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GIF MOLTEN SALT REACTOR

Proliferation Resistance and Physical Protection White Paper

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Preface to the 2021-2022 edition of the SSCs, pSSCs & PRPPWG white papers on the PR&PP features of the six GIF technologies

This report is part of a series of six white papers, prepared jointly by the Proliferation Resistance and Physical Protection Working Group (PRPPWG) and the six System Steering Committees (SSCs) and provisional System Steering Committees (pSSCs). This publication is an update to a similar series published in 2011 presenting the status of Proliferation Resistance & Physical Protection (PR&PP) characteristics for each of the six systems selected by the Generation IV International Forum (GIF) for further research and development, namely: the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Molten Salt Reactor (MSR), the Sodium-cooled Fast Reactor (SFR), the Super Critical Water–cooled Reactor (SCWR) and the Very High Temperature Reactor (VHTR).

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) was established by GIF to develop, implement and foster the use of an evaluation methodology to assess Generation IV nuclear energy systems with respect to the GIF PR&PP goal, whereby: "*Generation IV nuclear energy systems will increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism"*.

The methodology provides designers and policy makers a technology neutral framework and a formal comprehensive approach to evaluate, through measures and metrics, the Proliferation Resistance (PR) and Physical Protection (PP) characteristics of advanced nuclear systems. As such, the application of the evaluation methodology offers opportunities to improve the PR and PP robustness of system concepts throughout their development cycle starting from the early design phases according to the PR&PP by design philosophy. The working group released the current version (Revision 6) of the methodology for general distribution in 2011. The methodology has been applied in a number of studies and the PRPPWG maintains a bibliography of official reports and publications, applications and related studies in the PR&PP domain.

In parallel, the PRPPWG, through a series of workshops, began interaction with the Systems Steering Committees (SSCs) and Provisional Systems Steering Committees (pSSCs) of the six GIF concepts. White papers on the PR&PP features of each of the six GIF technologies were developed collaboratively between the PRPPWG and the SSCs/pSSCs according to a common template. The intent was to generate preliminary information about the PR&PP merits of each system and to recommend directions for optimizing its PR&PP performance. The initial release of the white papers was published by GIF in 2011 as individual chapters in a compendium report.

In April 2017, as a result of a consultation with all the GIF SSCs and pSSCs, a joint workshop was organized and hosted at OECD-NEA in Paris. During two days of technical discussions, the advancements in the six GIF designs were presented, the PR&PP evaluation methodology was illustrated together with its case study and other applications in national programmes. The need to update the 2011 white papers emerged from the discussions and was agreed by all parties and officially launched at the PRPPWG meeting held at the EC Joint Research Centre in Ispra (IT) in November 2017.

The current update reflects changes in designs, new tracks added, and advancements in designing the six GIF systems with enhanced intrinsic PR&PP features and in a better understating of the PR&PP concepts. The update uses a revised common template. The template entails elements of the PR&PP evaluation methodology and allows a systematic discussion of the system elements of the proposed design concepts, the potential proliferation and physical protection targets, and the response of the concepts to threats posed by a national actor (diversion & misuse, breakout and replication of the technology in clandestine facilities), or by a subnational/terrorist group (theft of material or sabotage).

The SSCs and pSSC representatives were invited to attend PRPPWG meetings, where progress on the white papers was discussed in dedicated sessions. A session with all the SSCs and pSSCs was organized in Paris in October 2018 on the sideline of the GIF 2018 Symposium. A drafting and reviewing meeting on all the papers was held at Brookhaven National Laboratory in Upton, NY (US) in November 2019, followed by a virtual meeting in December 2020 to discuss all six drafts.

Individual white papers, after endorsement by both the PRPPWG and the responsible SSC/pSSC, are transmitted to the Expert Group (EG) and Policy Group (PG) of GIF for approval and publication as a GIF document. Cross-cutting PR&PP aspects that transcend all six GIF systems are also being updated and will be published as a companion report to the six white papers.

Abstract

This document represents the status of Proliferation Resistance and Physical Protection (PR&PP) characteristics for the Molten Salt Reactor designs selected by the Generation IV International Forum (GIF) Molten Salt Reactor (MSR) provisional System Steering Committee (pSSC) as representatives of three broad classes of MSRs. The three classes of MSRs are: (1) Liquid-fueled with integrated salt processing, (2) Liquid-fueled without integrated salt processing, (3) Solid-fueled with salt coolant. The intent is to generate preliminary information about the PR&PP merits of the MSR Reactor Technology and to provide insights for optimizing their PR&PP performance for the benefit of MSR system designers. It updates the MSR analysis published in the 2011 report "Proliferation Resistance and Physical Protection of the Six Generation IV Nuclear Energy Systems", prepared Jointly by the Proliferation Resistance and Physical Protection Working Group (PRPPWG) and the System Steering Committees of the Generation IV International Forum, taking into account the evolution of both the systems and the GIF R&D activities since its publication.

The document, prepared jointly by the GIF PRPPWG and the GIF MSR pSSC, follows the high-level paradigm of the GIF Proliferation Resistance and Physical Protection Evaluation Methodology to investigate the Proliferation Resistance and Physical Protection features of the GIF MSR representative designs. For PR, the document analyzes and discusses the proliferation resistance aspects in terms of robustness against State-based threats associated with diversion of materials, misuse of facilities, breakout scenarios, and production in clandestine facilities. Similarly, for physical protection, the document discusses the robustness against theft of material and sabotage by non-State actors.

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(This is an update of the 2011 white paper authored by D. Heuer, C. Le Brun, E. Merle-Lucotte, C. Renault)

Acknowledgements

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List of Acronyms

CNEC	China Nuclear Engineering & Construction Group
CoK	Continuity of Knowledge
C/S	Containment/Surveillance
DIV	Design Information Verification
GA	General Atomics
GIF	Generation-IV International Forum
GTHTR300C	Gas Turbine High Temperature Reactor 300 for Cogeneration
GT-MHR	Gas-Turbine Modular Helium Reactor
HALEU	High-Assay Low-Enriched Uranium
HEU	Highly Enriched Uranium
HTTR	High Temperature Engineering Test Reactor
HTR	High Temperature Reactor
HTR-PM	High-Temperature Gas-cooled Reactor Pebble-Bed Module
HTR-TN	High-Temperature Reactor-Technology Network
IAEA	International Atomic Energy Agency
INET	Tsinghua University's Institute of Nuclear and New Energy Technology
JAEA	Japan Atomic Energy Agency
KAERI	Korea Atomic Energy Research Institute
KI	Kurchatov Institute
LEU	Low Enriched Uranium
LWR	Light Water Reactor
MA	Minor Actinides
MOX	Mixed Oxide
NED	Nuclear Explosive Device
NHDD	Nuclear Hydrogen Development and Demonstration
NNSA	National Nuclear Security Administration
OKBM	Experimental Design Bureau of Mechanical Engineering in Nizhniy-Novgorod
PBMR	Pebble Bed Modular Reactor
PP	Physical Protection
PR	Proliferation Resistance
PR&PP	Proliferation Resistance & Physical Protection
PWR	Pressurized Water Reactor
RCCS	Reactor Cavity Cooling System
RDD	Radiological Dispersal Device
PCU	Power Conversion Unit
SQ	Significant Quantity
SC-HTGR	Steam Cycle High-Temperature Gas-Cooled Reactor

SSC	System Steering Committee
TRISO	Tri-Isotopic
UOX	Uranium Oxide
VHTR	Very-High-Temperature Reactor

1. Overview of the Technology

Molten Salt Reactors (MSRs) have seen a resurgence of interest in the past decade around the world. Support for these activities is provided from both national and private sources. The largest difference from the 2011 GIF MSR PR&PP evaluation[1] consequently is the transition from evaluating academic systems focused on exploring the technical potential of MSRs to those of companies and countries focusing on deployment.

A wide variety of designs currently exist ranging from solid to liquid-fueled designs, with salt processing on-site or off-site, and a variety of fuel choices [1, 2]. As such, the proliferation resistance and physical protection (PR&PP) aspects will have significantly more variation depending on reactor design than the other advanced reactors. This document, prepared jointly by the GIF PRPPWG and the GIF MSR pSSC, follows the high-level paradigm of the GIF Proliferation Resistance and Physical Protection Evaluation Methodology[2] to investigate the PR&PP features of the GIF MSR representative designs. The rapid introduction and evolution of innovative MSR designs inevitably means that technology specific details of overview reports, such as this one, become rapidly outdated. Consequently, this report focuses on essential features required for any MSR rather than specific design aspects.

MSRs were originally intended as liquid-fueled reactors with liquid fuel processing connected to the fuel salt circuit. Between 1949 and 1976, an MSR development program was conducted in the United States [3, 4]. Two test reactors (the Aircraft Reactor Experiment (ARE) [5] and the Molten Salt Reactor Experiment (MSRE) [6, 7]), four hot-critical assemblies, and about a dozen in-pile test loops were successfully operated. A preliminary design of a 1000-MWe reactor, the Molten Salt Breeder Reactor (MSBR) [8, 9, 10] based on the Th/U^{233} cycle was also completed. Ultimately, the U.S. decided to concentrate on the development of a single breeder reactor concept - the sodium-cooled fast reactor - and development of the MSR was stopped. The substantial U.S. investment in MSR technology created the technology basis of today's MSR resurgence. The 8-MWth MSRE, in particular, provided a remarkably successful demonstration of many aspects of MSR technology.

According to the GIF R&D Outlook: 2018 Update [11], the MSR reference concepts that were then under development within the GIF framework are the liquid fuel Molten Salt Fast Reactor (MSFR) (European Union) and Molten Salt Actinide Recycler and Transmuter (MOSART) (Russia), and the solid-fuel Fluoride-salt-cooled High-temperature Reactor (FHR) demonstration reactor (DR) (United States). More than double the 2018 tabulation of MSRs are, however, currently under some phase of development. The MSFR is one of the example designs considered in this PR&PP study. Appended to this study is some discussion related to the PR&PP aspects of the MOSART system. All FHR-DR activities have been discontinued to avoid any competition with Kairos Power's commercial FHR design. This study uses the University of California at Berkeley's Mk1 PB-FHR, which was the origin of the Kairos power design, as an example system for the salt-cooled variant of the MSR concepts.

Liquid fuels and on-site processing are fundamentally different from a solid fuel reactor where separate facilities produce the fresh solid fuel and process the Spent Nuclear Fuel (SNF). Due to the lack of radiation damage to the fuel, the ability to compensate for fission product build-up, and the wide solubility range of fissile isotopes in halide salts, MSRs can operate with widely varying fuel compositions. Because the choice of fuel cycle affects the safeguards and non-proliferation characteristics of the reactor system, different MSR concepts may have

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[1] "Proliferation Resistance and Physical Protection of the Six Generation IV Nuclear Energy Systems," The Proliferation Resistance and Physical Protection Working Group and the System Steering Committees of the Generation IV International Forum, GIF, 2011.

[2] "Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems - Revision 6," The Proliferation Resistance and Physical Protection Evaluation Methodology Working Group of the Generation IV International Forum, September 2011. (https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/gif_prppem_rev6_final.pdf)

different proliferation resistance and physical protection characteristics. A case in point is an MSR design variant that uses solid fuel but cooled by liquid salt, e.g. the Fluoride-Salt-Cooled High-Temperature Reactor (FHR). From the viewpoint of fuel and fuel cycle, the FHR concept is very close to the Very High Temperature Reactor (VHTR) design concepts and has to be analyzed separately from liquid-fueled MSRs.

The variety of fuel salt configurations illustrates the challenge in generalizing MSR evaluation metrics. One set of proposed MSRs relies on fuel salt in tubes that circulates within the tubes by natural convection while employing forced circulation cooling on the exterior of the fuel tubes. Another set employs an integral fuel salt configuration where the primary heat exchangers are located ex-core within the reactor vessel with both the fuel and coolant salt employing forced circulation. Still others employ a vessel and heat exchanger in a loop configuration within a guard vessel blending features of integral and loop configurations. Two fluid MSRs employ separate fuel and fertile salt circuits that are processed independently. As a further complication, some of the plants include fuel salt drain tanks intended to provide decay heat removal while others are designed to provide decay heat removal while the salt is maintained within the reactor vessel.

Some lessons learned from the prior molten salt breeder reactor program are reflected in all of the new designs. Interior reflectors/shielding or an unfuelled (fertile) downcomer (breeding blanket) are typically employed to reduce the radiation damage to the reactor vessel, and fuel salt chemistry control is employed to substantially limit oxidizing the container alloy constituents. However, even with the vessel interior shielding, the containment environment around both solid and liquid fueled MSRs during operation is likely to have substantially higher dose rates than at LWRs due to limited shielding of fission process and fission products (especially gaseous fission produces) in the case of circulating liquid fueled reactors, and the short-lived activation products of fluorine (^{16}N, ^{20}F, and ^{19}O) in the case of FHRs. MSR designs with insufficient shielding from the fission and/or activation products containments will be remote access only or remote access only during operation.

Recently, private development of advanced reactors has led to a resurgence of MSRs with designs that vary considerably from the MSRE. The following sections break up the various MSR designs into three general classes. The three classes were chosen since each class will have unique materials accountancy and physical protection challenges:

- Liquid-fueled with integrated salt processing (including two fluid MSRs) – onsite salt processing is part of normal operation and fissile materials are both removed from and reintroduced to the active salt circuit.
- Liquid-fueled without integrated chemical fuel salt processing - only the materials that inherently separate from the liquid fuel salt (gases and insoluble (noble) metals) are removed, but additional dissolved elements are not chemically separated onsite as part of the normal operation.
- Solid-fueled with salt coolant.

Several MSR designs do not fit cleanly into a single class and may even change proliferation relevant characteristics based upon non-obvious changes in non-actinide isotope ratios in their feedstock materials. For example, chloride salt reactors will generate sulfur through (n,p) reactions. Chlorine-35 (^{35}Cl) has a substantially higher sulfur production cross section than ^{37}Cl. Sulfur is extremely corrosive to engineering materials in liquid halide salts, so would need to be removed. The presence of sulfur, also, promotes the formation of uranium sulfide, which can precipitate from the fuel salt. The removal of sulfur from the fuel salt, thus, prevents separation of fissile material from the fuel salt (i.e., blocking a potential diversion pathway). The same reactor operating with a different chlorine isotope ratio may or may not be considered to require chemical fuel salt processing and thus fit into different evaluation categories. In addition, to providing a pathway for separation of uranium, shifting the chlorine isotope ratio towards ^{37}Cl increases plutonium breeding both by decreasing parasitic neutron absorption and hardening the spectrum potentially shifting the reactor from a burner to a breeder.

1.1. Liquid-Fueled with Integrated Salt Processing

Several liquid-fueled MSR designs had been publicly disclosed by 2018 including:

- Molten Salt Fast Reactor (MSFR) [12, 13] developed by EURATOM
- Molten Salt Actinide Recycler and Transmuter (MOSART) [14, 15] developed by Kurchatov and other Russian Federation Institutes
- Thorium Molten Salt Reactor Liquid Fuel #1&2 (TMSR-LF1&2) [16] developed by Shanghai Institute of Applied Physics
- Indian Molten Salt Breeder Reactor (IMSBR) [17] developed by Bhabha Atomic Research Centre.
- Liquid Fluoride Thorium Reactor (LFTR) [18] developed by Flibe Energy Inc.
- etc.[3].

While these designs do have considerable variation in fuel choice, one or two fluids, core neutronics, and power density, they all have liquid-fueled cores and propose to perform fissile material separation on-site to enable achieving a closed fuel cycle. The concept of on-site processing also varies as the TMSR-LF designs call for performing some separations on-site and transferring a fuel salt concentrate to another facility to perform additional separations. The plans for the TMSR-LF1 is for the additional facility to be located adjacent to the reactor facility.

The European university development effort has focused on the Molten Salt Fast Reactor (MSFR) using the Th/^{233}U fuel cycle with fluoride salts [20]. The MSFR will be used as the reference for the liquid-fueled designs with integrated salt processing. The concept does not employ a moderator material, thus, resulting in a fast neutron spectrum. The reference MFSR [21] is a 3000 MWth reactor (~1500 MWe) with a total fuel salt volume of 18 m^3, operated at a mean fuel temperature of 725°C.

Figure.1.1 provides a cross section of the MSFR. The core is the central region where nuclear criticality is maintained within the flowing fuel salt. The reactor vessel salt is divided into three regions: the active core, the upper plenum and the lower plenum.

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[3] It is noted that the Dual Fluid Reactor (DFR) [19] developed by the German research institute, the Institute for Solid-State Nuclear Physics (*Institut für Festkörper-Kernphysik)* no longer uses molten salt in the systems (has been replaced by molten metallic fuel).

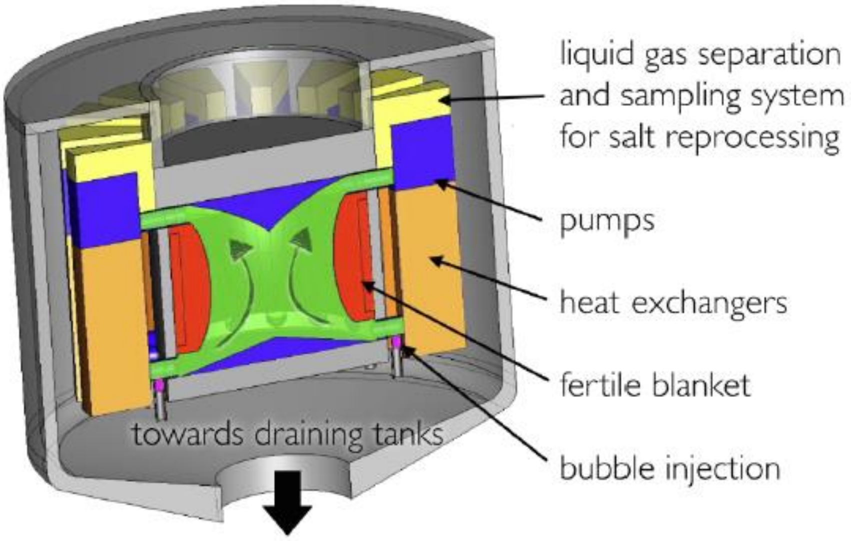

Figure 1.1: Schematic view of the MSFR fuel circuit. [21]

MSFR simulations have been performed using a binary salt, composed of LiF enriched in ^{7}Li to 99.999 % and a heavy nuclei (HN) mixture initially composed of fertile thorium and fissile component, either ^{233}U or Pu. The (HN)F_4 proportion is set at 22.5 mole % (eutectic point), corresponding to a melting temperature of 565°C. The operating temperatures chosen for the neutronic studies range between 700°C and 800°C, the lower limit due to the salt's melting point, the upper limit due to the structural materials, classically Ni-based alloys.

The external core structures and the fuel salt heat exchangers are protected by thick reflectors made of nickel-based alloys, which have been designed to reduce the escaping neutron flux by more than 80%. These reflectors are themselves surrounded by a 10 cm thick layer of B_4C, which provides shielding from the remaining neutrons. In one MSFR design variant, the radial reflector includes a fertile blanket (50 cm thick - red area in Figure 1.1) to increase the breeding ratio. This blanket is filled with a fertile salt of LiF-ThF_4 with 22.5% mol of ^{232}Th.

The normal way to shutdown the reactor and stop the nuclear reaction will be to drain the fuel into tanks located under the core [21].

The fuel salt flows upward in the core until it reaches an extraction area which leads to salt-bubble separators through salt collectors (see description of the gaseous extraction system of fission products in section 2.0). The salt then flows downward in the fuel heat exchangers and the pumps before re-entering the bottom of the core through injectors. The fuel salt runs through the total cycle in around 3-4 seconds, depending on the salt flow velocity. The total fuel salt volume is distributed half in the core and half in the external fuel circuit (salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes). This external fuel circuit is broken up in 16 identical modules distributed around the core, outside the fertile blanket. The fuel circuit, including the core and the external fuel circuit, represents the first barrier for the nuclear fuel and is enclosed in the reactor vessel.

The combination of the reactor vessel and the intermediate circuit of the MSFR is equivalent to the primary circuit of PWRs and represents the second barrier for the nuclear fuel. The reactor vessel and intermediate circuit, together with the intermediate heat exchangers (between the intermediate and the secondary circuits) are enclosed in the reactor building which is the third barrier.

1.2. Liquid-Fueled Without Integrated Salt Processing

The second general class of MSRs do not include fissile materials separation as part of the reactor operation. Only the materials that inherently separate from the liquid fuel salt (gases and insoluble (noble) metals) are removed, but no separations are performed on the dissolved elements on-site as part of the normal operation. In some designs the entire fuel salt, or entire core will be replaced periodically (on the order of 7-8 years), and the salt may be subsequently processed at an external site.

Any core internal materials and the fuel salt container materials will suffer radiation damage and have finite service life and so will require replacement multiple times over the course of the plant life. The fuel salt will be removed from the fuel salt circuit prior to replacing either the core internals or the vessel. The stored fuel salt would be transferred to the replacement fuel salt circuit allowing power generation to continue with the original salt. In this fuel cycle scheme the fuel salt will reach equilibrium concentration of fission products over time as some are burned out while others are produced giving the fuel salt an indefinite lifetime without fissile materials separation.

In the case of burner reactors, additional fissile materials will need to be added to the fuel salt over time resulting in a net increase of fuel salt volume. The additional fuel salt volume would be available as start-up fuel for daughter plants. In the case of breeder reactors additional fertile material will need to be added as fissile material is removed from the fuel salt circuit necessitating an in-containment bred fuel salt storage location. In some designs after the fuel salt becomes saturated with trivalent heavy metals and fission products (e.g. in a thorium converter) the used salt is removed from the reactor, cooled and then shipped for processing and disposal, similar to the batch refueling of the current light-water reactors.

The following liquid-fueled designs without fissile materials separations had been publicly disclosed by 2018:

- Molten Salt Demonstration Reactor (MSDR) [22] developed at Oak Ridge National Laboratory
- ThorCon Reactor [23] developed by Thorcon Power
- Integral Molten Salt Reactor (IMSR) [24] developed by Terrestrial Energy
- Molten Chloride Fast Reactor (MCFR) [25] developed by TerraPower
- Molten Chloride Salt Fast Reactor (MCSFR) [26] developed by Elysium Industries
- Stable Salt Reactor (SSR) [27] developed by Moltex Energy Ltd.
- Compact Molten Salt Reactor (CMSR) developed by Seaborg [https://www.seaborg.com/]
- etc..

As an illustrative example, the Integrated Molten Salt Reactor (IMSR) from Terrestrial Energy [24] consists of a core of graphite moderator rods whose boundaries form channels in which a molten salt containing UF_4 and UF_3 flows upward from a bottom header to a top chimney, then is pumped back down through heat exchangers and a cold return on the outside of the core (see Figure 1.2). The heat exchangers contain a second molten salt, without fissionable materials, which removes the heat from the fuel salt. A gas plenum is provided above the fuel salt where noble fission gases that release from the salt can be initially retained. The core, heat exchangers, pumps inlet plenum, outlet chimney and gas plenum are all contained in a sealed reactor vessel having few penetrations and which is designed to operate for a period of up to 7 years.

The reactor vessel fits inside a guard vessel of similar shape which forms a second containment layer of the system and is capable of holding all the molten salt in case of a core leak. Thermal radiation transfer from the reactor vessel to the guard vessel, itself coupled to an infinite heat sink by a passively coupled flowing gas system, is capable of providing adequate decay heat removal from a subcritical configuration.

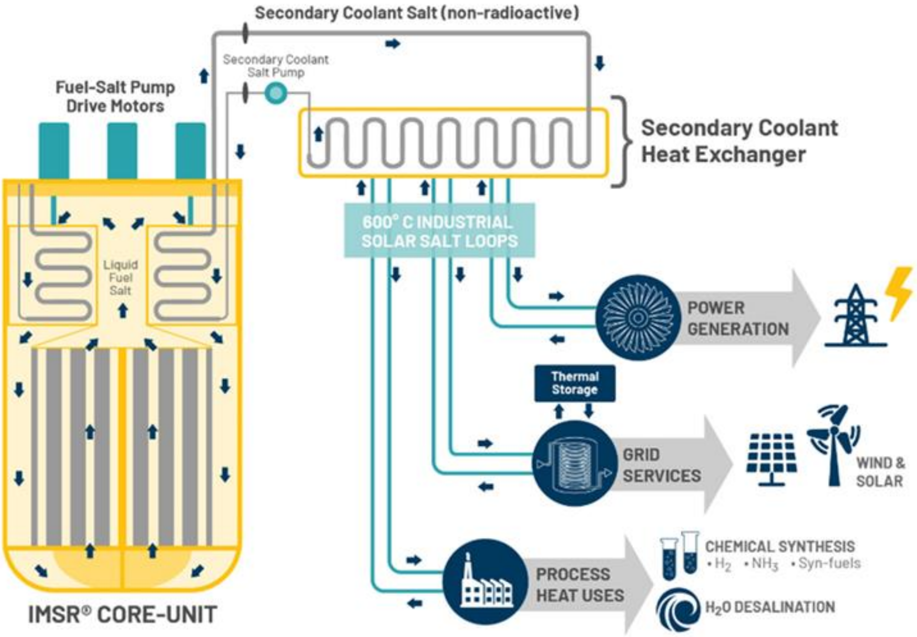

Figure 1.2: Schematic view of IMSR from Terrestrial Energy

An example plant layout for IMSR reactor is shown in Figure 1.3, which features the IMSR being positioned below grade, and with space for storage of multiple (here, up to 4) reactor core units.

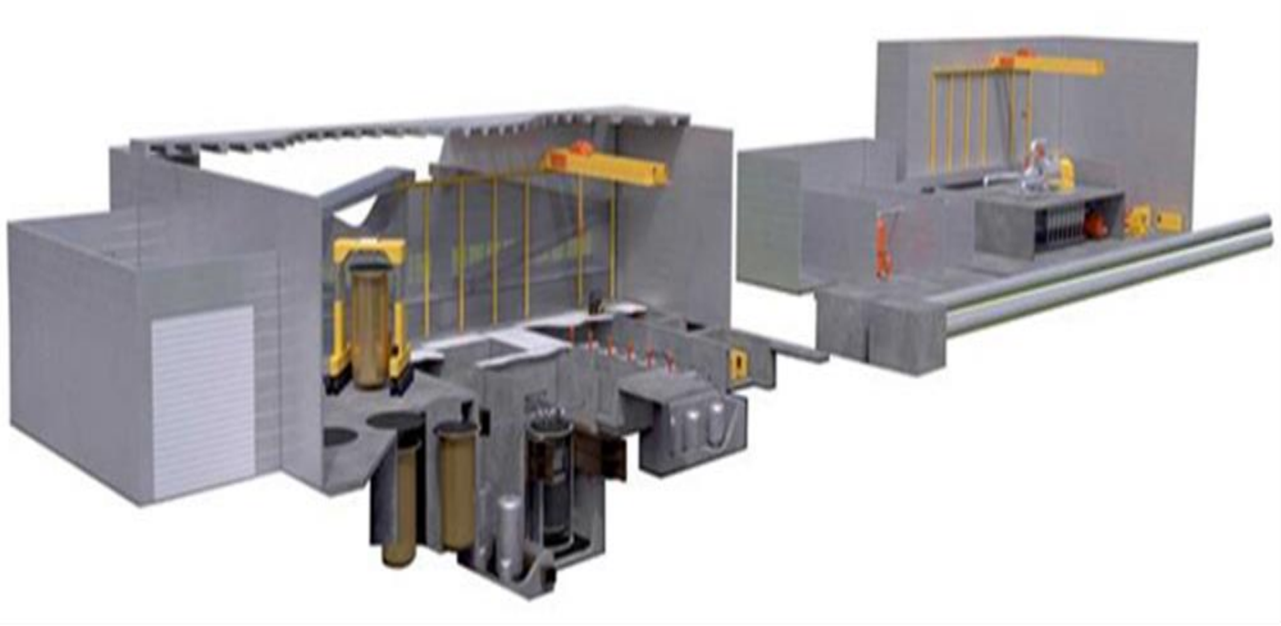

Figure 1.3: Schematic view of IMSR plant layout (from https://www.terrestrialenergy.com/)

In the reference design, provision is made for periodic top-ups of fissile material, pre-mixed with barren molten salt, through a fueling line. Top ups will occur via a system which is engineered not to permit a reverse flow of material. While the fresh startup fuel is uranium enriched to 2% in ^{235}U, the top-up fuel is further enriched to 4.95%. At the end of the 7-year life of the core, there will be an approximately 50% increase in the total volume of the salt [28]. Defueling (into a number of spent fuel storage tanks) in the reference design is through a different line and is intended to be done only once, at the end of the seven-year lifetime of the core unit.

1.3. Solid-Fueled with Molten Salt Coolant

The third general class of MSRs are solid-fueled cores that use molten salts for the coolant. These reactor designs typically use a TRISO fuel in pebble beds with a FLiBe coolant with isotopically separated lithium-7. FLiBe is a molten salt made from a mixture of lithium fluoride (LiF) and beryllium fluoride (BeF_2). Though FLiBe is a solvent for fertile or fissile material, for this class of solid-fueled MSRs the molten salt functions only as a nuclear reactor coolant. The FLiBe coolant provides efficient neutron utilization while maintaining a total net negative reactivity feedback. TRISO fueled fluoride high temperature reactor (FHR) designs tend to be more similar to a high temperature gas reactor. The following are examples of solid-fueled MSR designs:

- Kairos Power Fluoride Salt-Cooled, High Temperature Reactor (KP-FHR) [29] developed by Kairos Power LLC.
- TMSR-SF1 [16]
- Indian High Temperature Reactor [30] developed by Bhabha Atomic Research Centre in India
- Mark-1 Pebble Bed Fluoride High-Temperature Reactor (Mk1 PB-FHR) [31] developed at University of California, Berkeley.

Due to availability of information, the Mk1 PB-FHR will be used as a reference case for this class of MSR [31]. The baseline design is a small modular reactor concept, 100 MWe unit designed for multi-unit deployment per site. The fuel pebbles are 3 cm in diameter containing 4730 TRISO fuel particles at 19.9% enriched uranium, with 1.5 g heavy metal (HM) per pebble. The reactor core and defueling chute contain 470,000 fuel pebbles and 218,000 graphite pebbles. The pebbles are contained in an annular core region with graphite blanket reflector pebbles surrounding the fuel pebbles. The pebbles float in molten salt, so pebble injection occurs at the bottom, and the pebbles slowly move upward with a residence time of about 2.1 months. Defueling occurs at the top of the core. The fuel reaches full depletion (180 MW-d/kg) in 1.4 years. The center region is a graphite reflector with channels for control rods.

The molten salt coolant operates between 600-700 ºC. Each reactor contains 91,900 kg of FLiBe salt. There is no intermediate coolant loop, so the molten salt coolant directly heats the power conversion fluid. A General Electric gas turbine is assumed for power conversion. Tritium control is important and handled by minimizing transport through heat exchangers (tritium filter cartridge and diffusion barrier coating on heat exchanger tubes). Figure 1.4 shows a schematic of the reactor design.

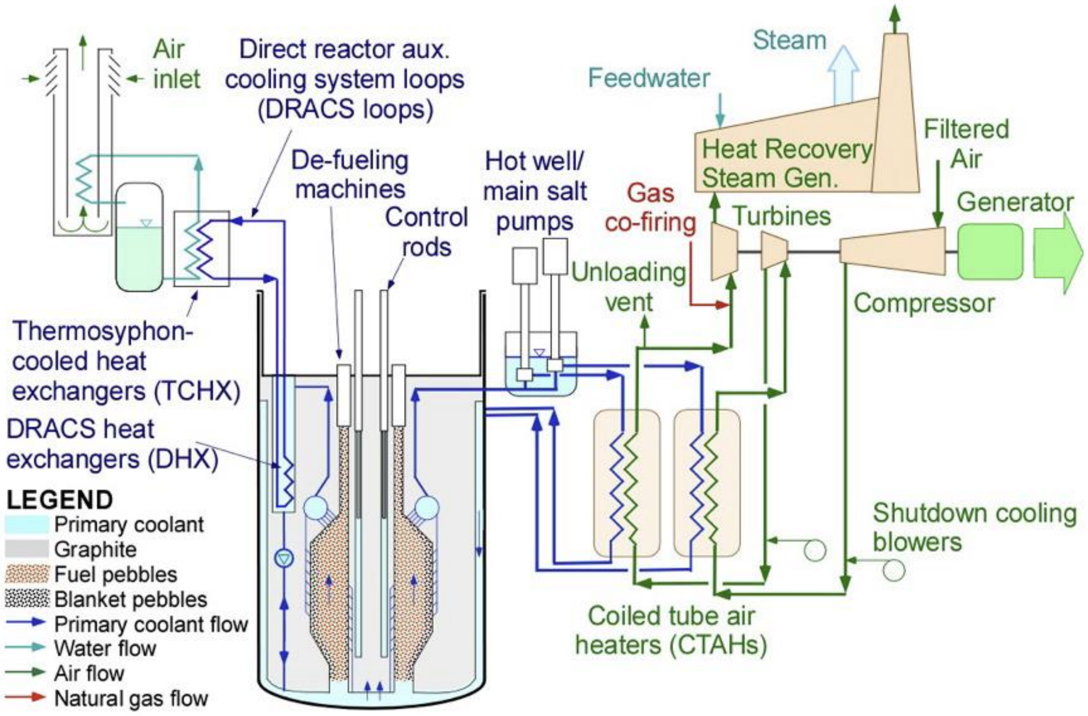

Figure 1.4: Mk1 PB-FHR flow schematic (reproduced from Ref. [31])

The reactor building is slightly below grade with the reactor and all coolant/duct work below grade. A cylindrical shield building provides protection from external missiles and secondary confinement volume in case of contamination. A full site layout is included in reference [31] and includes standard physical protection elements. This layout keeps 12 reactor modules along with the rad waste building, control building, fuel handling and storage, backup generator, and dry cask storage inside a protected area. Additional plant facilities and buildings are located surrounding the protected area. Figure 1.5 shows the plant layout.

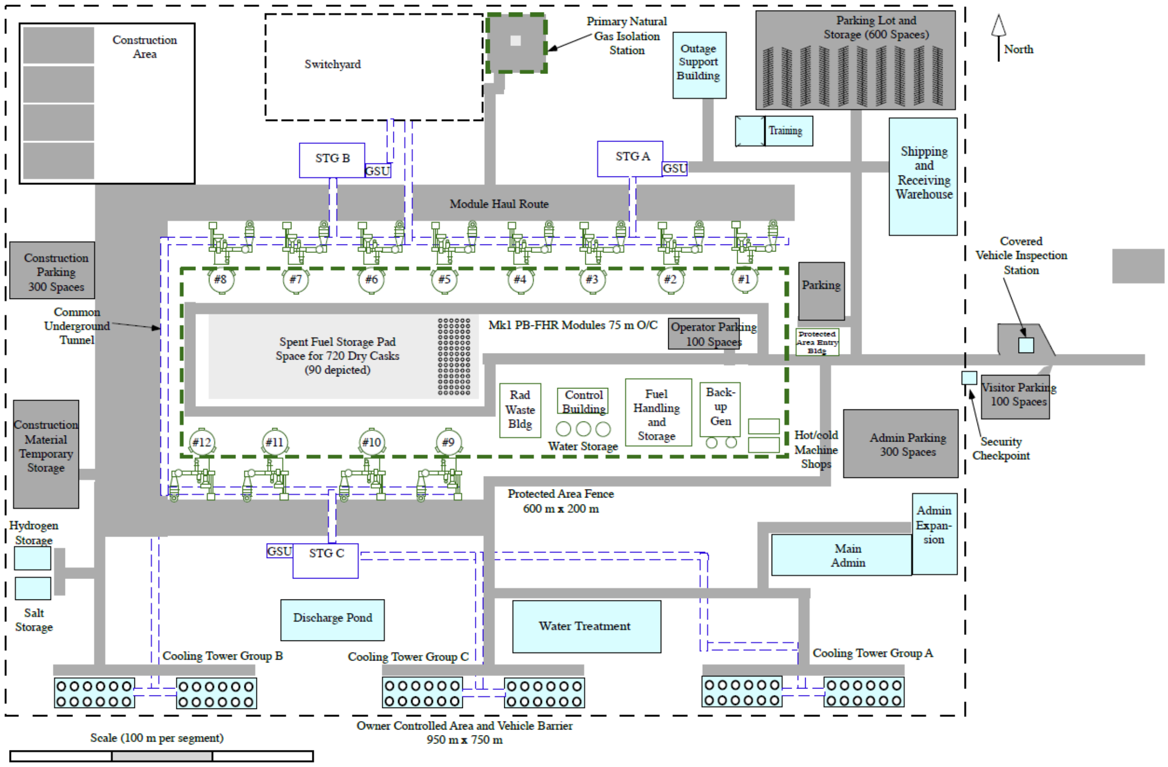

Figure 1.5: Mk1 PB-FHR for a 12-unit plant (reproduced from Ref [31])

1.4. Molten Salt Small Modular Reactor (SMR)

A 2022 booklet [32], a supplement to the IAEA Advanced Reactor Information System (ARIS, http://aris.iaea.org), includes a presentation of SMRs that utilize molten salt fueled and cooled advanced reactor technology. The following table adapted from the ARIS booklet shows that several MSR concepts cited earlier in Sections 1.1-1.3 belong to the SMR category.

Table 1.1: Summary of Molten Salt Small Modular Reactor Designs [32]

Design	Output MWe	Designers	Country	Status
IMSR	195	Terrestrial Energy	Canada	Under Design
smTMSR-400	168	SINAP, CAS,	China	Pre-conceptual Design
CMSR	200	Seaborg	Denmark	Conceptual Design
CA Waste Burner 0.2.5	not defined (100 MWt)	Copenhagen Atomics	Denmark	Detailed design/ Equipment manufacturing in progress
ThorCon	250	ThorCon International	USA & Indonesia	Detailed Design
MSR-FUJI	200	International Molten-Salt Forum: ITMSF	Japan	Conceptual Design

Stable Salt Reactor - Wasteburner	300	Moltex Energy	Canada	Conceptual Design
Stable Salt Reactor - Uranium	16	Moltex Energy	United Kingdom	Basic Design
LFTR	250	Flibe Energy	USA	Conceptual Design
KP-FHR	140	Kairos Power	USA	Conceptual Design
Mk1 PB-FHR	100	University of California, Berkeley	USA	Under Design
MCSFR	50 / 200	Elysium Industries	USA	Conceptual Design

2. Overview of Fuel Cycle(s)

The proposed new MSRs have widely varying fuel cycles to the point that identifying common elements is challenging. Proposed neutron spectra range from very thermal (to minimize uranium enrichment requirements) to very fast (to maximize breeding) and also include time varying spectra. Almost every known form of fissile or fertile material is under consideration as a fuel source. Uranium enrichments as low as 2% and as high as 19.75% have been proposed. Some of the development organizations have published significant information about their proposed fuel cycles (Moltex [33] and Terrestrial Energy [28]) and more information is available about the academic reactors (MSBR, MSDR, MSFR [34] and MOSART [35]). Other developers emphasize the flexibility of their reactor concept to adapt to different fuel feeds but do not provide details of even the in-reactor portion of the fuel cycle. FHRs rely upon high-assay low-enrichment uranium (HA-LEU) as their fissile material and have a fuel cycle very similar to high-temperature gas-cooled reactors (HTGRs). FHRs, however, may require additional short-term cooling shortly following removal from core as their fuel will contain relatively higher loadings for fissile material due to the improved cooling provided by the salt.

MSRs can be advantageous or disadvantageous in terms of fuel material attractiveness. The concept of a denatured MSR (DMSR) was developed in the 1970s [36, 37]. The low material attractiveness DMSR fuel cycle has been incorporated in some of the new MSR designs (Terrestrial Energy and ThorCon). Several MSRs blend a low enrichment U-Pu fuel cycle along with a Th-U fuel cycle to decrease the material attractiveness of the uranium in the fuel salt. Fuel salt additions in MSRs become part of a homogeneous mixture upon being added to operating fuel, so bred fissile isotopes cannot be readily chemically separated from non-fissile isotopes. Isotopes with higher fission cross sections tend to preferentially burn out, lowering the fissile actinide fraction and generating deep-burn fuel over time. Development of a lower fissile fraction plutonium isotopic composition tends to happen faster in thermal spectrum reactors because their fissions are predominantly generated from fissile materials, whereas the fast spectrum systems consume both fissile and fissionable nuclei. The thermal spectrum neutron flux peak in MSRs can also be tailored to match the ^{239}Pu's fission peak near 300 meV to maximize ^{239}Pu consumption.

Several of the prospective vendors (Flibe, Thoreact, Alpha Tech Research, European universities, Shanghai Institute of Applied Physics, and Bhabha Atomic Research Centre) have indicated that they intend to employ a Th – ^{233}U breeding fuel cycle. Breeding ^{233}U from thorium in MSRs, however, continues to involve creating a separated stream of ^{233}Pa that is allowed to decay in a lower thermal flux region or alternatively multiple batches of fissile material which are co-extracted to enable ^{233}Pa to decay outside of a high neutron flux environment. Fast spectrum Th/U reactors can take advantage of the higher neutron yield from fast fissions and return protactinium to the reactor without separated decay. Separated ^{233}Pa which decays to ^{233}U outside of an LEU environment has high material attractiveness.

Few of the new reactor developers provides information on where the fuel salt synthesis is intended to occur. It is not clear whether uranium or thorium metal will be transported to the site and then converted to a fuel salt or fuel salt will be synthesized elsewhere and then transported to the site. Moltex indicates that used water-cooled reactor fuel will be processed into fuel salt in an adjacent, separately licensed facility. The recent and on-going development of non-aqueous, organic fuel salt synthesis methods under mild conditions increase the likelihood of nearby fuel salt synthesis [38]. Several of the reactor vendors explicitly indicate that a desirable fuel feedstock is the TRU from used LWR fuel, but do not indicate where their fuel feedstock will be synthesized. The prospective vendors also do not disclose information on how much or where fresh fuel will be stored on site. Also, no information is currently available on the transport or storage of initial fuel salt loads. This will be especially important for fast spectrum reactors, which will require substantial quantities of fissile materials. Very little plant layout information is available for those reactors that include on-site chemical separations.

Nearly all of the developers indicate that they intend to reuse their fuel salt in future generations of reactors and do not plan to produce fuel salt waste until the end of the reactor class. General information about possible fluoride and chloride salt waste forms is available [39, 40]. Fluoride salts have the particular concern that as the salt cools radiolysis begins to generate gaseous F_2, which can in turn generate gaseous UF_6 [41]. Chloride salts have the challenge of containing ^{36}Cl which is a long-lived ($t_{1/2}$ = 301,000 years) energetic beta emitter which can be mobile in geologic repositories and waste forms.

For MSRs that employ on-site salt processing, Figure 2.1 shows a schematic representation of one option for a fuel salt treatment system, and Figure 2.2 shows the corresponding example processing unit from the MSFR. The on-site salt management of the MSFR combines a salt control unit, an online gaseous extraction system, and an offline lanthanide extraction component by pyrochemistry. The only continuous salt chemistry process is the gaseous extraction system, where helium bubbles are injected in the core to remove all the non-soluble fission products (noble metals and gaseous fission products) via flotation. This gaseous extraction system is composed of a pumping system to circulate the helium gas and a filter which removes the gaseous and the metallic fission products from the salt. This first part of the gaseous extraction system is integrated in the fuel circuit and is thus part of the first barrier. Following this filtration, a part of the gas is withdrawn in order to let the fission products decay, and the remaining part of gas is sent back to the lower part of the core.

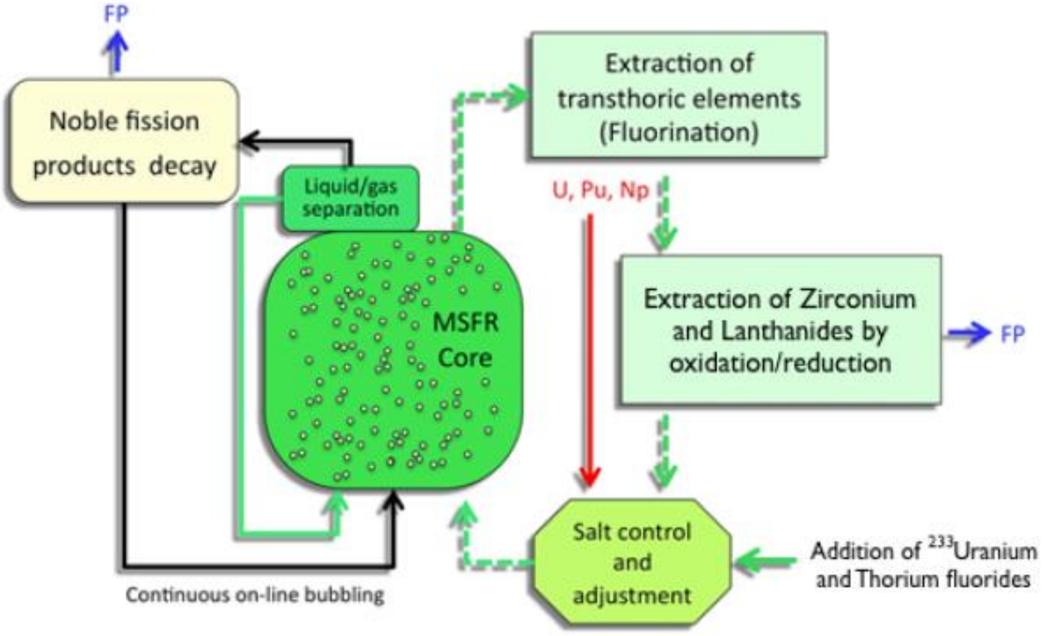

Figure 2.1: Schematic representation of the fuel salt treatment. [20]

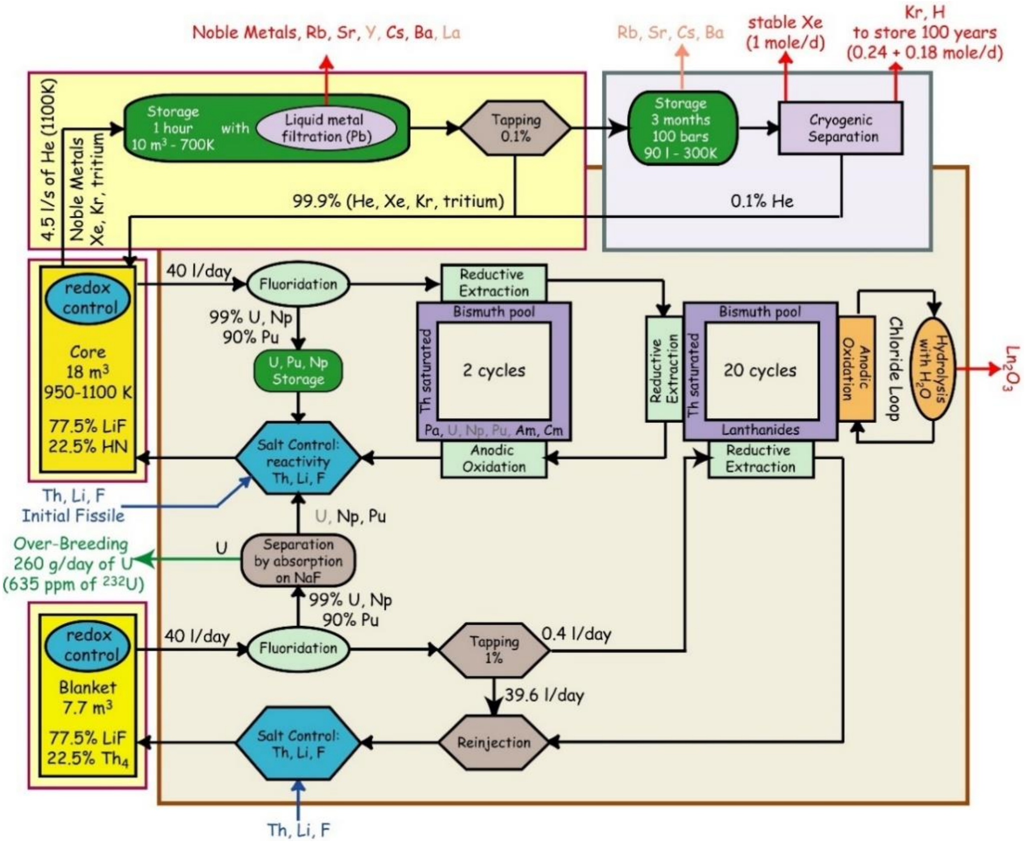

Figure 2.2: Overall scheme of the fuel salt management including the online gaseous extraction (top) and the off-line reprocessing unit (bottom) – The yellow boxes surrounded by a red line are enclosed in the reactor vessel [2011 MSR White Paper]

The salt properties and composition are monitored through the online chemistry control and adjustment unit. A fraction of salt is periodically withdrawn and reprocessed offline in order to extract the lanthanides before it is sent back into the core. In this separate batch reprocessing unit, 99% of uranium (including ^{233}U) and neptunium and 90% of plutonium are extracted by fluorination and directly and immediately reintroduced in the core. The remaining actinides are then extracted together with protactinium and also sent back to the core. In the last step, a second reductive extraction is performed to separate the thorium from the lanthanides which are then sent to waste disposal.

For the reference MSFR the initial fuel salt is composed of either ^{7}LiF-ThF$_4$-(TRU)F$_3$ or ^{7}LiF-ThF$_4$-(^{233}U)F$_3$ with 77.5 mole % of LiF, this fraction being kept constant during reactor operation. For the simulations of the TRU-started MSFR version, the chosen mix of Pu, Np, Am and Cm corresponds to the transuranic elements of an UOX fuel discharged from a standard PWR and after five years of storage. The initial fuel cycle is thus either Th/^{233}U or Th/Pu while the fuel cycle at equilibrium tends to Th/^{233}U.

3. PR&PP Relevant System Elements and Potential Adversary Targets

The relevant system elements for the PR&PP analysis of MSRs varies depending on the design. Figs. 3-1 and 3-2 show two versions depending on whether the design is liquid-fueled or solid-fueled. The liquid-fueled designs have an option depending on whether the reactor includes fissile material separation, i.e. with or without on-site fuel salt processing. The solid-fueled designs are similar to most other reactors that have solid fuel components. Pebble bed designs may have some variations from Figure 3-2, but the same general areas are required. The following components are described in more detail.

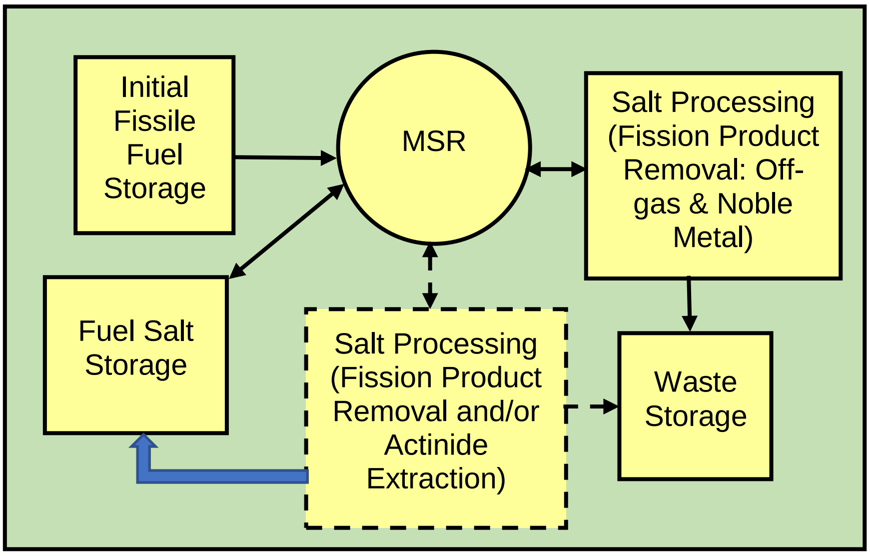

Figure 3.1: Diagram of liquid-fueled MSR nuclear system elements

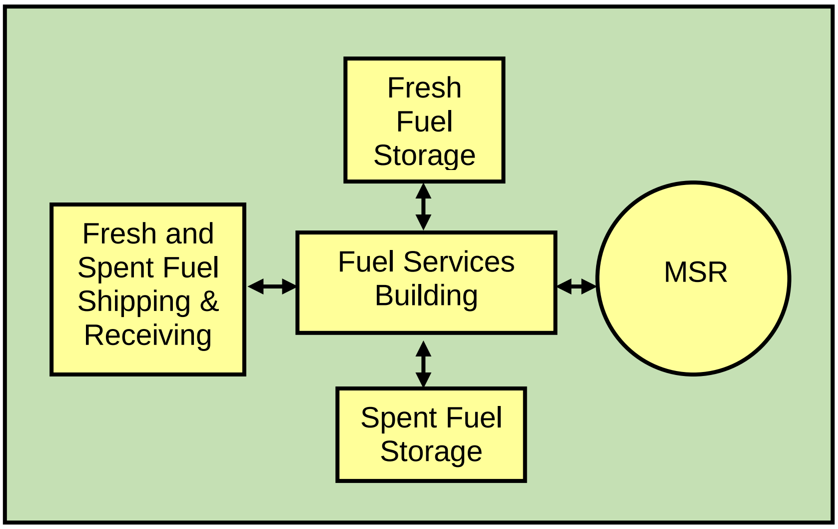

Figure 3.2: Diagram of solid-fueled MSR nuclear system elements

Reactor Building

The reactor core, whether solid- or liquid-fueled and the intermediate circuits and heat exchangers are enclosed in containment within the reactor building. The cover gas system will typically be in a separate containment structure also within the reactor building. The fuel salt storage tanks and possibly the drain tanks will also likely be in separately shielded containments within the reactor building. Solid-fueled versus liquid fueled designs provide differences in sabotage or theft targets, but generally the layers of shielding and other plant components along with the high radiation dose provide barriers to theft and sabotage.

Salt Processing Unit(s)

Salt processing only applies to the liquid-fueled designs. MSRs with liquid fuel may elect to remove fission-gases and noble metals. Fission gases have low solubilities in molten salts, so will tend to inherently evolve from the fuel salt. The noble metals will plate out onto salt wetted surfaces and tend to diffuse into the substrates and may provide some corrosion protection, so are unlikely to need to be separately removed. The neutronic efficiency of fast spectrum MSRs would be only mildly impacted by the presence of fission gases due to the mismatch of their neutron spectra (fast) and the fission gas neutron absorption cross sections (thermal). The neutronic efficiency of thermal spectrum MSRs would be improved by stripping ^{135}Xe and/or its precursor ^{135}I from the fuel salt. Some designs, such as the MSFR, have a separate bubbling or sparging system for off-gas capture and removal of noble metals. An additional salt processing unit may be included for fission product removal and actinide separation (blue arrow in Figure 3.1) for the case of breeder designs. Due to the very high level of radiation in the salt processing steps, all the stages of the reprocessing unit will be automated and performed within a well shielded facility.

Initial Fissile Fuel Storage

The initial load of fissile material to startup an MSR will need to be stored on-site prior to use. Initial fuel load storage may include actinides that are prepared as a molten salt or fresh fuel pebbles in the case of the solid-fueled designs.

Fuel Salt Storage

Fuel (fissile or fertile or mixed depending on burner or breeder) will need to be added to the reactor periodically. This may either be thorium or uranium at various enrichment levels. The safeguards constraints on the storage of thorium are that of fertile materials. Transfer of this material into the reactor site also needs to be considered.

Waste storage

The waste storage unit for liquid-fueled reactors is designed to manage the activated structural materials with retained fissile materials (heel (residual) or impacted atoms), salt impregnated graphite, radioactive fission products coming from the gaseous extraction (mainly gases and noble metals) and from the reprocessing unit (mainly lanthanides) if separation of fission products is done on-site. They will be stored to reach, after radioactive decay, an acceptable radioactivity level and thus a reasonable decay heat. Solid-fueled designs will have spent fuel storage instead.

3.1. The MSFR Study

A PR case study for the MSFR [42] was conducted during the Euratom SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) project of the Horizon 2020 program. By applying some elements of the GIF PRPP evaluation methodology, the study identified the system elements of the nuclear power site, targets for material diversion and the pathways to achieve a concealed diversion of material by a host state having unlimited means.

The nuclear power site is assumed to contain several MSFRs, sharing common facilities, such as the fuel processing facility, waste handling facility and fuel storage. Figure 3.3 is a schematic representation of a nuclear site with 4 reactors sharing common facilities. In the figure the green rectangles with red outlines represent safeguards monitoring stations for transfers in and out of the elements. The ^{232}U co-produced with ^{233}U in the MSFR has ^{208}Tl daughter products that emit highly energetic (2.6 MeV) gamma rays with high absolute emission probability. Consequently, any material transfer between the reactor and processing, handling and storage facilities has to be performed using remote handling, indicated in yellow in Figure 3.3. One has to note that this scheme is not final: the question of which elements are shared between reactors and which are dedicated to a single reactor has not been decided. It is likely that a more complex structure will be necessary, in particular for the fuel cleaning unit, depending on the proliferation resistance analysis results. The schematic will be modified as the design progresses.

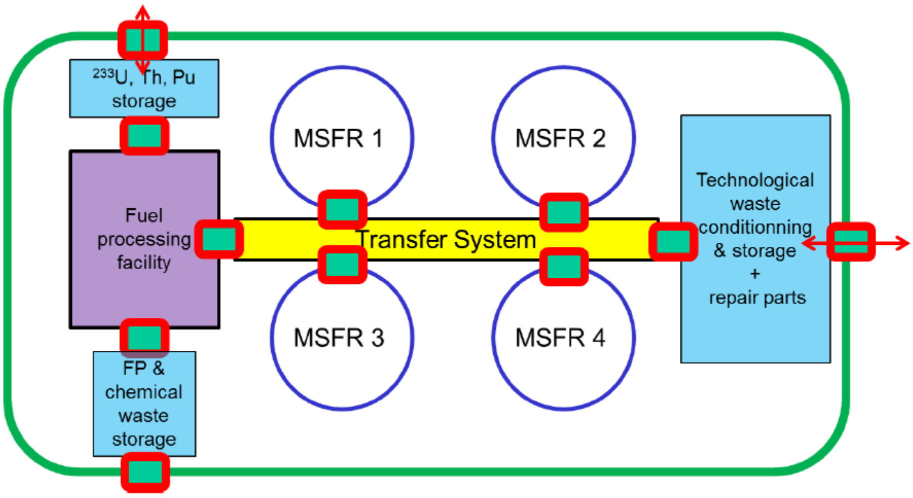

Figure 3.3: Schematic representation of the MSFR nuclear site system elements. [42]

One of the attributes that characterizes the attractiveness of targets for diversion and theft is the isotopic composition of the fuel salt. Table 3.1 [42] lists the most attractive isotopes for the MSFR, considering two startup conditions after one year (^{233}U started reactor and ^{enr}U+TRU started reactor) and the equilibrium composition of the fuel salt (200 years after starting) and the fertile salt in the blanket. Table 3.1 is a subset and can be compared with Table 1 from a more recent study in Ref. [59]. A previous study [43], considering two main limitations to the power density in the MSFR (materials damage and heat exchangers capability), have shown that a fissile inventory between 2.5 and 4 metric tons per GWe may be reached.

Table 3.1: Main Isotopic Inventories for a 3000 MWt MSFR (in kg unless noted) [42]

Isotope	^{233}U -started after 1 year	^{enr}U+TRU - started after 1 year	Fuel salt steady state 200 years	Fertile salt
^{232}U	3.5	142 g	13	34 g
^{233}U	4976	514	4658	58.5
^{234}U	143.9	12.8	1769	0
^{235}U	4.9	2506	510	0
^{236}U	0	149.5	562	0
^{238}U	0	16300	1	0
^{232}U/U	700 ppm	50 ppm	1700 ppm	600 ppm
^{233}U/U	97%	2.7%	62%	99%
^{238}Pu	0	239	161	0
^{239}Pu	0	3265	66	0
^{240}Pu	0	1617	57	0
^{241}Pu	0	641	48	0
^{242}Pu	0	491	10	0
^{239}Pu/Pu	0	52 %	19 %	0

Isotope	^{233}U -started after 1 year	^{enr}U+TRU - started after 1 year	Fuel salt steady state 200 years	Fertile salt
^{232}Pa	3.9 g	0	15 g	15.4 g
^{233}Pa	124	45.6	108	13

The two fuel materials that are likely targets are isotopes of uranium and plutonium. The co-production of ^{232}U in a Th/U based fuel cycle potentially complicates use of ^{233}U for nuclear weapons material because of high gamma radiation. However, there is potential to divert the fuel salt (also the fertile salt) out of the neutron irradiation environment to a chemical processing facility, wait for most of the ^{232}Pa to have decayed to ^{232}U, chemically separate out the remaining Pa, and then wait for the ^{233}Pa decay to ^{233}U. As for Pu, the ^{238}Pu provides some intrinsic radiation barrier. An initial Pu mixture containing ^{238}Pu[4] increases the technical difficulty for diversion (but does not decrease the IAEA material category), which is the case when using the MA mix produced in LWRs.

3.2. The Mk1 PB-FHR Design

The solid-fueled designs are described separately since the fuel is in such a different form as compared to the liquid fueled designs. The fuel and core design for the Mk1 PB-FHR design is shown in Table 3.2, reproduced from reference [31]. The fuel pebbles would be the only theft target on site and would be contained either in the core, within the pebble handling systems, or in storage.

The core will generate 920 spent fuel pebbles per day during steady-state operation. The report suggests that the spent fuel pebbles will be stored in canisters that can hold approximately 29,440 pebbles. The canisters have a height of 1.75 m and diameter of 0.71 m, so they are very large storage canisters that would be difficult to move. Approximately 250,000 pebbles would need to be removed to acquire one significant quantity of uranium (for low-enriched uranium (enrichment <20%) it is 75 kg of U)[5], so multiple canisters would be required. Another technical difficulty (discussed further in Section 4.3) is that mature technology to physically and chemically separate Pu from TRISO fuel particles does not exist.

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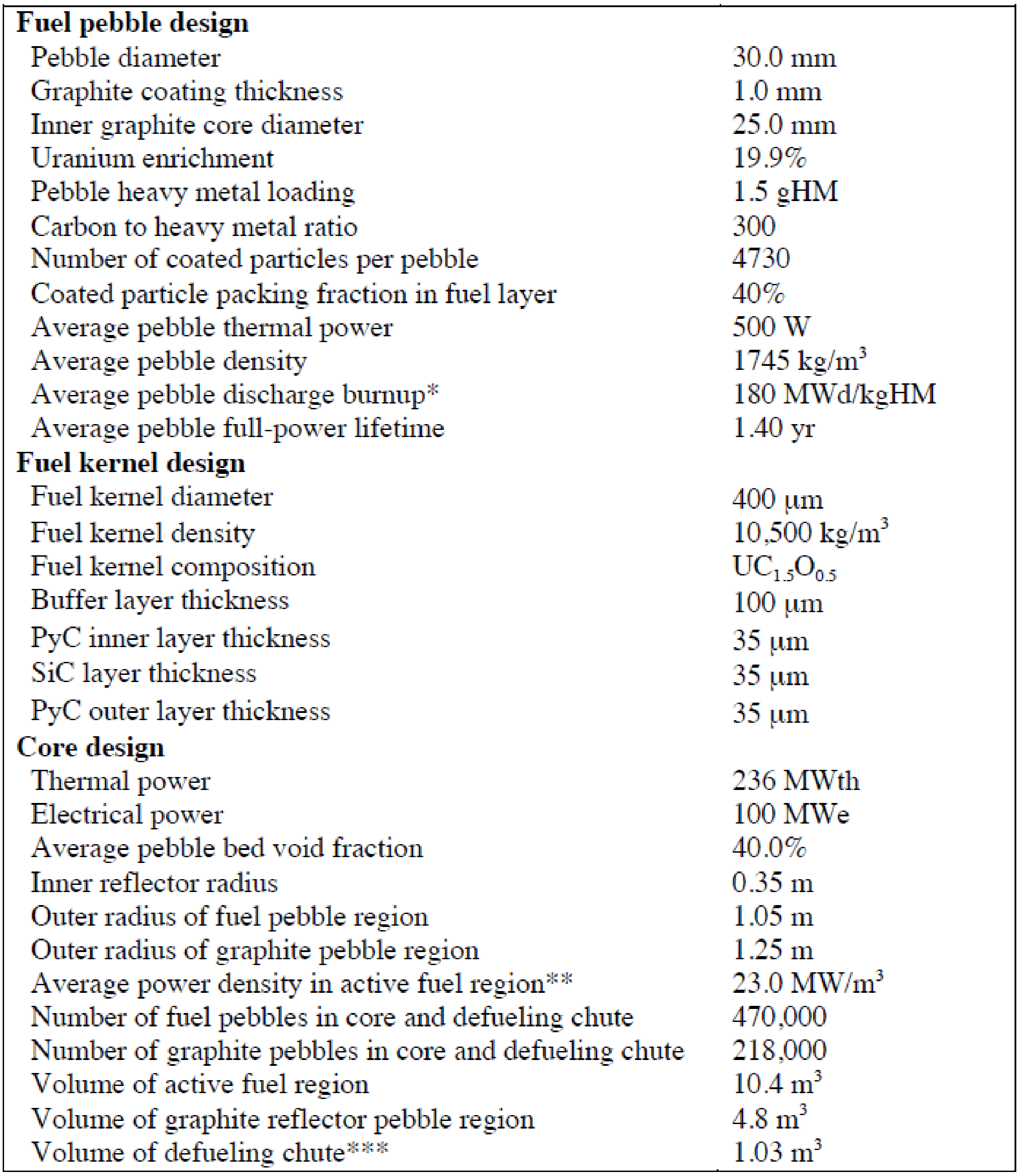
[4] For nuclear safeguards verification activities there is no distinction for Pu with less than 80% Pu-238. However, the heat generated by Pu isotopic containing more than a few percent of Pu-238 would substantially increase the technical difficulty related to the fabrication phase (weaponization).Using a set of figures of merit (FOM) for attractiveness, Bathke, et al. [“The Attractiveness of Materials in Advanced Nuclear Fuel Cycles for Various Proliferation and Theft Scenarios,” Nuclear Technology, Vol 179, Issue 1, 2012] estimated that about 8% ^{238}Pu is required to render the plutonium isotopic unattractive for an unadvanced proliferant state that requires reliably high-yield nuclear devices, however it remains attractive for both technologically advanced states, which can handle it, and subnational groups for which high reliability might not be a requirement. An evaluation by Kessler, et al. [“A new scientific solution for preventing the misuse of reactor-grade plutonium as nuclear explosive,” Nucl. Eng. Des. 238, 3429–3444, 2008] observed that plutonium with up to 9% ^{238}Pu is weapons usable only if high technology is used. Since the technical challenges go up with the increase of the Pu-238 abundance in the Pu isotopics the technical capability of the proliferant state is an important attribute in the overall evaluation of the PR against Pu diversion.

[5] The IAEA Safeguards Glossary defines Significant Quantity (SQ) as “the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded”. For Pu (containing less than 80% ^{238}Pu) and for ^{233}U a SQ corresponds to 8 kg. A SQ is 25 kg for U enriched in ^{235}U at 20%, or above, 75 kg for U enriched below 20% in ^{235}U (or 10 t for natural U or 20 t for depleted U) and 20 t for Th. See the IAEA Safeguards Glossary for more details.

Table 3.2: Mk1 PB-FHR fuel and core design [31]



Fuel pebble design	
Pebble diameter	30.0 mm
Graphite coating thickness	1.0 mm
Inner graphite core diameter	25.0 mm
Uranium enrichment	19.9%
Pebble heavy metal loading	1.5 gHM
Carbon to heavy metal ratio	300
Number of coated particles per pebble	4730
Coated particle packing fraction in fuel layer	40%
Average pebble thermal power	500 W
Average pebble density	1745 kg/m^3
Average pebble discharge burnup*	180 MWd/kgHM
Average pebble full-power lifetime	1.40 yr
Fuel kernel design	
Fuel kernel diameter	400 μm
Fuel kernel density	10,500 kg/m^3
Fuel kernel composition	$UC_{1.5}O_{0.5}$
Buffer layer thickness	100 μm
PyC inner layer thickness	35 μm
SiC layer thickness	35 μm
PyC outer layer thickness	35 μm
Core design	
Thermal power	236 MWth
Electrical power	100 MWe
Average pebble bed void fraction	40.0%
Inner reflector radius	0.35 m
Outer radius of fuel pebble region	1.05 m
Outer radius of graphite pebble region	1.25 m
Average power density in active fuel region**	23.0 MW/m^3
Number of fuel pebbles in core and defueling chute	470,000
Number of graphite pebbles in core and defueling chute	218,000
Volume of active fuel region	10.4 m^3
Volume of graphite reflector pebble region	4.8 m^3
Volume of defueling chute***	1.03 m^3

4. Proliferation Resistance Features

The proliferation resistance features of MSRs will be described based on the three general reactor classes, and in some cases these features are very distinctive for advanced reactors.

4.1. Liquid-Fueled with Integrated Salt Processing

Liquid-fueled MSRs do not contain their fuel in assemblies. It is then not possible to perform traditional item counting and visual accountability of the salt fuel. The facility is closer to a bulk accounting facility like a reprocessing plant, and inventories would need to be determined based on measurements of the actinide content in the salt. However, unlike bulk facilities, MSRs do not have high material throughput and both create and consume fissile materials presenting a challenge to assessing what a correct inventory would contain. Further, the nuclear cross sections and reactor physics models have sufficient remaining uncertainty to prevent accurate evaluation of the intended fissile inventory. Moreover, sampling and destructive analysis of actinide-laden molten salts remains technically challenging. Some of the challenges stem from the unique combination of high temperature and high radiation environments present in the salt fuel. Another challenge is the continuous variation of isotopic concentrations in the fuel salt from burnup, transmutation, plating out, and online chemical processing. With on-site salt processing there is a potential to have fuel inventory present outside the reactor containment vessel.

4.1.1. Concealed Diversion or Production of Material

The following sub-sections discuss several potential proliferation pathways for concealed diversion or production of nuclear materials associated with the MSFR. Table 4.1, using data from Table 3.1, lists some of the potential target materials and their quantity in the fuel salt volume. Several countermeasures against proliferation of nuclear materials from the MSFR have been suggested in a recent study [59]. The countermeasure discussion is reproduced in Appendix 3 to this paper as an example of potential opportunities to implement safeguards by design features to enhance the proliferation resistance of a particular reactor design.

Table 4.1: Inventory of potential nuclear materials in a 3GW MSFR expressed in terms of Significant Quantities (SQ) and in liters of salt to get the corresponding material for one SQ.

Target		**^{233}U-started after 1 year**		**^{enr}U+TRU-started after 1 year**		**Fuel salt steady state at 200 years**		**Fertile salt after 1 year**	
		Total SQ	l/SQ	Total SQ	l/SQ	Total SQ	l/SQ	Total SQ	l/SQ
Pure ^{233}U		622	29	64	280	582	31	7	1100
Plutonium		0		780	23	43	420	0	
U	^{233}U			64	280	580	31	-	-
	^{235}U			33	550	20	900		
^{233}Pa converted to ^{233}U		15	1160	6	3000	13	1380	1.6	4800

4.1.1.1 Diversion of ^{233}U

The radiological dose of the reactor and salt processing lines may provide an advantage for proliferation resistance because it could be difficult to handle or acquire the molten salts. As an example, the MSFR can be started with ^{233}U or TRU. Figure 4.1 shows the actinide inventory in the reactor as a function of time. The first case is *a priori* more sensitive because ^{233}U, due to rather small critical mass (around 16 kg for pure ^{233}U and 26 kg for the Uranium mix present in the salt), very low spontaneous fission rate, and long half-life (1.6 10^5 years), might be used for nuclear weapons.

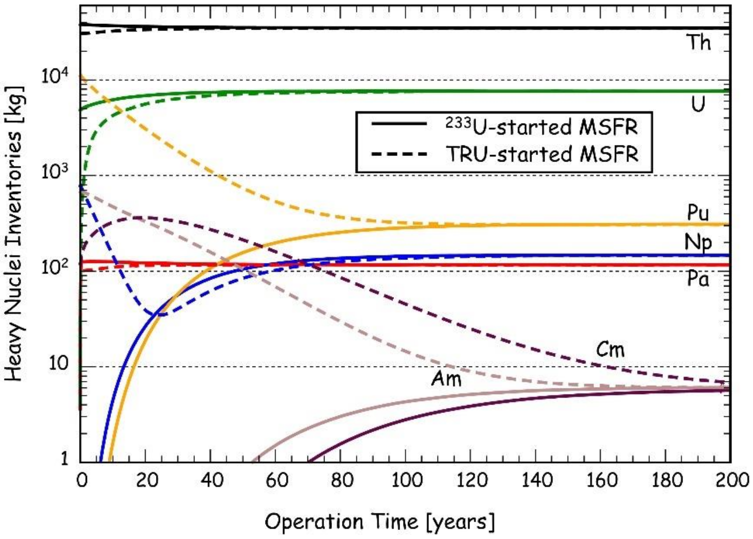


Figure 4.1: Heavy element inventory for the ^{233}U-started MSFR (solid lines) and for the transuranic-started MSFR (dashed lines) [60]

The Uranium fuel in the MSFR is diluted in the salt and represents a small fraction (2 to 3 mol %) of the salt[6] (see Figure 4.1 - solid lines).

Table 4.1 gives an evaluation of the salt volume to be processed to get one significant quantity of the various nuclear materials.

A diversion of this amount of fuel will be detectable by a couple of methods:

- *Fuel salt composition monitoring:* The very short recirculation time of the fuel salt implies that the majority of the fuel, except any hold up in processing equipment or other locations, has the same concentration of fissile materials, fertile materials, and fission products. A large amount of information can be gained by monitoring the elemental and isotopic composition of the salt either through sampling and destructive analysis or non-destructive analysis techniques. Concentrations can be determined either through direct measurements of actinides or indirect measurements of fission products. Further study of fuel composition monitoring methods will need to be developed.
- *Reactor operation temperature monitoring:* The reactor reactivity, and thus the fissile inventory of the core, may be controlled by stabilizing the operation temperature of the reactor. This is due to the largely negative feedback coefficients of the MSFR concept: a decrease of operating temperature at constant power would reveal a decrease of reactivity due to a leak of fissile matter. Studies of the MSFR [44, 45, 46] showed that a disappearance of 1 kg of ^{233}U leads to a reactivity variation of 9.5 pcm. Assuming a feedback coefficient value of about -5pcm/°C, a loss of 1 kg of ^{233}U would thus lead to a decrease of 2°C of the operation temperature. The diversion of one ^{233}U SQ would then lead to a decrease of about 16°C of the operating temperature, which is easy to measure. Reactor temperature monitoring will only be able to detect larger, abrupt losses of fuel salt and not smaller, protracted losses since the uncertainty in the power output would be greater than any small losses of fissionable material. For smaller

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[6] Polyvalent fluorides have limited actinide tri-fluoride solubility. Chloride salts and monovalent fluorides can contain substantially more fissile material.

protracted losses the decrease in reactor temperature may not be obvious immediately and would only be observed incrementally.

Past work on pyroprocessing safeguards has investigated measurement technologies for determining actinide content in molten salts—these technologies may be applicable for MSRs, except the radiological dose will be much higher since any samples may come right out of the reactor. The use of decay tanks in some designs for salt processing may provide time for short-lived fission products to decay, so sampling after these tanks would be more optimal.

One significant difficulty of measurements of molten salts will be measuring the total bulk salt mass for the entire system. An MSR will have a unique geometry for the core, heat exchangers, pipes, and salt processing systems. It will be very challenging to determine total salt mass with precision.

Another difficulty is that molten salts can potentially present a highly corrosive environment, which makes the maintenance of safeguards monitoring instrumentation challenging. On-line instruments in direct contact with salt may not last long.

Plate-out of rare earth elements and noble metals in reactor components and piping can also present measurement challenges. The reactors will be designed to remove these materials as needed, but it is still an engineering problem that must be addressed.

For MSRs that employ fuel salt processing on-site, the potential extraction of fission products and/or bred actinides can present more attractive targets of proliferation concern. Designs that minimize or eliminate separation of fissionable species may be more proliferation resistant. The uranium isotope compositions of the salts are given in Table 3.1 for the MSFR concept both for the fuel salt and for the fertile fuel if a fertile blanket is present. As noted previously, the presence of those isotopes increases the critical mass (26kg instead of 16 kg for pure ^{233}U); that is equivalent to an isotope dilution (or denature).

Concerning proliferation resistance, the most interesting product is Uranium 232, which is primarily produced by an energetic neutron (n, 2n) reaction (reaction Q value -6.43 MeV) on Thorium-232 according to:

$$^{232}\text{Th+n} \underset{(n,2n)}{\longrightarrow} {}^{231}\text{Th} \underset{\beta^-}{\longrightarrow} {}^{231}\text{Pa}$$

$$^{231}\text{Pa+n} \underset{(n,\gamma)}{\longrightarrow} {}^{232}\text{Pa} \underset{\beta^-}{\longrightarrow} {}^{232}\text{U}$$

For thorium derived from mixed thorium uranium ores, the thorium will include thorium-230, which has a significant neutron absorption cross section to product thorium-231. However, thorium-230 has a short half-life (75,380 years) compared to geological processes and only exists in nature as a member of the uranium-238 decay chain. Consequently, thorium derived from ore bodies without collocated uranium will not include significant amounts of thorium-230. In the case of the MSFR concept, all isotopes of Pa are quickly sent back to the core (see Figure 2.2). For fuel cycles in which the protactinium isotopes are removed on a 10-day cycle and not reintroduced into the fuel (as in the MSBR design), production of uranium-232 is significantly suppressed due to the removal of intermediate isotopes. The ratio of ^{232}U over U in the fuel salt and in the fertile salt of the MSFR is displayed on Figure 4.2. For a ^{233}U started MSFR, the ratio ^{232}U/U varies from 30 ppm (in the fertile salt) after one year of operation to around 3000 ppm for the fuel salt at equilibrium.

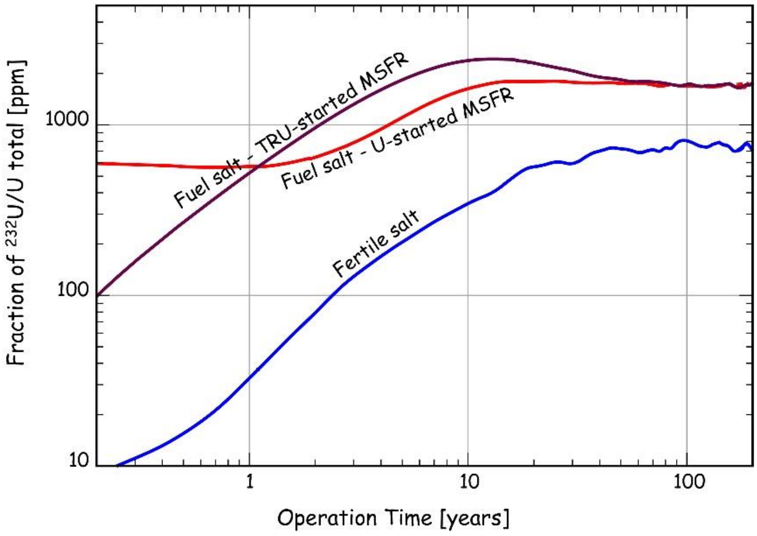

Figure 4.2: Evolution of the 232U/U ratio in the core (fuel salt) and in the fertile blanket during reactor operation for both U-started MSFR and TRU-started MSFR [derived from [59]]

The decay scheme of ^{232}U (half-life 68.9 years) is given in Figure 4.3. The main feature related to the proliferation resistance is the presence of a considerable fraction (36%) of ^{232}U decay products with a very energetic (2.6 MeV) γ ray, preventing easy handling of the salt and extracted uranium. This may also help to detect the diversion of uranium even in very small quantities. The slowest step in the decay chain is the ^{228}Th decay (1.91 year). The activity of 1g ^{232}U, related only to that γ ray and assuming equilibrium among decay products, increases by 0.3 GBq per day during the first three months of the reactor operation, with the maximum activity reached after 10 years equal to 270 GBq (from the ^{208}Tl activity). This value, combined with the γ energy, explains why the handling and transport of diverted uranium is difficult without detection and presenting a serious health hazard. This would also generate a distinct signature for the uranium contained in the fuel salt and in the fertile salt.

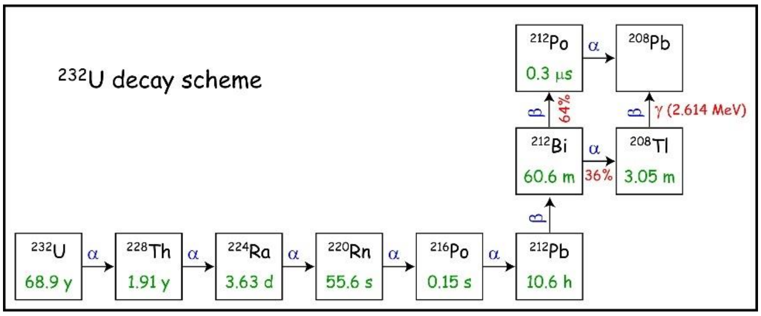

Figure 4.3: Decay scheme of ^{232}U [59]

4.1.1.2 Diversion of Pu

From the proliferation viewpoint, plutonium production has to be accounted for. In the case of ^{233}U-started MSFR, Pu is produced in very limited quantity (Figure 4.1 solid lines). Moreover, the most abundant isotope is ^{238}Pu, which represents more than 50% of the Pu (see Figure 4.4) and is characterized by an extremely high spontaneous fission rate[7]. MSFRs operated on a thorium fuel cycle cannot be used to make plutonium readily usable for nuclear weapons.

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[7] 1 kg of ^{238}Pu emitting 0.12 GBq of spontaneous fissions

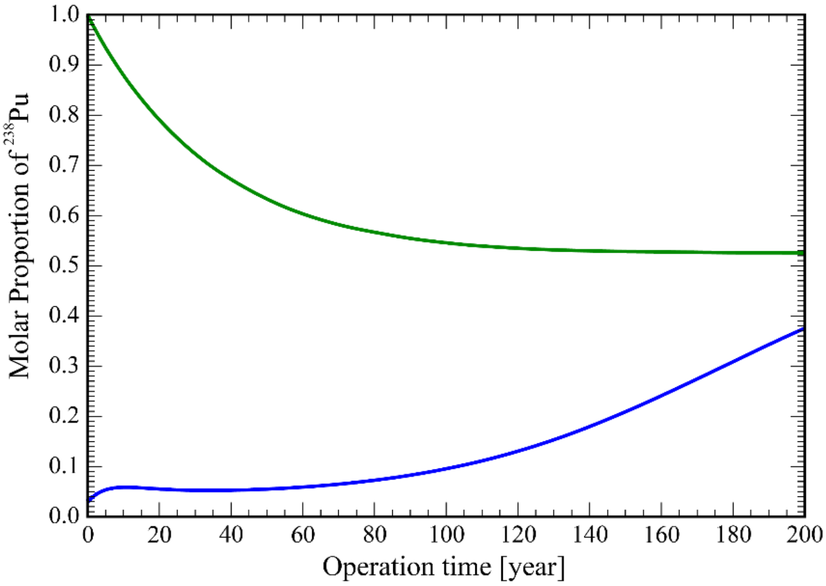

Figure 4.4: Time evolution of the ^{238}Pu content in the total Pu for a reactor started with ^{233}U (green curve) and with enriched@13%U+TRU (blue curve) [59]

The issue is different for the case of MSFR started with the Pu and minor actinides produced in LWRs instead of ^{233}U, because the amount of Pu is initially larger, as shown in Figure 4.1 (dashed lines). To avoid proliferation problems, this initial Pu has to contain enough ^{238}Pu (more than 3 to 5%), which is the case when using the MA mix produced in LWRs. As shown in Figure 4.4, the ^{238}Pu/Pu proportion then increases when the fissile isotopes are burnt during reactor operation, reaching more than 50% in the long term.

The decision to use a fertile blanket should also be based on proliferation risk considerations in addition to operational parameters. MSRs can be designed without a separate fertile blanket, which should be considered. In the case of the MSFR, even if fertile blankets are used, the production of ^{232}U is large enough to make difficult the utilization of blankets for proliferation purpose (see Table 3.1 and Figure 4.2).

4.1.1.3 Diversion of Pa

The 2.6 MeV gamma radiation can be suppressed in two ways. One is to isolate the Pa from all the other actinides, then wait for the decay of the ^{232}Pa so as to divert ^{233}Pa after having extracted from it the U and its descendants, in one or several passages within the fuel salt cleaning unit (see Figure 4.5). The other is to efficiently separate the Th and its descendants from the U to cut the decay chain at the ^{228}Th level. The second option suspends the 2.6 MeV gamma radiation while the first attenuates it indefinitely. The procedures used to clean the fuel or extract the U from the blanket have to be evaluated in this perspective [59].

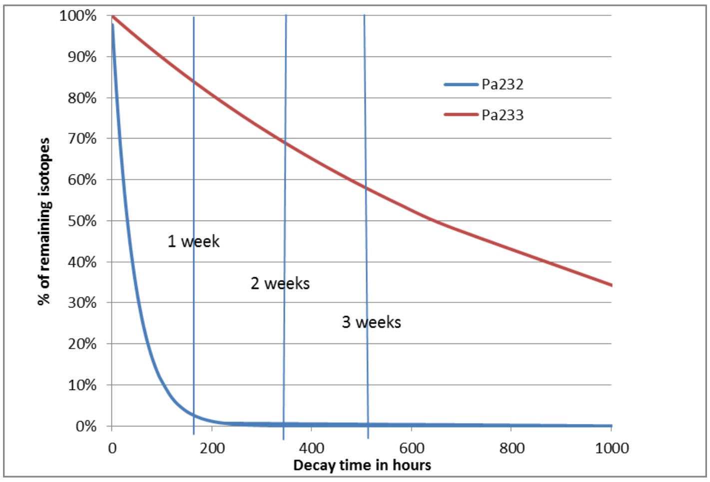

Figure 4.5: Remaining Pa isotope fractions after Pa isolation. After 3 weeks of decay, the remaining fractions are 58% ^{233}Pa and 13 ppm ^{232}Pa (respectively 70% and 560 ppm after 2 weeks) [59]

According to the isotopic inventory given in Table 3.1 for the reference MSFR operating in equilibrium cycle, the 18 m^3 of fuel salt contains more than 100 kg of ^{233}Pa, and the blanket contains 13 kg of this isotope in 7.7 m^3 of fertile salt. A pathway to Pa diversion that leads to clandestine production of ^{233}U has been postulated [42].

The idea is to isolate Pa from the fuel/fertile salt by chemical processing and then wait for most of the ^{232}Pa decayed into ^{232}U before separating out the remaining ^{233}Pa from other isotopes. It is noted that after 3 weeks of storage the fraction of ^{233}Pa remaining is 58% but the amount of ^{232}Pa has been reduced by a factor 4x10^4. After 4 weeks this factor becomes 1.5x10^6 and the remaining ^{233}Pa is still 49% of the initial amount. In such condition getting one Significant Quantity (SQ) of ^{233}U (8 kg) would require the processing of about 10 m^3 of fertile salt or 2.7 m^3 of fuel salt. Moreover, it will not be possible to avoid some Th to be carried along with the Pa coming from the fuel salt, preventing a total cut of the decay chain resulting in the 2.6MeV radiation. Although theoretically possible this would not remain un-noticed. Proliferation resistance could be increased if it were possible to design the facility to complicate or preclude the installation of equipment to separate ^{233}Pa. Safeguards will play an essential role.

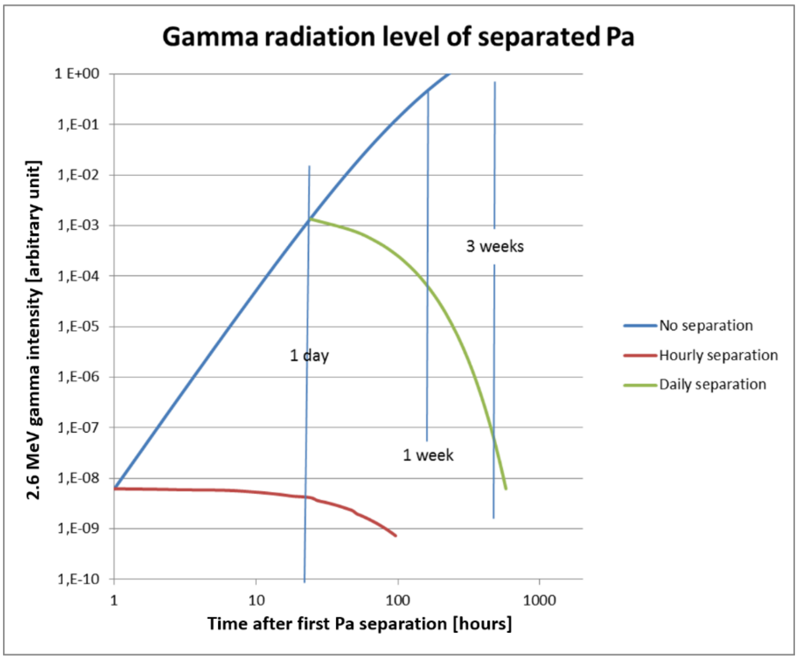

Figure 4.6: Influence on the radiation level of a periodic extraction of the U and its descendants. An hourly extraction seems to be the most frequent feasible. A daily extraction is easier to implement but the radiation level of the diverted materials

Figure 4.6 illustrates the reduction of the radiation emitted by the stored Pa that is obtained with a periodical extraction of the U. Such an extraction limits the radiation level so that the storage of Pa in the cleaning unit may be undetected. The recycling of ^{232}U in the fuel salt weakens the effect of the concealed storage on the fuel's gamma radiation emission. If the Pa remains in the cleaning unit for 3 weeks, the emission due to the Pa that has not been transformed into U becomes very small, making its diversion from the nuclear site much easier [59].

The MSFR concept is designed to operate without the need of producing extra uranium out of the system. The role of the reprocessing unit (Figure 2.2) is to extract lanthanides. Before lanthanide removal, the first steps consist of extracting uranium, minor actinides and Pa, and to send them back directly into the core. By diverting some of the fuel salt to a clandestine Pa processing facility some replacement fuel salt will be reintroduced into the reactor. This would significantly modify the isotopic composition of the fuel salt, such as reduction in the proportion of MA in the fuel salt (1/3 of Cm will disappear for example), which is easily detected through a check of the fuel salt composition.

4.1.1.4 Diversion from Fuel Salt Processing

There is a potential opportunity to divert uranium during the fuel salt processing step (see Figure 2.1), after extraction of the uranium by fluorination (UF_6) and before incorporation back to the fuel salt (UF_4). The diversion will upset the actinide balance of the fuel salt and should be readily detectable.

4.1.1.5 Concealed Production of ^{233}U

There is a potential for misuse of the MSR by modifying its fuel salt composition to produce more ^{233}U. This covert pathway will require a very efficient organization (significant and permanent modifications of the reprocessing scheme of the MSFR) which will be impossible for individuals and difficult to be done undetected for a state. Furthermore, this recurrent operation is easily detected through a check of the fuel salt composition.

Another scenario of concealed production of ^{233}U involves the diversion of Pa as discussed in Section 4.1.1.3

4.1.2. Breakout

In a breakout scenario institutional (extrinsic) barriers, such as safeguards are ineffective and only intrinsic barriers are still in force. Strategy under breakout usually has the objective of minimizing proliferation time while striving for the production of optimal isotopic composition for weapons grade material.

For the MSFR, the most attractive target would be the production of ^{233}U as discussed in Section 4.1.1.1 but with attempts to minimize ^{232}U contamination. In all cases, the product would contain some ^{232}U contamination. A related strategy is to separate ^{233}Pa overtly.

Alternatively, the MSFR could modify its fuel salt composition (e.g. adding ^{238}U) and operating conditions to favor production of Pu with minimal ^{238}Pu, as discussed in Section 4.1.1.2.

4.1.3. Production in Clandestine Facilities

A significant deterrent to using a clandestine MSFR to produce fissile materials is the additional precaution to shield operations against the energetic radiation from the decay of ^{232}U. Environmental emissions from the operation of the salt reprocessing systems would potentially provide signatures for safeguards monitoring.

4.2. Liquid-Fueled Without Integrated Salt Processing

For the liquid-fueled MSRs without fissile materials separations, many of the observations from the previous section apply, except salt processing is minimized. The reactors will still need some method of estimating total actinide content. These reactor designs reduce proliferation risk for the reactor by not separating any actinides during operation. However, that is balanced by the need for a centralized salt processing facility located elsewhere in the fuel cycle to handle the salt or cores every 4-8 years (e.g. this is an option for the ThorCon reactor). In effect, these reactor designs push off some of the accountancy challenges to a bulk reprocessing facility (much as in LWRs operating on a closed fuel cycle), which will have other challenges.

Many of the new designs employ limited lifetime salt wetted components and may replace major components multiple times over the course of a plant lifespan. Consequently, the plant designs need to accommodate systems to replace major components and to transfer fuel salt from the old components to their replacements. A particular issue for liquid fuel MSRs is the heel of fuel salt left on components that have been drained. Liquid fuel designs will likely employ a flush salt to reduce the amount of fuel left in replaced components. However, the flush salt itself will progressively acquire additional radionuclides and will need to be stored in containment when not in use. MSR plants will need an effective capability to transfer worn out components into secure, cooled storage and to perform fissile material inventory as components are transferred out of containment.

By way of further discussion using the IMSR from Terrestrial Energy as an example, the IMSR core is a vessel, the primary container of the radioactive fuel salt, into which the number of penetrations has been strictly minimized. Nevertheless, various penetrations into the core are unavoidable. The most significant penetrations are of course the fresh fuel supply lines and the spent fuel defueling lines. The fresh fuel supply lines are used for fuel top-up periodically over the seven-year life of the core. As a general principle, a core such as that of the IMSR, with low access opportunity and minimal inventory changes, presents a high level of proliferation resistance. The reference IMSR core, with no access and its single defueling/refueling operation, would appear to have a high degree of safeguardability. Yet, the inability to access the core for the purposes of safeguards verification purposes is simultaneously a significant safeguards concern: how can continuity of knowledge of core contents be maintained or assured during the seven-year period of core operation? Some upcoming techniques for novel safeguards approaches, such as with stand-off reactor monitoring using neutron detection [47], alpha spectroscopy [48], spectroelectrochemistry [49], voltammetry [50], and laser-induced breakdown spectroscopy [51] may provide solutions to this issue of maintaining continuity of knowledge with direct physical sampling of the core.

One potential diversion scenario for conventional reactors involves the addition of small amounts of fertile material in an inconspicuous location that is transmuted to weapons-grade material in the core and extracted with short in-core transit times. As previously mentioned, the system for regular fuel top-ups of the IMSR will not permit operators to subtract fuel from the core, precluding breeding and diversion in the main reactor vessel. Due to the high temperature and radiation levels of the fuel salt any significant transfer, such as the final spent fuel transfer to holding tanks, will only be done under established safeguards protocols and supervision. Remote monitoring may be used to safeguard minor transfers of fuel salt for chemistry monitoring and control.

Fresh fuel is delivered to the IMSR plant in sealed containers. The fresh startup fuel consists of fuel salt (solid at room temperatures) containing 2% enriched uranium. This material will require safeguards, but it is presumed that these will be applied upstream at the shipping site for the fuel. IAEA verification at the IMSR reception and storage area for fresh fuel might consist of solely checking serial numbers and the integrity of the seals.

Each IMSR core unit begins with 2% enriched fresh uranium fuel at the beginning of life, and 4.95% enriched uranium fuel is used to top up the core fuel at intervals during the nominal 7-year life of a core. The fuel from storage is melted with electric heaters and forced through supply lines into the core unit as a molten salt. This will undoubtedly represent a key flow-measurement point of the system for safeguards, as it provides the independent determination of core inventory. Any 'tee' extraction points along this route (for example, maintenance access), will require monitoring or sealing. In addition to mass flow, verification of contents will be required – potentially involving a similar technique used upstream to verify/reverify the contents of fresh fuel storage. At this stage the fuel salt is in liquid form, presenting an opportunity to take a physical sample for analysis. The safeguards approach for the fresh fuel transfer will also need to provide assurance of the one-way direction of fuel flow into the core.

In the reference IMSR design the spent fuel is removed from the core unit following a cool-down period at the end of the 7-year operational cycle. At this point a mass balance can be applied to verify the lack of diversion of fuel during the cycle. The spent fuel is transferred through flow lines to storage containers located a few meters away from core, within the same containment boundary. The spent fuel salt will contain a mixture of uranium, plutonium and other minor actinides, in addition to fission products. The safeguards applied to the spent fuel transfer and storage will generally feature the same challenges and solutions as that discussed previously, for the fresh fuel transfer to the core, with the added complexity of (a) higher ambient radiation fields, and (b) the presence of plutonium and other minor actinides. These two differences will affect the choice of technology used to verify inventory and flow, possibly including an automated process for taking samples and sending samples to a joint on-site laboratory.

The storage containers for spent fuel will require verification of inventory, followed by containment and surveillance (C&S), to maintain continuity of knowledge. For inventory verification, some method of weighing the containers is likely to be useful so that the weight of the fuel can be matched against the total fuel input. This method can be complemented by radiation profiling to match expected contents against predicted ones.

4.3. Solid-Fueled with Molten Salt Coolant

The solid-fueled MSR designs will have proliferation resistance features similar to any light water reactor or high temperature gas reactor with fixed fuel assemblies. The robust mechanical structure of the fuel, along with the difficulties of reprocessing TRISO fuel, provide a non-proliferation advantage for the solid-fueled designs as compared to liquid-fueled. This also places solid-fueled MSRs as item-accountancy facilities as opposed to quasi-bulk handling facilities for liquid-fueled designs (some analysts refer to pebble bed reactors as semi-bulk facilities). The fact that salt processing is not occurring on site also provides a non-proliferation advantage.

4.3.1. Diversion of Nuclear Material

Pebble bed reactors require the diversion of thousands of highly radioactive pebbles in order to accumulate one significant quantity, so diversion is unrealistic to carry out. Also, fuel pebbles are stored and transferred in canisters that have a height of 1.75 m and diameter of 0.71 m, so they are bulky and would be difficult to move. Concealed diversion is very difficult given that over 250,000 fresh fuel pebbles would need to be stolen to acquire enough U-235 for a significant quantity. The amount of spent fuel needed to acquire enough Pu-239 will vary depending on burnup, but will also likely require a similar order of magnitude in terms of the number of pebbles that would need to be taken. The sheer volume would make concealment, such as masquerading spent fuel pebbles as graphite blanket pebbles, all but impossible, and the radioactivity of spent pebbles makes diversion very difficult to carry out in practice.

4.3.2. Clandestine Production

Hidden nuclear material production would likely be as difficult as with any other reactor design since it would require placing fertile material near the core region. For a pebble bed core with continuous refueling once the undeclared target pebble (used for clandestine production) is introduced to the core it is no longer segregable from regular fuel pebbles. Misusing the MK1 for clandestine production would require selective retrival of large number of irradiated target pebbles, a task that is more challenging than simple diverison. Surveillance of areas outside the core is typically done to detect activities related to hidden production. After defueling, each pebble, fuel or graphite, is interrogated, examined, and sorted by kind. Thus, clandestine production by mixing target fuel pebbles with graphite blanket pebbles would most likely be detected by surveillance. Furthermore, clandestine production in the reactor using the same fuel would be unlikely due to the difficulties of reprocessing TRISO fuel.

4.3.3. Breakout

The use of TRISO fuel in a breakout strategy is also unlikely given that TRISO fuel is much more difficult to reprocess and currently much more expensive to manufacture. The use of a solid-fueled design would be unlikely due to the sheer size of the system that is needed. Like with the Very High Temperature Reactor designs, the reactor volume is large due to the distribution of fuel in the graphite matrix. TRISO fuel is designed to be a very mechanically robust fuel that will not be reprocessed. Therefore, it is unlikely that a state would choose this reactor method for a breakout scenario. There are many simpler reactor and fuel designs to produce material if breakout were intended.

4.4. Summary

The MSFR has interesting characteristics from the viewpoint of proliferation resistance. Liquid-fueled designs switch the reactor to a quasi-bulk handling facility that will require some method for measuring or estimating actinide content evolution over time in the molten salt. Due to relatively dilute concentrations of actinides in the MSFR, large amounts of salt with high radiological doses would be required for diversion of significant quantities, which presents handling and processing challenges for would-be proliferants. The unavoidable production of ^{232}U accompanying ^{233}U production, or Pu-238 and minor actinides for Pu-fueled systems, would generate very strong constraints on the handling of the material, preventing undesirable use and deterring any fuel transport. This would also produce a visible signature for the detection of fissile material transport. Liquid-fueled designs with replaceable cores remove some proliferation concerns, but push off some accountancy problems to a centralized salt-processing facility. Finally, solid-fueled MSR designs have proliferation resistance features similar to very high temperature reactors and can be dependent on item accounting. Some additional references for proliferation issues related to MSRs and their fuel cycles are Ref [52-56].

5. Physical Protection Features

This section discusses characteristics of MSRs that are important for the physical protection threats of theft and sabotage. All MSR fuel materials, regardless of fuel form, are radioactive to some degree. The presence of radiation presents an intrinsic barrier to theft in terms of health consequence and signature for detection. The typical remote handling of fuel salt in a hot cell environment (operating at temperatures above the salt melting temperature) makes physical access for theft or sabotage difficult or impossible. The relatively small quantity of fissile material in fuel pebbles renders them less desirable as targets for theft. The layered construction of fuel pebbles and their ability to sustain high temperatures also make them more robust against radiological sabotage. Furthermore, MSRs differ from most other reactor types because they use a low-pressure, chemically inert coolant, and thus do not have any stored energy sources to pressurize their containment boundary, minimizing a driving force for radiological releases during a sabotage event.

A more comprehensive evaluation of physical protection (PP) consequences is limited by the fact that neither academic plants nor prospective commercial plants have detailed physical layouts available for analysis. Plant layouts are generally of lower development priority for academic systems than reactor physics or thermal-hydraulics, whereas commercial designs will only provide cartoon level of design information in order to maintain control of the intellectual property within their design. Moreover, none of the prospective commercial designs have reached the level of maturity where all of the systems, structures, and components are fixed. Thus, discussion below of physical protection characteristics is mainly at the level of identifying potential targets.

5.1. Liquid-Fueled with Integrated Salt Processing

As the system layout is not firmly established for various liquid-fueled MSRs now, it is difficult to further discuss the physical protection issues linked to the reactor and the reprocessing unit. Two points are favorable. First, as previously mentioned, the unavoidable production of ^{232}U together with ^{233}U as well as the unavoidable co-production of minor actinides prevents easy handling and transport of fissile material. Secondly, one has to keep in mind that the fuel in an MSR remains in a hot cell environment because of the very high radiation levels, requiring shielding, remote operation and limiting access. The operation environment facilitates the application of containment and surveillance measurements for safeguards and physical security providing a large passive barrier to sabotage and the theft of materials. Though the fuel salt is in bulk form, it turns into solid when cooled below its melting temperature. About 31 liters of fuel salt from a fast spectrum equilibrium cycle contains the equivalent of 1 SQ of ^{233}U (8 kg). While a thermal spectrum MSR would require ~8x as much fuel for an SQ. The MSFR has very strong negative temperature reactivity, and the capability to drain the fuel provides the capability to provide long-term passive decay heat removal by passive means, which improves resilience against sabotage.

5.1.1. Theft of Material for Nuclear Explosives

The operating environment of the MSFR imposes passive barriers that guard the fuel salt against theft. The uranium from the MSFR is not directly usable in explosives, and the plutonium is diluted in a large volume of salt. These characteristics imply that fissile material theft by a sub-national group for construction of a nuclear weapon in most cases is not credible.

The only potential credible targets for theft in the MSFR system would be the ^{enr}U+TRU started reactor, where the ^{enr}U and the TRU material could be a credible target for theft. However, the fact that no fuel fabrication is required for the MSR makes it technically and economically possible not to have separated ^{enr}U and separated TRU material. In addition, spiking this ^{enr}U+TRU salt with a substantial concentration of gamma-emitting fission products (in particular, ^{137}CsF) could increase the detectability of the material and provide an enhanced barrier to theft during transport.

5.1.2. Radiological Sabotage

Sabotage of the molten salt coolant or cover gas should be considered for a physical protection analysis. Since most of the MSR designs keep the molten salt loops within containment, sabotage of these lines would be difficult. However, any possible penetrations outside of containment should be considered. Liquid-fueled MSRs require a free surface to provide a buffer against hydraulic shock (aka water hammer) and to allow fission gases to escape to avoid pressurizing the fuel salt container. Again, fission gas management system may extend beyond containment once the high activity gases have been allowed to decay (leaving primarily ^{85}Kr). Systems that are not protected by the containment could be a target. Liquid-fueled designs need to protect the molten salt coolants and cover gases from both accidents as well as intentional acts of sabotage. Note, designs have been proposed that submerge the entire primary loop, drain tanks, fission gas decay tanks, and fuel processing equipment under a pool of unfueled molten salt to serve a thermal buffer and radiation shield. Pool type designs provide smaller targets for physical sabotage.

5.2. Liquid-Fueled without Integrated Salt Processing

For physical protection of liquid-fueled MSRs without integrated salt processing, the IMSR from Terrestrial Energy can again be used as a reference design for discussion. A key feature of the IMSR design is that the reactor core, heat exchangers, pumps inlet plenum, outlet chimney, and gas plenum are all contained in a sealed reactor vessel (see Figure 1.2). The reactor vessel fits inside of a guard vessel of similar shape which forms the containment of the system and is capable of holding all the molten salt in case of a core leak. This integrated design provides an inherent barrier to theft or sabotage of the reactor core. There are, however, penetrations for the top-up of the reactor core during its 7-year operating lifetime. Fresh fuel is delivered to the IMSR plant in sealed containers; it is not clear in what volume of container the fuel would come. It is likely that sealed fresh fuel containers would be easier to steal than whole IMSR cores; if so, appropriate security measures would need to be in place to prevent this. Aside from this, a weak link is the transfer line between the fresh fuel and the IMSR core in operation. Depending on the design, this transfer line could in principle be tapped into temporarily for stealing fuel. As an IMSR does not require large fuel additions at a single time, the diversion would need to take place over an extended period. In the event of a sabotage attack on the operating core, the transfer line could be a vulnerable target to cause release of gaseous fission products.

The IMSR plant layout (see Figure 1.3) allows for the storage of multiple reactor core units on site. Similar to the reactor core in operation, the integrated design of the reactor units provides a significant barrier to theft of fuel from the units in storage, or to their sabotage; a weak point perhaps in either case are the few penetrations present in the design. The IMSR plant design also includes facilities for defueling the reactor core at the end of its operating life, into spent fuel storage tanks. Measures are necessary to prevent the use of the defueling facilities for theft or sabotage purposes, either on the core in operation, or on the core units in storage. Further, the spent fuel storage tanks could also be vulnerable targets for theft or sabotage, depending on what protective measures are put in place. Note, the spent fuel would consist of massive, robust solid blocks of highly radioactive fuel salt, so would be difficult to steal or damage.

5.3. Solid-Fueled with Molten Salt Coolant

The Mk1 PB FHR reference [31] does provide a site layout with typical physical protection features to use as a basis. Generally, most of the plant features are those typical of any reactor with vital systems protected accordingly. The purpose of this analysis is to determine if there are unique features to solid-fueled MSRs that might require additional thought. Of the three classes of MSRs, only the solid-fueled with salt coolant design has spent nuclear fuel (SNF). The FHR and the pebble-bed VHTR use the same basic fuel (graphic-matrix coated particle fuel). However, there are differences between the SNF from the two reactor designs that have physical protection implications [57]. The salt coolant provides more efficient cooling of the FHR fuel allowing power densities that are four to ten times higher than the VHTR without exceeding the fuel temperature limits. Also, the salt coolant provides some neutron moderation reducing the carbon-to-uranium ratio of the FHR fuel. Consequently, the volume of FHR SNF is half to a third that of a VHTR per unit energy output. The correspondingly higher heavy metal loading (and burnup) in the FHR fuel (optimized loading is two to three times that of the VHTR fuel) implies that the SNF decay heat is much higher in the FHR SNF per unit volume immediately after fuel discharge and is substantially higher in the long term as well relative to the VHTR SNF. Furthermore, the FHR SNF volume is about four times that of an LWR per unit energy output. These features of the FHR will impact the safeguards and protection of its SNF in storage and repository disposal.

The molten salt coolant will contain tritium and trace amounts of other radioactive components, so sabotage of the coolant lines should be considered. The location of radioactive gases should be identified to determine if all locations are contained within the protected area. Likewise, graphite components in the reactor will also contain significant radioactivity. However, this material is all contained in the protected area in the reference design. One particular area that should be evaluated is if there is any possibility for thermal decomposition of the graphite fuel or components, particularly non-nuclear grade graphite. Exposing the carbon materials to a flowing high temperature air environment would represent a significant release pathway, so the reactor should be designed with this particular sabotage threat in mind. In addition, spent fuel storage should also consider the risk of this sabotage threat.

There is little that can be done to a pebble bed reactor to create a reactor excursion event (from an external attack perspective). These designs are fairly robust to transients, and an external attack inside containment is also difficult to carry out.

The solid TRISO fuel is likely in a more stable or equivalent material form as compared to light water reactors, so no additional vulnerabilities would be present with fresh and spent TRISO fuel. The absence of any fuel salt processing on-site is an advantage in that it leads to less targets to consider as compared to the other MSR designs.

Theft of material for radiological dispersal devices could potentially be a more significant concern since the theft of one pebble may go un-noticed. In the absence of individual pebble accounting, more containment would be suggested to prevent theft for RDD devices.

5.4. Other Observations

The potential for large civilian aircraft impact and seismic performance requirements both provide substantial incentive for locating plant structures partially or fully below-grade. Because molten salt has high volumetric heat capacity, the primary loop and containment of an MSR will be much more compact than corresponding sodium, helium, or water-cooled reactors, reducing the cost of below-grade construction. However, salt processing facilities could significantly increase the size of the required below-grade facilities. Alternately, the site may be made effectively below grade by constructing a hoop or dome over surface facilities and covering it with a couple of meters of sand (similar to a munitions bunker).

Driven by market forces and enabled by the high degree of passive safety, MSRs appear likely to have substantially different staffing roles and concepts of operation than LWRs. For example, MSRs will have no safety-related diesel generators or connections to off-site power. Operators also will not be required to respond to any as yet postulated accident scenario to prevent the release of radionuclides to the environment. Some designs may have no accessible vital areas and consequently may rely on local law enforcement rather than dedicated plant security forces to respond to site intrusions during normal operations.

6. PR&PP Issues, Concerns, and Benefits

All three classes of MSR designs considered in this study have intrinsic and design features that are favorable to PR&PP. The differences in their PR&PP characteristics are most evident in their adopted fuel cycles that manifest in variations in their respective system elements and potential adversary targets (Section 3). In addition, remote operation behind shielded vaults and the use of low-pressure and chemically inert coolant contribute to the physical protection robustness of MSRs.

The MSFR is characterized by reduced fissile inventory in the fuel salt due to its high power density and the absence of excess fuel reactivity for operations. The fissile material is disseminated in small quantity (some %) in the fuel salt. Obtaining a significant quantity (SQ) of fissile material would require a sizable amount of fuel salt and its absence from the reactor is readily detectable. In order to avoid proliferation problems for MSFR started with the Pu and minor actinides produced in LWRs (TRU-started MSFR) the initial Pu is shown to contain enough ^{238}Pu (more than a few %) to reduce the material attractiveness. For the U-started MSFR the unavoidable production of ^{232}U accompanying ^{233}U production, even in small fractions, would generate strong constraints on the handling and transport of uranium, i.e. imposing additional technical difficulties.

Many of the PR observations for the MSFR are applicable to the liquid-fueled MSRs without integrated salt processing. However the option of a centralized salt processing facility located elsewhere in the fuel cycle to handle the salt or core every 4-8 years would still have some of the accountancy challenges to a bulk reprocessing facility, as in the case of the MSFR. A notable feature of the IMSR is the system for regular fuel top-ups that will not permit operators to subtract fuel from the core, precluding breeding and diversion in the main reactor vessel.
The distinguishing feature of the solid-fueled MSR design using TRISO fuel is its robust fuel, greatly diluted in carbonaceous material and high burnup. Solid-fueled MSRs are closer to item-counting facilities as opposed to bulk handling facilities for liquid-fueled designs. Diversion or misuse of the TRISO-particle-fueled MSR to acquire significant quantity of U or Pu is hindered by the need to obtain a large number of hard-to-process fuel particles.

A common feature of all MSRs, regardless of fuel form, that contributes to their physical protection robustness is their radioactive fuel materials. The high radiation presents an intrinsic barrier to theft. The typical remote handling of fuel salt in a hot cell environment (operating at temperatures above the salt melting temperature) makes physical access for theft or sabotage difficult or impossible. The relatively small quantity of fissile material in fuel pebbles renders them less desirable as targets for theft. The layered construction of fuel pebbles and their ability to sustain high temperatures also make them more robust against radiological sabotage. Furthermore, MSRs differ from most other reactor types because they use a low-pressure, chemically inert coolant, and thus do not have any stored energy sources to pressurize their containment boundary, minimizing a driving force for radiological releases during a sabotage event.

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APPENDIX 1: Summary of PR relevant intrinsic design features. Please refer to IAEA-STR-332 [58], for full explanations and complete definitions of terms and concepts.

Summary of PR relevant Intrinsic design features	**Liquid-Fueled with Integrated Salt Processing (ex: MSFR)**	**Liquid-Fueled without Integration Salt Processing (ex: IMSR)**	**Solid-Fueled with Molten Salt Coolant (ex: Mk1 PB FHR)**
Features reducing the attractiveness of the technology for nuclear weapons programmes			
1. The Reactor Technology needs an enrichment Fuel Cycle phase	Yes initially, but thorium cycles require no enrichment after startup.	Yes initially, but thorium cycles require no enrichment after startup.	Yes
2. The Reactor Technology produces SF with low % of fissile plutonium	Fissile content is maintained in the molten salt at a steady rate.	Fissile content is maintained in the molten salt at a steady rate	SF will be produced with low % fissile plutonium
3. Fissile material recycling performed without full separation from fission products	Recycling is performed on-site, but actinides do not need to be separated (only FP removal)	Only gases and noble metals are removed on-site, but centralized processing used to process the salt periodically (every 7-8 years)	No recycling
Features preventing or inhibiting diversion of nuclear material			
4. Fuel assemblies are large & difficult to dismantle	Liquid fueled. Fuel salt solidifies into robust masses during extended storage.	Liquid fueled. Fuel salt solidifies into robust masses during extended storage.	Fuel pebbles are small, but 250,000 required to accumulate a significant quantity.
5. Fissile material in fuel is difficult to extract	Fissile material requires molten salt separation process for removal	Fissile material requires molten salt separation process for removal	TRISO fuel is difficult to reprocess.

Summary of PR relevant Intrinsic design features	**Liquid-Fueled with Integrated Salt Processing (ex: MSFR)**	**Liquid-Fueled without Integration Salt Processing (ex: IMSR)**	**Solid-Fueled with Molten Salt Coolant (ex: Mk1 PB FHR)**
6. Fuel cycle facilities have few points of access to nuclear material, especially in separated form	Processing of fuel salt is done on site, but radiation environment is very high. Operation is done with limited access because of radiation and high temperature.	Contained reactor designs limit access points. Only top-off U is stored onsite.	Fuel cycle facilities mainly involve pebble handling, and remote operations are required.
7. Fuel cycle facilities can only be operated to process declared feed materials in declared quantities	Accounting of the salt processing loops may pose challenges due to radioactivity, high temperature, volume of salt, and lack of precision in the nuclear cross sections and reactor physics models.	Accounting of the salt processing loops may pose challenges due to radioactivity, high temperature, and volume of salt. No on-site fuel salt processing besides FP removal.	N/A
Features preventing or inhibiting undeclared production of direct-use material			
8. No locations in or near the core of a reactor where undeclared target materials could be irradiated	Blankets exist for some designs. Other designs submerge primary salt loop in unfueled molten salt making area inaccessible, but also difficult to observe.	Drop-in core designs may make it difficult to place undeclared target materials.	Difficult-to-access area There is no dedicated location in the core for introduction of undeclared target material. Any target material introduced will be mingled with the rest of fuel pebbles.
9. The core prevents operation of the reactor with undeclared target materials (e.g. small reactivity margins)	Other materials could be added to the fuel salt, but very unlikely to occur since it would get diluted in the large salt volume.	Other materials could be added to the fuel salt, but very unlikely to occur since it would get diluted in the large salt volume.	It is possible to introduce U-238 target pebbles for breeding, but would be difficult to sort out the target pebbles from regular fuel pebbles after irradiation.

Summary of PR relevant Intrinsic design features	**Liquid-Fueled with Integrated Salt Processing (ex: MSFR)**	**Liquid-Fueled without Integration Salt Processing (ex: IMSR)**	**Solid-Fueled with Molten Salt Coolant (ex: Mk1 PB FHR)**
10. Facilities are difficult to modify for undeclared production of nuclear material	Separations systems could be modified for undeclared production. However, very high radiation levels make undeclared modifications technically difficult.	Contained units make modification more difficult.	Unlikely with pebble bed designs.
11. The core is not accessible during reactor operation	Very high radiation environment.	Some designs have self-contained cores that are designed not to be accessed.	Very high radiation environment.
12. Uranium enrichment plants (if needed) cannot be used to produce HEU	Enrichment not needed after startup.	Enrichment may or may not be required after startup depending on operating in open or closed fuel cycle.	Expect international safeguards in place to deter HEU production.
Features facilitating verification, including continuity of knowledge			
13. The system allows for unambiguous Design Information Verification (DIV) throughout life cycle	DIV should be straight-forward.	DIV should be straight-forward, but sealed cores could be problematic.	DIV should be straight-forward.
14. The inventory and flow of nuclear material can be specified and accounted for in the clearest possible manner	Liquid-fueled designs will present MC&A challenges since they are more like bulk handling facilities.	Liquid-fueled designs will present MC&A challenges since they are more like bulk handling facilities.	Item accounting facility, pebbles can be accounted for.
15. Nuclear materials remain accessible for verification the greatest practical extent	Verification may be difficult due to the liquid fuel.	Verification may be difficult due to the liquid fuel.	Verification of pebbles may pose challenges.

Summary of PR relevant Intrinsic design features	**Liquid-Fueled with Integrated Salt Processing (ex: MSFR)**	**Liquid-Fueled without Integration Salt Processing (ex: IMSR)**	**Solid-Fueled with Molten Salt Coolant (ex: Mk1 PB FHR)**
16. The system makes the use of operation and safety/related sensors and measurement systems for verification possible, taking in to account the need for data authentication	Process monitoring technologies are being examined currently, but will require more R&D. Likely entrance and exit based assays will be primary measurements.	Process monitoring technologies are being examined currently, but will require more R&D.	Pebble bed measurement systems needed for operation can also be used for safeguards.
17. The system provides for the installation of measurement instruments, surveillance equipment and supporting infrastructure likely to be needed for verification	More R&D needed.	More R&D needed.	See above response.

APPENDIX 2: The Molten Salt Actinide Recycler and Transmuter (MOSART)

Besides the MSFR, under development within the GIF framework [11] is another fast neutron spectrum reactor with circulating fluoride-based fuel in a closed fuel cycle, the Molten Salt Actinide Recycler and Transmuter MOSART (Russia Federation). The main design objective of the single fluid 2.4 GWt MOSART is to close the nuclear fuel cycle for all actinides (burner option), including Np, Pu, Am and Cm and to be collocated with an aqueous spent nuclear fuel (SNF) reprocessing plant at the Mining and Chemical Combine (MCC) site [42]. It is assumed that the fuel cycle of this complex will be organized as follows (see Figure II.1): the bulk of the removed uranium and plutonium return to thermal and fast solid fuel reactors, and the remaining TRUs are transferred for utilization in the MOSART system.

In a preliminary PR analysis performed on MSR designs by members of the MSR pSSC from Europe and Russia, some observations were noted for the MOSART in ref. [42] and they are repeated here.

"The main advantages of MOSART are the ability to vary widely the MA content in fuel salt without losing the inherent safety and the absence of stages related to the fuel fabrication and re-fabrication in multiple actinides recycling. As result there are significant PR and safeguards implications related to the fuel make up and chemical processing in MOSART plant: (1) there will be continuous variation of isotopic concentrations in the fuel salt from both TRU transmutation and chemical processing; (2) refueling scheme include the ability to continuously feed the core with fresh fissile material; (3) plate-out of noble metals in the primary circuit could complicate inventory tracking.

Fuel salt represents a unique combination of high-temperature and high-radiation environments that will be challenging for diversion as well as measurement techniques and instrumentation: (a) temperature in the reactor or fuel processing plant will always be kept in liquid state within 550 -7200C; and (b) fuel salt will be highly radioactive even outside the primary circuit.

In order to avoid nuclear matter diversion MOSART reactor plant is integrated (1) at the front end with VVER SNF aqueous reprocessing plant and (2) at the back end with the high temperature fuel salt clean up facility all located at the MCC site. All fresh fuel fluorides containing significant quantities of fissile materials (Pu+MA) for initial loading and make up, will be manufactured onsite by hydrofluorination process. In molten salt pyroprocessing facility the higher actinides would always accompany the plutonium, this operation would never produce a "clean" material would be attractive for diversion. Last TRU loading will be transferred to the next MOSART reactor plant to be constructed at the MCC site"

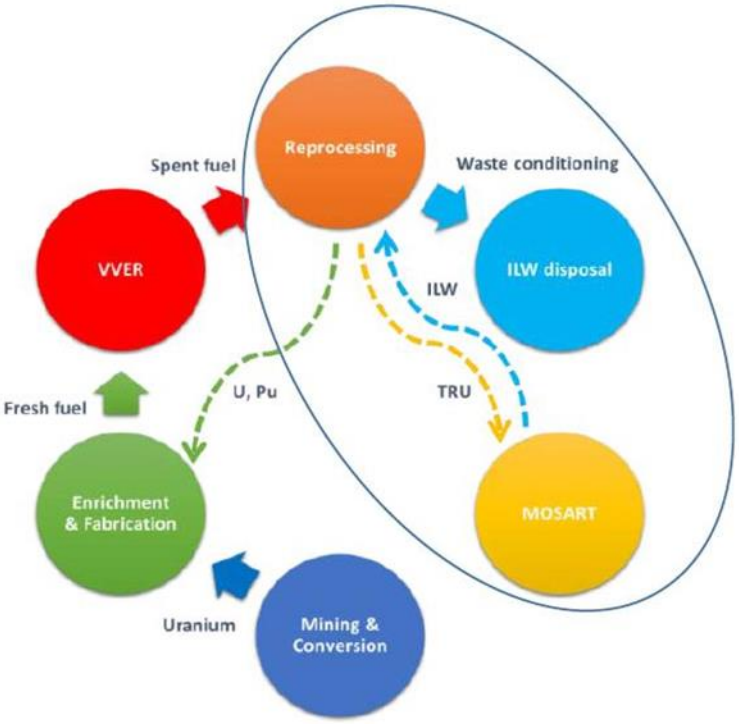

Figure II-1. Nuclear fuel cycle with MOSART at MCC site. [42]

APPENDIX 3: Countermeasures for the MSFR

The following is a reproduction of material from “Preliminary Proliferation Study of the Molten Salt Fast Reactor”, EPJ Nuclear Sci. Technol. 6, 5 (2020) [59]

1. Countermeasures

“The main target for Pa or U diversion is the fertile blanket of a breeder reactor. Since an MSFR can be operated without a blanket while ensuring quasi break-even fuel breeding, a first option consists in delivering only blanket-free MSFRs to risk prone States. The need that then arises to periodically inject fissile material in the fuel salt so as to ensure good reactivity precludes any diversion of Pa: the flow of necessary fissile material would have to be increased to compensate for the missing U that the diverted Pa would have produced. In the presence of a blanket, the most efficient diversion is that of Pa that rests on the ability to separate the elements in the fuel cleaning unit. The methods used in this unit are not precisely determined and options remain to be chosen. Similarly, work needs to be done to determine how this unit will be organized.”

1.1 Choice of Actinide Separation Methods

“The main proliferation risk is related to the possibility of separating the Pa from the other actinides and from all the ^{232}*Pa descendants (U, Th, and Ra essentially). This separation would be done at first when the Pa is extracted from the fuel salt and the blanket and subsequently repeated regularly to conceal the storage of Pa. The two operations can be distinct but must make use of the methodology available in the fuel salt cleaning unit. The less efficient the separation techniques are, the better the proliferation resistance will be. Indeed, the fuel composition adjustment as well as the utilization of the U from breeding do not require a good separation efficiency, since the actinides have to be recycled. It is thus possible to limit the risks associated to these means of separation by opting for inefficient separation methods.*

Two methods are being considered for the extraction of the actinides: fluorination and reduction (chemical or electrochemical) in a metallic bath.

Fluorination consists in forming gaseous actinide fluorides via the oxidation of the salt by gaseous fluorine. These fluorides are produced at temperatures ranging between 600 and 900°C, the gases being subsequently cooled and condensed on inert or reactive (alkaline fluorides) media. Depending on the operating conditions, the U (UF_6*) and other actinides (Pa, Np, Pu) are also removed but not the Th, or the minor actinides. The fluorination has another function, i.e. the extraction of some elements such as O, I, S, Se, Te, Cr, Mo which produce fluorides with low condensation temperatures, lower than or similar to that of* UF_6*. This means that it is not easy to condense the wastes and the actinides separately. Ideally, all the actinide fluorides would be condensed together in a temperature range that would allow the separation of a large part of the wastes. The non-separation of the actinides on distinct physical containers could be a means to reinforce proliferation resistance. This issue needs further study.*

Using the fluorination device to periodically remove the U produced by the decay of Pa, by vaporizing only the U, would leave the Th and the Ra with the Pa without suspending the decay chain leading to ^{208}*Pb. If the U and the Pa were to be vaporized together (requiring high temperature), then another separation, that of Pa/U, would have to be done immediately, while avoiding the vaporization of* PaF_5 *(at low temperature).*

The reduction of actinides to a metallic state dissolved in liquid Bi is a method that, in principle, does not allow as good a separation of the elements as fluorination (on the order of 90% in one passage, compared to >99% in the case of fluorination) A difficulty, that has already been identified, is that a fraction of the Th is transferred to the metal along with all of the reduced actinides. It is thus not possible to fully break up the decay chain of ^{232}*Pa to* ^{208}*Pb. This actinide extraction method is less proliferant than the vaporization of the fluorides but it is much more* *cumbersome because it requires many more steps. It has not yet been validated experimentally but it could be if this method were to be considered essential for the extraction of the lanthanides in the presence of Th (see section 2.3 of Ref [59]).*

The methods used for salt cleaning and ^{233}U *extraction from the blanket are still an open issue, the final choice will have to consider the possible consequences on proliferation resistance."*

1.2 Detection of Material Transfers

"Batch transfers of materials can be observed as they transit through the control chambers, or they can be detected by way of their consequences on the isotopic balances. Provided a full history of the power generated by each reactor, of the amounts of salt processed, and of the fuel temperatures is available, it is possible to monitor the full data set consistency with a simulation program. The reliability and the precision of such a program remains to be assessed.

Note that, to obtain one significant quantity (SQ) of ^{233}U *(8 kg) from a diversion of Pa dissolved in Bi, one would have to execute 50 out of site transfers of a Bi mass on the order of 500 kg, the Bi having been stored and processed in the cleaning unit during 2 weeks; the Pa would then have to be concealed for 3 months in a separate installation to finally obtain the desired* ^{233}U*.*

The salts originating from a reactor generate residual heat that can be considerable so that the transfer vehicles need to have a large thermal inertia; their mass must then be large compared to that of the salt they carry. By limiting as tightly as possible the transfer capacities, with the possibility of more frequent transfers if needed, a limit is set on the masses that can be covertly handled. In this respect, the question arises: should the transfers within the fuel cleaning unit be submitted to specific monitoring to allow the detection of illegitimate storage that is required by the diversion of Pa? This unit would then be subdivided to form multiple elements, each containing a chemical reactor or a temporary storage. Each element would be placed in a well surrounded by a radiation shield to reduce the background noise in the unit and allow, via directional radiation detection, to monitor the inputs and outputs of each well. To prevent any modification of the initial design, the space available in each well would be limited to the exact size of the chemical reactor or to the dimensions of the device for the foreseen temporary storage needs."

1.3 Fuel Storage Before Processing

"Fluorination is a very efficient method for the extraction of the U from the blanket salt, which is the main source of Pa (116 kg inventory in 7.7 m^3 *of salt). About 40 liters of this salt have to be processed each day (0.63 kg Pa per day). This technique is generally considered efficient for Pa extraction. In order to reduce proliferation risks, it could be advisable to store the samples taken from the blanket for 6 months before transferring to the U extraction unit. During this time span, 99% of the* ^{233}Pa *has decayed and produced* ^{233}U *mixed with* ^{232}U*. In this manner, the source of Pa would not reach the chamber containing the devices that could be used to divert the Pa. However such a storage would generate higher operating costs so that doing without a blanket altogether might be a preferred solution."*

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THE GENERATION IV INTERNATIONAL FORUM

Established in 2001, the Generation IV International Forum (GIF) was created as a co-operative international endeavor seeking to develop the research necessary to test the feasibility and performance of fourth generation nuclear systems, and to make them available for industrial deployment by 2030. The GIF brings together 13 countries (Argentina, Australia, Brazil, Canada, China, France, Japan, Korea, Russia, South Africa, Switzerland, the United Kingdom and the United States), as well as Euratom – representing the 27 European Union members and the United Kingdom – to co-ordinate research and develop these systems. The GIF has selected six reactor technologies for further research and development: the gas-cooled fast reactor (GFR), the lead-cooled fast reactor (LFR), the molten salt reactor (MSR), the sodium-cooled fast reactor (SFR), the supercritical-water-cooled reactor (SCWR) and the very-high-temperature reactor (VHTR).

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