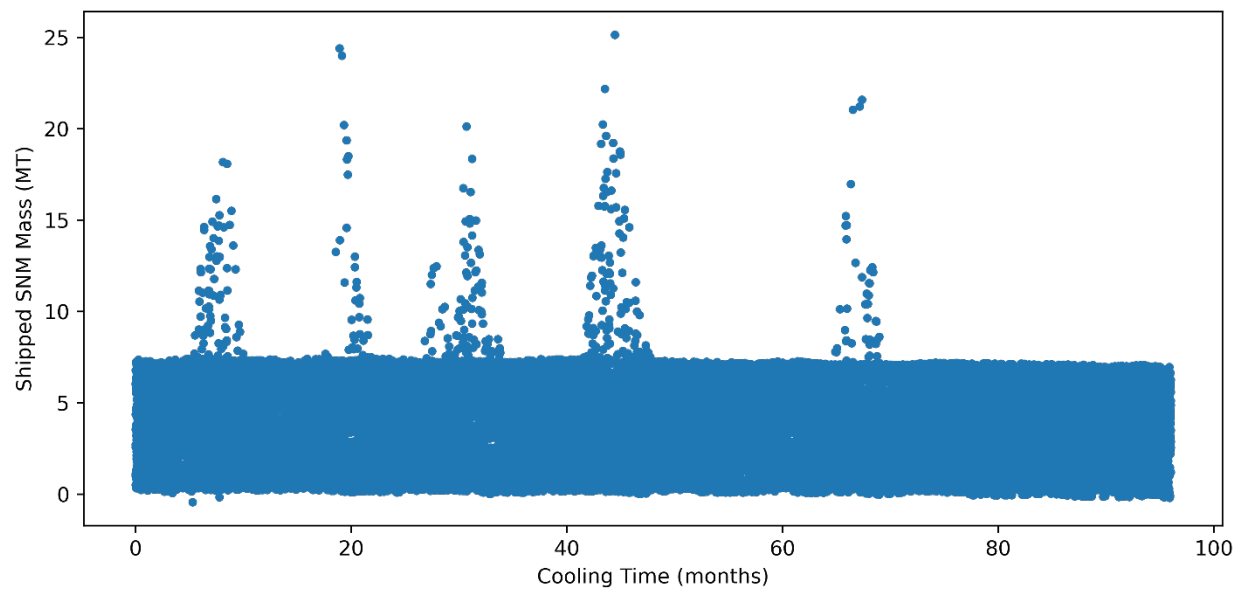
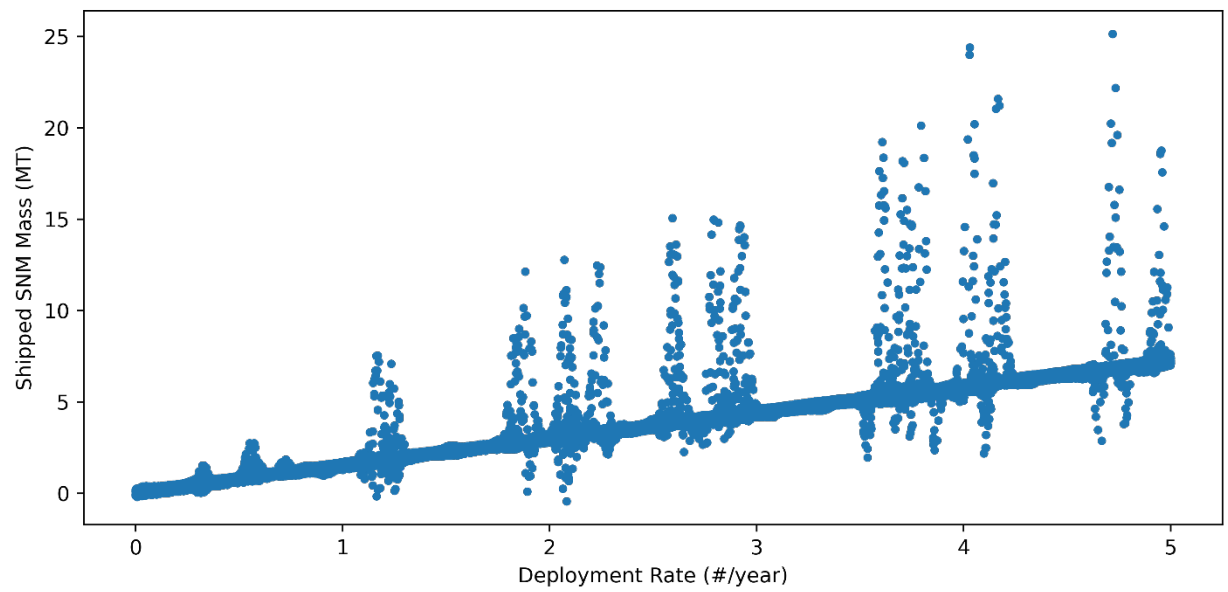


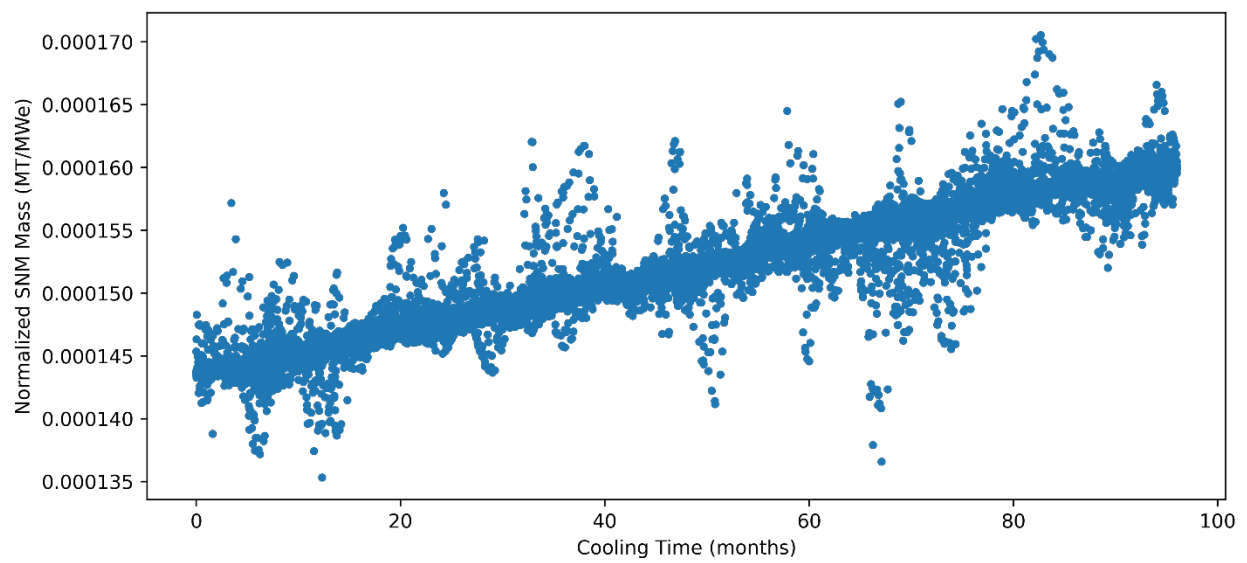
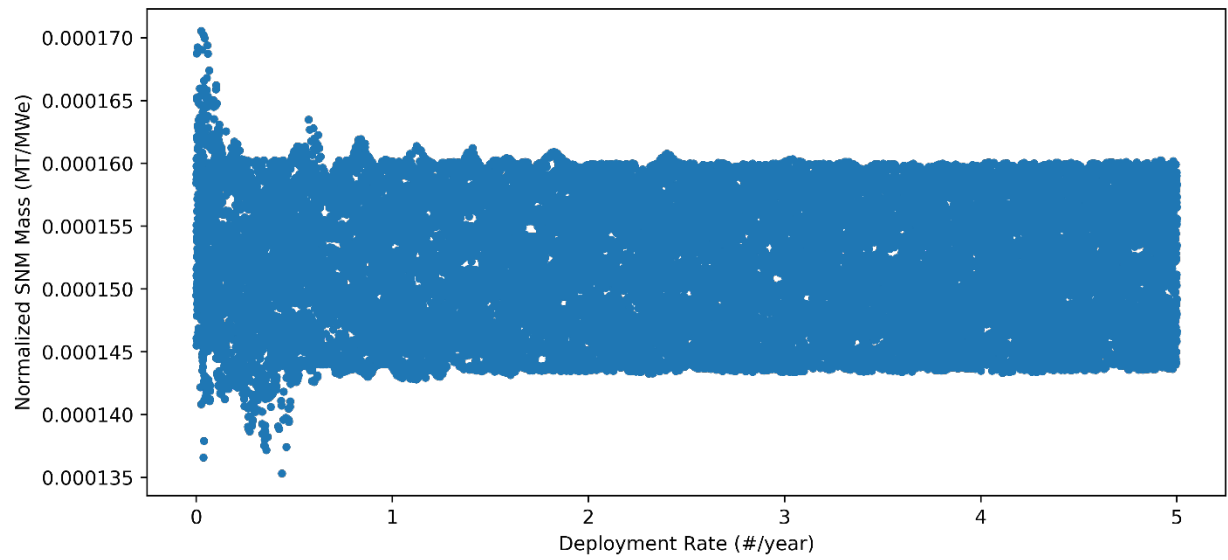


Gaussian Process Surrogate Model Sampling Results

Table C.3 Model Fit Quality for Gaussian Process Surrogate Model

Normalized Fuel Cycle Backend SNM Mass	Root Mean Squared	0
	Sum of Absolute Error	0
	R squared	1
Equilibrium Shipment SNM Mass	Root Mean Squared	1.325×10^{-15}
	Sum of Absolute Error	1.814×10^{-12}
	R squared	1





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