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GEN IV International Forum

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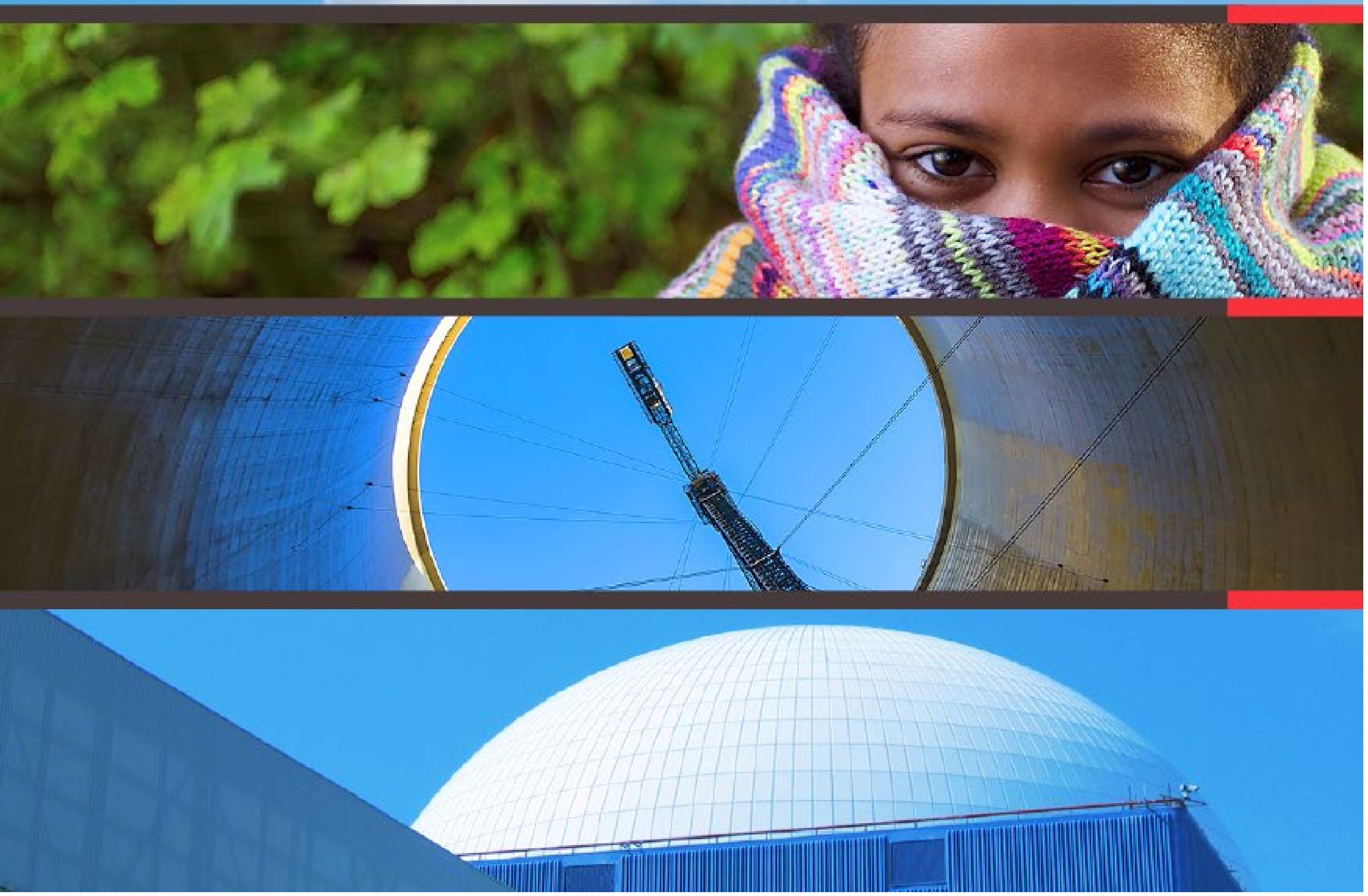
GIF LEAD-COOLED FAST REACTOR

Proliferation Resistance and Physical Protection White Paper

October 2021



Preface to the 2021 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 Sodium-cooled fast Reactor (SFR), the Very high temperature reactor (VHTR), the gas-cooled fast reactor (GFR), the Molten salt reactor (MSR) and the Supercritical water–cooled reactor (SCWR).

The Proliferation Resistance and Physical Protection Working Group (PRPPWG) was established 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 systems 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 Lead Fast Reactor reference designs selected by the Generation IV International Forum (GIF) Lead Fast Reactor (LFR) provisional System Steering Committee (pSSC). The intent is to generate preliminary information about the PR&PP merits of the LFR reactor technology and to provide insights for optimizing their PR&PP performance for the benefit of LFR system designers. It updates the LFR 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 LFR pSSC, follows the high-level paradigm of the GIF Proliferation Resistance and Physical Protection Evaluation Methodology to investigate the PR&PP features of the GIF LFR reference designs. For PR, the document analyses 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 PP, the document discusses the robustness against theft of material and sabotage by non-State actors.

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List of Acronyms

BREST-OD	Russian acronym for 'Fast Reactor with Lead Coolant'
CBR	Core Breeding Ratio
C/S	Containment & Surveillance
DB-Pu	Deep Burn Plutonium
DCs	Decay Heat Removal Dip Coolers
DHR	Decay Heat Removal
ELFR	European Lead-cooled Fast Reactor
FA	Fuel Assemblies
FP	Fission Products
GIF	Generation IV International Forum
HM	Heavy Metal
IDU	Irradiated Direct-Use
IAEA	International Atomic Energy Agency
IV	Inner Vessel
LEU	Low-Enriched Uranium
LFR	Lead-cooled Fast Reactor
MA	Minor Actinides
MOX	Mixed Oxide Uranium Plutonium
MNUP	Mixed Nitrides Uranium Plutonium
NPPs	Nuclear Power Plants
pSSC	Provisional System Steering Committee
PR&PP	Proliferation Resistance & Physical Protection
PP	Physical Protection
PPSs	Physical Protection Systems
PPs	Primary Pumps
PR	Proliferation Resistance
RG-Pu	Reactor-grade Plutonium
RSWG	Risk and Safety Working Group
RV	Reactor Vessel
SGs	Steam Generators
SQ	Significant Quantity
SMRs	Small Modular Reactors
SSTAR	Small Secure Transportable Autonomous Reactor
SRP	System Research Plan
TRU	Trans Uranium
UDU	Un-irradiated Direct-Use
WG-Pu	Weapons-Grade Plutonium

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1 Overview of Technology

Among the promising reactor technologies being considered by the Generation IV International Forum (GIF), the Lead-cooled Fast Reactor (LFR) has been identified as a system with great potential to meet needs for both remote sites and central power stations [1]. The LFR promises to readily meet the Generation IV objectives of Sustainability, Economics, Safety and Reliability and Proliferation Resistance & Physical Protection (PR&PP), based both on the inherent features of lead as a coolant and on the specific engineered designs. In this document, the PR&PP approaches and characteristics of selected LFR concepts (i.e., the reference designs identified by the LFR provisional System Steering Committee, LFR-pSSC) are presented and, where appropriate, features common to LFRs (i.e., applicable to other concepts under development beyond the reference designs) are also noted.

The Sustainability, Economic, Safety and Reliability attributes of the LFR are largely driven by the fundamental characteristics of lead as a coolant, and in some cases these characteristics also have important implications for the PR&PP characterization of these systems. Because lead is a coolant with very low neutron absorption and energy moderation, LFRs operate in a fast neutron flux and can readily achieve a conversion ratio of 1, thus enabling a long core life and a high fuel burnup while enabling effective minor actinide management. Molten lead offers excellent neutronic performance, is chemically relatively inert with air and water, and exhibits low vapour pressures with the advantage of allowing operation of the primary system at atmospheric pressure. A low dose to the operators can also be predicted, owing to its low vapour pressure, high capability of trapping fission products, and high shielding of gamma radiation. Because of the fundamental characteristics of molten lead, it is possible to significantly simplify LFR systems and produce systems that are compact and hence deliver a high level of power from a relatively small primary system volume, enhancing safety [2], [3] as well as proliferation resistance and physical protection.

This document, an update of a previous White Paper [4][1], presents the three reference-LFR systems ([5], [6]) that have been chosen as a basis for the pSSC activities:

- A large system rated at about 600 MWe, based on the system concept known as the European Lead-cooled Fast Reactor (ELFR) [7], [8][2];
- A medium sized system, the BREST-OD-300 (300 MWe) representative of an intermediate power level plant [9];
- A small transportable system of 10–100 MWe size, based on the system concept known as the Small Secure Transportable Autonomous Reactor (SSTAR) [10], [11].

Other concepts are being developed worldwide but are not included in detail here. Examples of these include LFR concepts being developed by Westinghouse, Hydromine, LeadCold, INEST and NUTRECK ([12], [13], [14], [15], [16]). The three systems chosen as GIF reference systems, however, are thought to be reasonably representative of possible sizes (including SMR option) and design choices.

Of the three reference systems, two (i.e., BREST-OD-300 and SSTAR) can be considered within the power range of Small Modular Reactors (SMRs) as they both have design output powers within the 300 MWe limit usually considered as an upper bound for SMRs (i.e., 300 MWe and 20 MWe, respectively). While the BREST-OD-300 is intended to be a demonstration reactor leading to commercial systems of significantly larger size and might not necessarily be considered as modular, factory built and transported on site, the SSTAR was intended to be a small,

[1] This White Paper differs significantly from the previous version published in 2011 [4] by virtue of text revision, updates to reactor design information, extended analysis and the inclusion of a new reference system (BREST-OD-300).
[2] The ELFR evolves from the former ELSY design, which was considered in the 2011 LFR WP [4].

transportable system from its inception. Note further that several of the other LFR concepts under development (i.e., those under development by Hydromine, LeadCold, INEST and NUTRECK) [17] are also considered SMRs or micro-reactors designs.

Power conversion efficiencies for each of the reference systems are in the 40% range (42% for the steam cycle of both the BREST and ELFR plant and 44% for the supercritical CO_2 Brayton cycle of SSTAR). Besides obvious differences related to size, the three systems taken as references for the GIF-LFR-pSSC activities share a significant number of technical issues and many common features, especially as far as safety design is concerned; thus, there are many commonalities from the point of view of design and engineering of these systems and the solutions adopted [2].

As discussed later in the paper, the use of mixed oxide or nitride fuels containing Minor Actinides (MA) might increase intrinsic Proliferation Resistance (PR) because of the inherent properties of the nuclear material. Moreover, some LFR concepts (e.g. SSTAR) have been designed from the beginning to enhance non-proliferation goals by incorporating a sealed core and very long-life fuel thereby significantly reducing major potential pathways for unauthorized use of nuclear materials. The use of a coolant chemically compatible with air and water and operating at ambient pressure greatly enhances Physical Protection (PP) robustness to sabotage efforts. In particular there is reduced need for robust protection against the risk of catastrophic events, initiated by acts of sabotage, and there is a little risk of fire propagation. There are no credible scenarios of significant containment pressurization.

Sections 1.1.1, 1.1.2 and 1.1.3 present the description of the three reference systems as approved by the pSSC and used in other GIF documents (e.g., the Risk and Safety Working Group –RSWG- LFR White Paper [2]). Other additional information needed for the purposes of this paper will be included in the relevant Sections.

An important point to be noted is that in the case of the BREST OD-300 reactor, fuel cycle facilities (i.e., fuel fabrication and reprocessing facilities) are co-located on the reactor site for this demonstration project. This co-location is not an intrinsic characteristic of the BREST concept, and in future commercial installations it is possible that such co-location would not take place in favour of centralized fuel cycle facilities to support multiple reactor sites. As a result, for the analyses and discussions of this White Paper, the focus is on the reactor facility and operations directly related to it (e.g., fresh and spent fuel transfer and storage), and not on the incidentally co-located fuel fabrication and processing processes and facilities. The impact of co-location on PR&PP will be addressed as a cross-cutting topic for all the six GIF reactor technologies in a future work.

1.1 Reference System Descriptions

A brief description of the three reference systems for GIF LFR related activities is here presented [2].

1.1.1 The European Lead-cooled Fast Reactor ELFR

The ELFR system is an evolutionary design representing a modification to the earlier ELSY reactor concept. Figure 1 provides an overview sketch of the ELFR reactor vessel and its contents.

Some characteristics of the ELFR design – in one of its configuration options, as disclosed within the GIF – are summarized in Table 1. The ELFR primary system has a pool-type configuration, with the main vessels supported by a Y-support holding the main vessel in the upper part.

The Reactor Vessel (RV) has been kept as compact as possible to reduce the coolant inventory and the corresponding seismic loads, while being of sufficient size to accommodate the required number of components (i.e., 8 Steam Generators (SGs), 8 Primary Pumps (PPs), and 8 Decay Heat Removal Dip Coolers (DCs)).

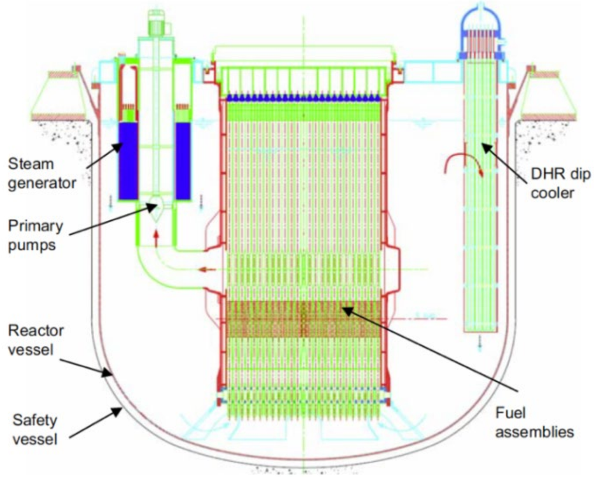

Figure 1: ELFR - the European Lead-cooled Fast Reactor [18].

Table 1: ELFR Summary Parameters.

Power 1,500 MW (th)	~600 MW(e)
Core diameter	4.5 m
Core height	1.0 m
Core fuel	MA-bearing MOX
Coolant temp.	400/480°C
Maximum clad temp.	550°C
Net efficiency	~42%
Core breeding ratio -CBR	~ 1

The hot pool of the ELFR vessel is enclosed by an Inner Vessel (IV), connected to the Primary Pumps (PPs) through suction pipes. Each primary pump is installed at the centre of its corresponding SG, which transfers the heat from primary lead coolant to water-steam in a superheated cycle. The free level of the hot pools inside each SG/PP unit is higher than the free level inside the Inner Vessel, the different heads depending on the pressure losses across component parts of the primary circuit. The design is based on a core pressure loss of 0.9 bar and a total primary pressure loss of 1.4 bar. The core inlet and outlet temperatures are 400 °C and 480 °C, allowing for a sufficient margin in the cold plenum from the freezing point (327°C) of the lead coolant, while reducing the potential for embrittlement (for structures wetted by cold lead) and corrosion (for structures in hot molten lead). The speed of the primary coolant is well below 2 m/s (reaching 10 m/s only at the tips of the pump impeller) to limit erosion.

The internal reactor component arrangement and design presents a simple flow path for the primary coolant. The locations of the heat source (within the core) and of the heat sinks (SGs) allow for efficient natural circulation of the coolant under emergency shutdown conditions. Two safety systems for Decay Heat Removal (DHR) have been considered as an integral part of the design from the beginning of the activities. They are characterized by passive operation, diversity and redundancy while, in addition, being completely independent from one another.

The design of the core has been driven by the implementation of the so-called "adiabatic" [19] reactor concept. The adiabatic reactor concept concerns the operation of a reactor with an equilibrium fuel cycle, so that the fuel composition remains the same between two successive loadings, ensuring the full recycling of all the actinides, with either natural or depleted uranium as only top-up/input material and fission products as well as reprocessing and fabrication losses as outputs, as illustrated below in Figure 2 (for the n^{th} cycle, at the equilibrium). This approach is conceptually very similar to that used for BREST-OD-300.

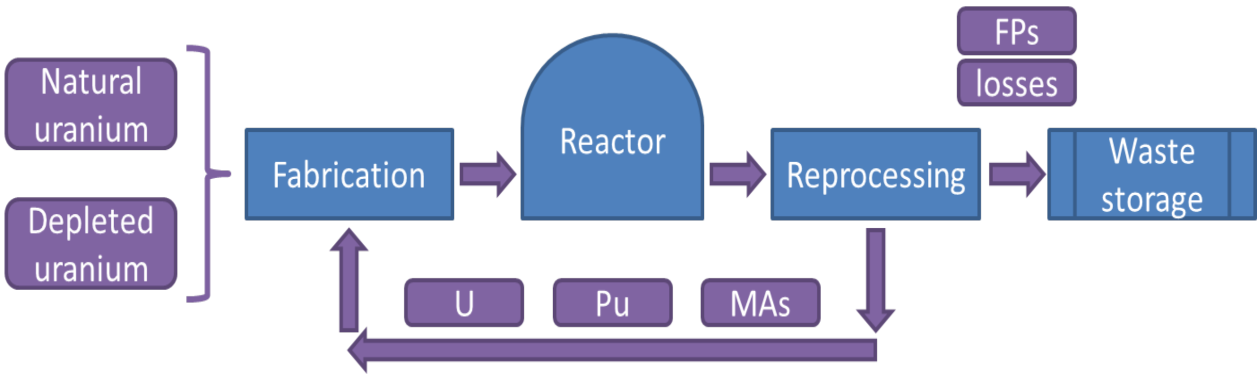

Figure 2: ELFR Fuel Cycle at equilibrium in adiabatic operation [2].

The adiabatic operation implies (by definition) iso-generation of plutonium within the driver fuel: accordingly, no blanket position was foreseen in designing the ELFR core.

Several options exist for the first core loading, the choice of the reference one depending on the actual policy and materials availability by the operating utility. Among the main options, a first core made of pure MOX (with depleted uranium and reactor-grade plutonium) and a MOX fuel doped with the equilibrium content of MAs (although with different isotopic composition than in the equilibrium fuel vector) are the main ones: the former to ease the first core manufacturing; the latter to prevent additional MAs formation already from the first operation cycle[3].

1.1.2 The BREST-OD-300

The BREST-OD-300 reactor is a pilot demonstration reactor (300 MWe) considered as a prototype of future commercial reactors of the BREST family for large-scale nuclear plants characterized by the idea of “natural safety.” Figure 3 provides an overview sketch of the BREST-OD-300 system.

Some of the relevant characteristics of the BREST-OD-300 design are summarized in Table 2.

BREST-OD-300 is a reactor facility of pool-type design, which incorporates within the pool the reactor core with reflectors and control rods; the lead coolant circulation circuit with steam generators and pumps; equipment for fuel reloading and management; and safety and auxiliary systems. The reactor equipment is arranged in a steel-lined, thermally insulated concrete vault.

BREST has a widely-spaced fuel lattice with a large coolant flow area, resulting in low pressure losses, favouring the establishment of primary natural circulation for decay heat removal. It shares with other designs the absence of uranium blankets, replaced by a lead reflector with the proper albedo to improve power distribution, providing negative void and density coefficients, and ruling out the net production of weapons-grade plutonium in its standard closed fuel cycle. The BREST decay heat removal systems are characterized by passive and time-unlimited residual heat removal directly from the lead circuit by natural circulation of air through air-cooled heat exchangers, with the heated air vented to the atmosphere.

[3] The analysis of Section 3, 4 and 5 considers the reactor operating at equilibrium (Nth fuel cycle), therefore these different options do not have any impact on the following sections.

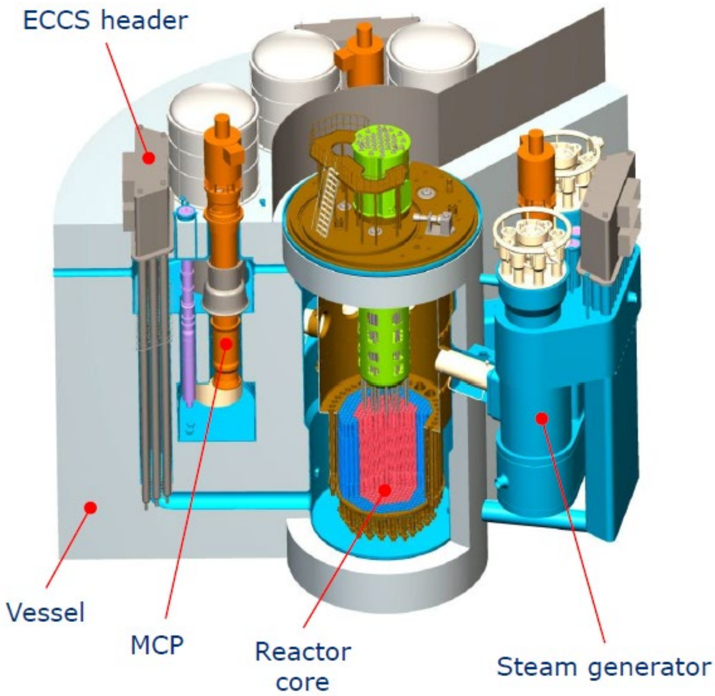

Figure 3: BREST-OD-300 reactor core and main systems [20].

The fuel type considered for the first core of the BREST-OD-300 fast reactor is nitride of depleted uranium mixed with plutonium and Minor Actinides (MA)[4], whose composition corresponds to that of irradiated (spent) fuel from VVER's following reprocessing and subsequent cooling for ~25 years.

Table 2: BREST Summary Parameters.

Power 700 MW(th)	700 MW(th)/300 MW(e)
Core diameter	2.6 m
Core height	1.1 m
Core fuel	(U-Pu)N UN + PuN
Coolant temp.	420/540°C
Maximum clad temp.	650°C
Efficiency	43-44%
Core breeding ratio (CBR)	~ 1

The characteristics of lead allow for the operation with such fuel at an equilibrium composition. This mode of operation is characterized by full reproduction of fissile nuclides in the core (Core Breeding Ratio – CBR–~1) with irradiated fuel reprocessing in the closed fuel cycle. Reprocessing is limited to the removal of fission products without separating Pu and minor actinides (MA) from the mix (U-Pu-MA). Removed fission products from the fuel are replaced by depleted uranium. There is no need to have enriched uranium for the initial core load and for further reloading in closed fuel cycle. One of the notable characteristics of the BREST system demonstrator is that a reprocessing plant happens to be co-located with the reactor, eliminating any accident or problem due to fuel transportation to other sites. While this may be an option for future commercial systems, the option of relying on a centralized (i.e., remote from the reactor site) recycle facility to serve multiple BREST reactor sites is also possible.

1.1.3 SSTAR design

The SSTAR concept was developed using proliferation resistant concepts in its design [21], [22]. The reactor has a small-size core and a very long core life of up to 30 years. The reactor

[4] The first start-up core of the BREST-OD-300 demonstrator will not contain MA. For future subsequent BREST reactors it will be possible to include MA already in the start-up core.

module is designed to be factory fabricated and shipped to the plant site. It would require relatively little action from the operators, who have no access to the fuel as the vessel is sealed and would be installed at a site lacking any refueling or reactor vessel disassembly equipment. The concept envisions the delivery of the reactor as a fully assembled sealed unit, and its removal as a sealed unit at the end of its fuel life. The vessel has a high height-to-diameter ratio, large enough to completely rely on natural circulation for primary cooling, reference [23] depicts selected features of the current reference design for the SSTAR system. Figure 4 [10] provides an overview sketch of the SSTAR reactor and Table 3 summarizes the main reactor's parameters.

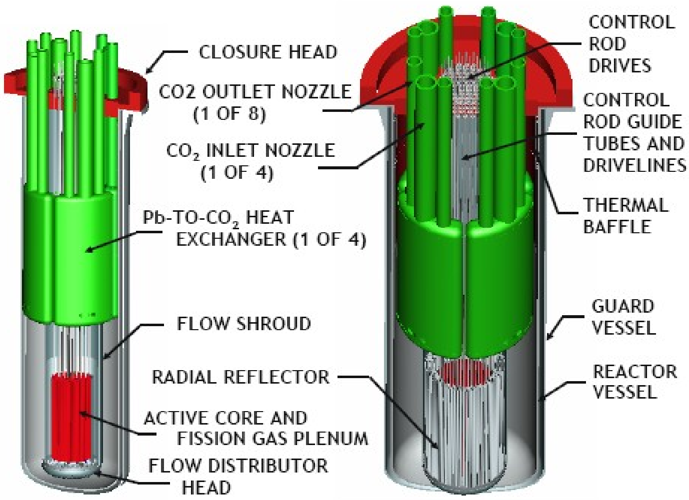

Figure 4: The Small Secure Transportable Autonomous Reactor (SSTAR) [10].

Table 3: SSTAR Summary Parameters.

Power 45 MW(th)	19.8 MW(e)
Core diameter	.976 m
Core height	1.22 m
Core fuel	TRU Nitride with ^{15}N enriched nitrogen
Coolant temp.	420/567°C
Maximum clad temp.	650°C
Efficiency	44%
Core breeding ratio (CBR)	~ 1

The SSTAR system was developed to the preliminary conceptual design stage at which time the design was frozen pending potential future funding to further advance the concept [24]. At the present time, the concept is considered to be "on the shelf." It has been included as one of the LFR-pSSC reference concepts to complete the range of plant-size options and to maintain attention to some of its features which differ from the other larger reference designs, including its intrinsic PR&PP characteristics. The pSSC committee holds open the possibility that, at a future date, this reference design could be supplanted by another concept in a similar size category to the extent that such a new system was undergoing significant advancement.

It should be noted that the genesis for the SSTAR concept was the idea of developing a reactor that was, by design, low in proliferation risk and therefore deployable virtually anywhere in the world [25]. The objectives resulting from this goal included factory fabrication (and fuelling); transportability of the reactor system to the site and installation without the requirement for handling fresh or spent fuel, or for developing any on-site fuel supply infrastructure; ultra-long core life to enable long-term operation without refuelling; and robustness and simplicity of design (e.g., reliance on natural convection flow for heat removal) to minimize operational complexity and maintenance requirements. The SSTAR pre-conceptual design is a small (19.8 MWe / 45 MWth) natural circulation fast reactor incorporating lead as the coolant and able to use transuranic nitride fuel.

The SSTAR design features a pool vessel configuration, natural circulation of the primary coolant for operational as well as decay heat removal, and a supercritical CO_2 Brayton cycle power conversion system. SSTAR is intended to be installed below grade, providing both enhanced safety and physical protection, and can meet the electricity requirements of a small town (~ 25,000 people).

1.2 Summary of the Main Characteristics of the ELFR, BREST and SSTAR

The main characteristics of the three reference systems are summarized in Table 4. Other PR relevant data are reported in Section 3.

Table 4: Main Characteristics of the ELFR, BREST-OD-300 and SSTAR Systems.

	ELFR	***BREST OD-300***	***SSTAR***
Power (MWth)	1,500	700	45
Power (MWe)	600	300	19.8
Thermal efficiency (%)	42	42	44
Primary coolant	Lead	Lead	Lead
Primary circulation power)	Forced	Forced	Natural
Core inlet temperature (°C)	400	420	420
Core outlet temperature	480	540	567
Fuel	MOX	Mixed Nitrides	Mixed Nitrides
Peak cladding temperature	550	650	650
Secondary system working	Water 18 MPa 354°C	Water 17 MPa	S-CO_2 20 MPa
Prim./Sec. heat transfer	Eight Pb-H_2O SGs	Eight Pb-H2O SGs	Four Pb-CO2 HXs
FA geometry	Wrapped Hexagonal	Open Hexagonal	Open lattice
PRPP Relevant Features	Equilibrium closed fuel cycle with full recycle of actinides	On-site closed fuel cycle with full actinide recycle	Sealed vessel; long core life without refuelling; underground siting; MA-containing fuel

2 Overview of Fuel Cycle(s)

Fuels being considered include nitride (of uranium or mixed actinides) for the BREST-OD-300 and SSTAR systems, and oxide (of mixed actinides) for the central station ELFR system. In all cases, the main fertile material is U-238. The use of thorium as a fertile component is a theoretical possibility, but this option has received little consideration so far and will not be considered in the analyses reported in the following sections. The plutonium content envisioned for these reactors lays in the range 15%-20%, defined as the ratio of the mass of fissile material vs. the total mass of heavy metal (HM): (Pu/(Pu+U)). Multiple enrichment zones including fuels with substantially lower levels are normally used in the design. All reactor concepts envision a production of fissile isotopes equal to or slightly above their consumption (conversion ratio equal to or slightly above 1.0) to enable long core life and no or infrequent refuelling. As such, breeding blankets, either radial or axial, have not been included in any core concepts of the three reference reactor designs. Reload cores could draw from recycled material from LWR spent fuel and eventually from fuel undergoing multiple-recycling. For the SSTAR reactor, the core could be fabricated from fresh enriched uranium.

The current reference designs envision fuel inventories of 44 tHM for ELFR, 20.6 tHM for BREST and 4.5 tHM for SSTAR.

The refuelling frequency for SSTAR is unique with no refuelling expected with the possible exception of the whole-core (cassette) refuelling under supplier control at the end of core life. The original design envisioned a 30-year core life, but recognized the possibility of a shorter core life of 15 years in which case whole core cassette refuelling could be considered [23].

It is expected that the duration of the fuel campaign in BREST will be of the order of 5 years with subsequent additional fuel exposure throughout the year due to in-vessel storage. Partial reloads are planned once per year during scheduled reactor shutdowns. Fuel assemblies are discharged from the core when the target fuel burnup is reached (10% heavy atoms - h.a.). Fuel assemblies with maximum fuel burnup form a group of partial reload (30-35 FAs). The internally generated minor actinides (Np+Am)[5] will be recycled at equilibrium content (~0,5% in heavy metal) and will be homogeneously transmuted in the fuel which is, in the case of the Demonstrator, to be fabricated at the co-located high security closed nuclear fuel cycle facilities.

For the ELFR concept, the in-core residence time is about 6 years with planned outages. Off-line refuelling would be required. Neutronic analyses performed for ELFR have demonstrated that it is possible to burn all the generated minor actinides with an equilibrium content of MA in the core of about 1.3% of heavy metal [19]. The recycle approach has not at this time been detailed, but homogeneous recycling technologies are envisioned for extraction of all actinides in a single stream (i.e., no separation among the species during reprocessing). It is expected that the approach would involve central recycle facilities.

Heterogeneous recycle of MA, as well as recycle of Low-Level Fission Products (LLFP), has not yet been fully investigated. The recycle technology and its attributes (e.g., recycle efficiency and waste forms) would be expected to be similar to that of the oxide-fuelled variant of the Sodium Fast Reactor (SFR).

For the ELFR, conceived as an “adiabatic reactor” - meaning that it has a conversion ratio of about 1 and burns its own MAs – the homogeneous recycle of all actinides (i.e., with no separation of U, Pu and MA streams) is foreseen, to be followed by addition of depleted uranium to compensate burnup.

[5] Cm is foreseen to be separated and not present in the re-fabricated fresh fuel.

For SSTAR, given its preliminary conceptual design stage, details of spent fuel processing have not been fully detailed. However, it is envisioned that the entire reactor (or cassette core, in the case of refuelling) would be removed from the site under supplier control (following a suitable cool-down period) and taken to a supplier-managed reactor recycle facility where both the fuel and reactor components would undergo appropriate recycling activities under international oversight.

The R&D activity on fuels included in the LFR System Research Plan (SRP) [5] is limited to aspects related to the use of fuels and cladding materials in a lead environment. With regard to the fabrication of fresh fuel, the situation of the LFR is similar to that of the SFR for MOX and LFR fuel development would benefit from this prior experience. R&D activity on Mixed Nitrides Uranium Plutonium (MNUP) for BREST (and for sodium BN-1200) are currently carried out leveraging on the possibility to irradiate fuel assemblies in SFR reactors, namely BOR-60 and BN-600 and are currently carried out according to their specific fuel development programs. Since fuel recycling and partitioning technologies are not included in the Generation-IV scope, a detailed description of closed fuel cycle options for the Generation-IV LFR concepts is not available in the frame of the GIF cooperation.

3 PR&PP Relevant System Elements and Potential Adversary Targets

This Section identifies and describes the main PR&PP-relevant System Elements and Targets of the three reference reactor designs. When fuel fabrication and recycle activities are co-located on the reactor site (as e.g., in the case of BREST-OD-300), the analysis will concentrate only on the reactor facility.

For the identification of potential adversary targets [26], the aspects that are to be looked for are the quality and quantity of nuclear material potentially usable in a military nuclear programme (material targets) and processes that might be misused to produce nuclear material suited in a military nuclear programme (process targets). The GIF PR&PP Evaluation Methodology [26] categorizes the nuclear material inside a given target "based on the degree to which its characteristics affect its utility for use in nuclear explosives". It defines HEU as "high-enriched uranium, nominally 95% U-235", WG-Pu as "weapons-grade plutonium, nominally 94% fissile Pu isotopes", RG-Pu as "reactor-grade plutonium, nominally 70% fissile Pu isotopes", DB-Pu as "deep burn plutonium, nominally 43% fissile Pu isotopes", and LEU as "low-enriched uranium, nominally 5% U-235". This characterization is used in the next sections and is different from the one used by the IAEA. [6]

3.1 ELFR System Elements *and PR Targets*

The ELFR fuel cycle is composed of three sub-systems, separately described in the following subsections:

- ELFR reactor system;
- ELFR front-end system;
- ELFR back-end system.

The front-end and the back-end part of the fuel cycle are not planned to be co-located with the nuclear reactors (i.e., ELFR postulates central/remote fuel fabrication and reprocessing). Figure 5 illustrates the ELFR reactor system elements in both single-unit layout and multi-unit layout.

As reported in Section 2, it is assumed that the first LFRs will be fuelled with Pu-based fuels, and subsequently, with depleted uranium, plutonium and MA. Fuel constituents loading is in a homogeneous configuration.

Nuclear material is only present in fuel-related items. For ELFR, the potential targets for diversion are the entire fuel assemblies (fresh and spent) or the active parts of the fuel assemblies (fresh and spent), which are comparable in size or even of larger size than those of SFRs. No dismantling activities of the active part of the ELFR fuel assemblies are foreseen on the site. A leaking pin is not replaced in the fuel element.

[6] IAEA Safeguards Glossary [38] defines the strategic value of nuclear material as a relative measure of the usefulness of a nuclear material to a potential diverter for producing nuclear explosives. The IAEA defines two categories of nuclear material. There is direct-use material nuclear material that can be used for the manufacture of nuclear explosives components without transmutation or further enrichment, such as plutonium containing less than 80% Pu-238 [39], uranium enriched 20% and higher in U-235 (highly enriched uranium (HEU)) and U-233. Chemical compounds, mixtures of direct-use materials, such as MOX and thorium and U-233 mixtures, transuranic fuels, and plutonium contained in spent nuclear fuel also fall into this category. Unirradiated direct-use (UDU) material would require less processing time and effort as opposed to the category of irradiated direct-use (IDU) material (contained in spent fuel). Indirect-use (IU) material encompasses all nuclear material except direct-use material such as natural uranium, or LEU which must be further enriched to be converted into HEU or inserted into a reactor to produce Pu-239 which can be separated in a reprocessing plant, or thorium which needs irradiation to produce U-233 which can be separated in a reprocessing plant.

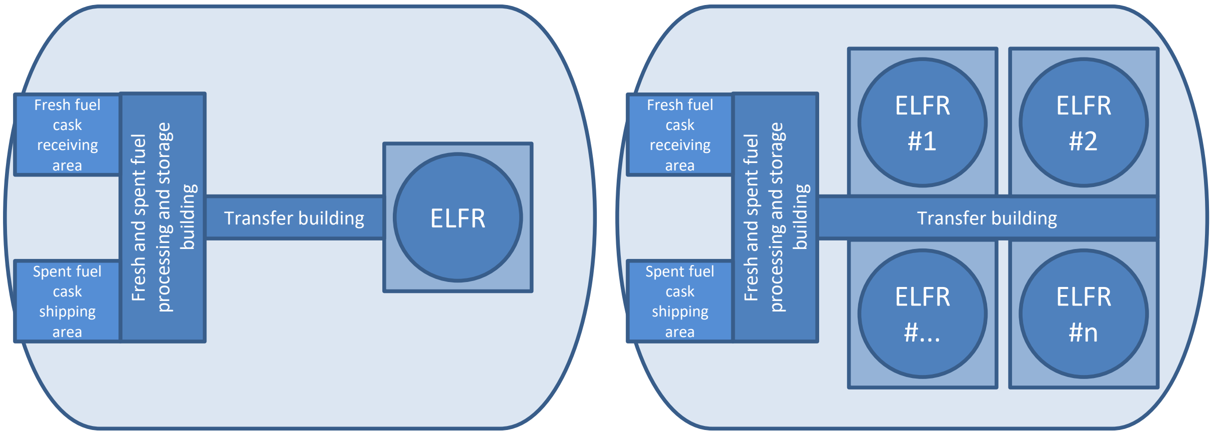

Figure 5: Diagram of ELFR reactor system elements: single-unit layout (left) or multi-unit layout (right).

The main ELFR site system elements and their related PR targets categorized in terms of nuclear material quality[7] are reported in Table 5, together with the type of proliferation strategy that the targets contained might potentially enable.

Table 5: ELFR site system elements, PR targets and related PR strategies.

System Element	***PR Target***	***NM Quality***	***Potential PR Strategy***
Fresh fuel cask receiving area	Fresh fuel	RG-Pu	Diversion
Fresh and spent fuel processing and storage facility	Fresh fuel	RG-Pu	Diversion
	Spent fuel	RG-Pu	Diversion
Reactor core	Fresh fuel	RG-Pu	Diversion[8]
	Spent fuel	RG-Pu	Diversion[8]
	Undeclared irradiation	WG-Pu	Misuse, breakout
Spent fuel cask shipping area	Spent fuel	RG-Pu	Diversion

The ELFR in-core residence time is about 6 years with planned outages every 36 months for periodic partial refuelling. Spent fuel assemblies, each enclosed within a sealed flask, are placed in interim storage for cooling inside an appropriate area in the fuel building for at least one year before introduction into transport casks for shipping to the reprocessing site. The flasks are designed to allow the continuous monitoring (temperature, radiation field) of the enclosed assemblies. The spent fuel area is sized to accept all FA of a core. A separate spent fuel storage is foreseen in case the fuel building capacity is exceeded.

Fresh fuel for ELFR can be delivered to the site fresh fuel storage with capacity at least compatible with the periodic partial refuelling needs. Figure 6 shows the fresh and spent fuel targets flows among the ELFR reactor system elements.

Being designed as an adiabatic reactor [19], the fresh and spent fuel isotopic composition is comparable except for the presence of fission products in the latter. Table 6 reports the fresh and spent fuel composition (at an average discharge burnup of approximately 61 MWd/kgHM) expected in the ELFR system elements, and Table 7 reports the fraction of even-numbered and

[7] The characterization is here performed along the lines foreseen by the GIF PR&PP Evaluation Methodology [26].
[8] Only possible through the fresh and spent fuel processing and storage facility.

odd-numbered Pu isotopes expected in the fresh and spent fuel mixtures at the equilibrium. The ELFR nuclear site does not foresee the presence of separated special fissionable material.

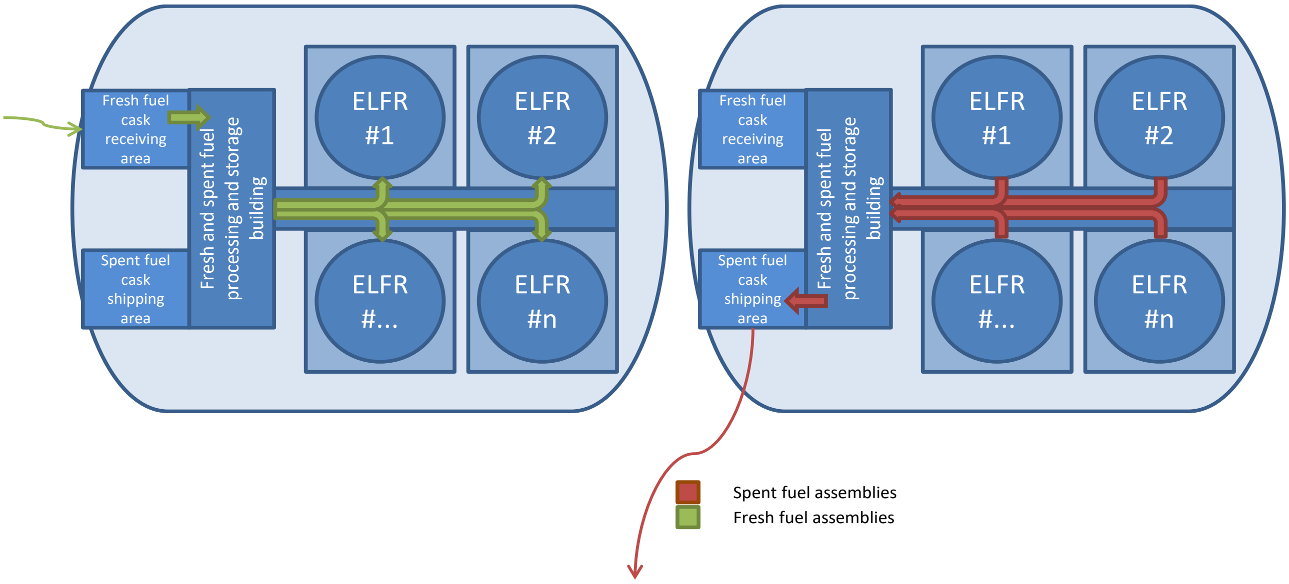

Figure 6: Fresh and spent fuel targets flows among the ELFR reactor system elements.

Table 6: Fresh and spent fuel isotopic composition at equilibrium.

Isotope	*Fresh fuel [wt.%]*	*Spent fuel [wt.%]*
U-235	0.112	0.084
U-236	0.176	0.170
U-238	79.964	73.416
Np237	0.107	0.099
Np239	0	0.011
Pu238	0.513	0.516
Pu239	9.727	9.718
Pu240	6.765	6.738
Pu241	0.518	0.745
Pu242	0.695	0.695
Am241	0.786	0.567
Am242m	0.026	0.027
Am243	0.209	0.209
Cm242	0	0.028
Cm243	0.003	0.003
Cm244	0.099	0.132
Cm245	0.033	0.033
Cm246	0.023	0.023
Cm247	0.005	0.005
Cm248	0.004	0.004
FP+other	0	6.574

Table 7: Fraction of Even-number and Odd-number Pu Isotopes in the Fuel Mixtures at equilibrium.

	Fresh Fuel (%)	*Spent Fuel (%)*
Pu-238	2.82	2.80
Pu-240	37.13	36.60
Pu-242	3.81	3.77
Even Pu/total Pu	43.8	43.2
Pu-239	53.39	52.78
Pu-241	2.84	4.05
Odd Pu/total Pu	56.2	56.8

Table 8 reports some PR-relevant characteristics of the nuclear material targets available in the ELFR system elements.

Table 8: ELFR nuclear material targets relevant characteristics.

Total fuel inventory	44 t_{HM}
No. of assemblies per core	427
Weight of fuel assembly	~ 850 kg
Pu enrichment	~ 18.3 wt.%
No. of assemblies to reach 1SQ[9] of Pu	0.4/0.5 (outer and inner core assembly, respectively). Same amount for both fresh and spent fuel assemblies
Residence time	6 y
Fuel assembly burn-up	~ 61 MWd/kg_{HM}, with peaks of ~ 104 MWd/kg_{HM}

Optimization of the overall fuel cycle strategy for large plants has yet to be completed. An alternative credible option, not yet evaluated, can be in-core residence time of 5-6 years with only one refueling at the end of the fuel cycle. The frequency of spent and fresh fuel operations is already envisioned to be 2 to 3 times longer than that of conventional nuclear power plants (NPPs), but a fast reactor has the potential to have even longer fuel cycles, and this can be addressed in the future.

For completeness, Figure 7 shows the functional system elements expected at the front-end and back-end of the ELFR nuclear fuel cycle, system elements that are not planned to be co-located with the nuclear reactor. Finally, Figure 8 illustrates the nuclear material targets that are being transferred between the front-end, the reactors and the back-end, and the related flows.

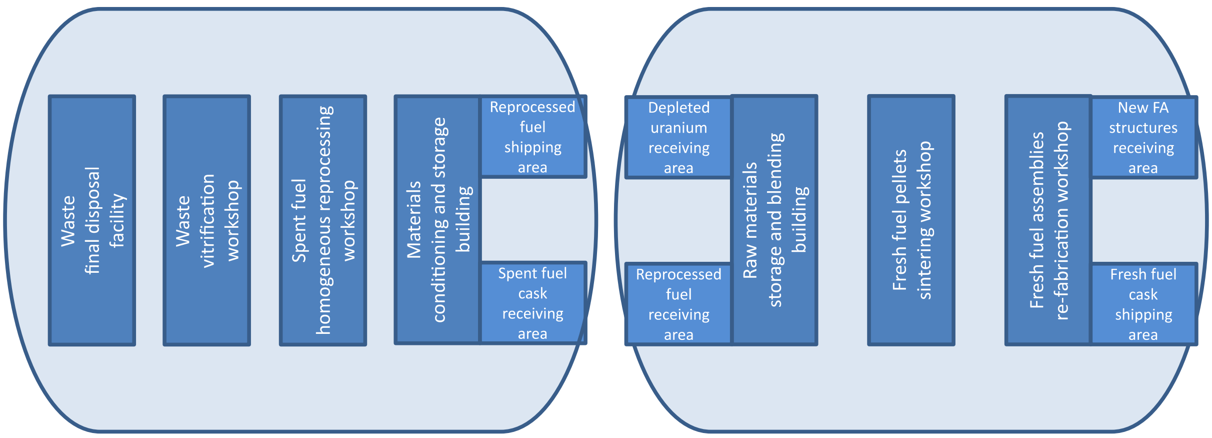

Figure 7: Diagram of ELFR back-end (left) and front-end (right) system elements.

[9] The IAEA Safeguards Glossary [38] defines Significant Quantity (SQ) as "the approximate amount of nuclear material for which the possibility of manufacturing a nuclear explosive device cannot be excluded".

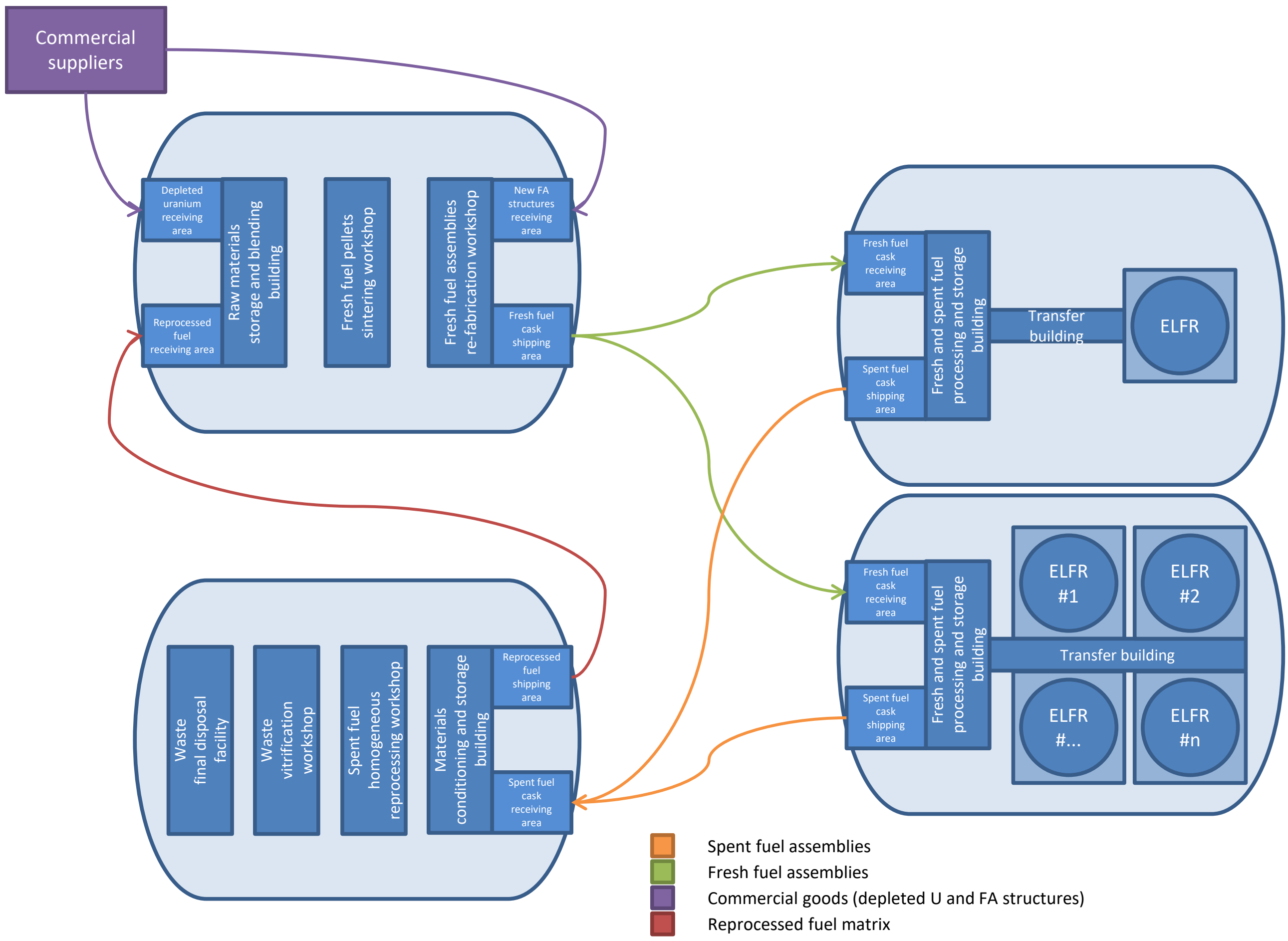

Figure 8: Diagram of the ELFR nuclear fuel cycle, with the flow of the nuclear material targets between the front-end, the reactors and the back-end.

3.2 BREST BREST-OD-300 System Elements and PR Targets

The BREST BREST-OD-300 nuclear energy system demonstrator will have a closed nuclear fuel cycle ([27], [28], [29]), in which the nuclear reactors and the fuel fabrication and recycling facilities are co-located on the same site[10]. This setup has the advantage of potentially eliminating proliferation threats or physical protection problems due to off-site fuel transportation, and concentrates the PR relevant system elements and nuclear material targets on the site. Figure 9 provides a *3D* rendering of the site, with the various fuel cycle facilities colour coded, Figure 10 indicates PR relevant system elements for the reactor facilities. Using this information, Table 9 illustrates the main PR relevant system elements and PR targets of the BREST-OD-300 reactor part of the site.

The co-location of the reactor unit, the fabrication/re-fabrication facility and the spent fuel recycling facility implies the presence at the co-located reactor/fuel cycle site of potential nuclear material targets in bulk form, usually not available on a nuclear reactor site. However, since the co-location of these facilities is not an intrinsic attribute of the BREST reactor concept, and in installations after the demonstration project such co-location may not take place, in this white paper, only the System elements and targets strictly related to the Reactor facility and its operation will be considered[10]. In any case, the BREST-OD-300 fuel cycle does not foresee the presence of separated special fissionable material at any stage.

[10] As previously mentioned, in this document the co-located fuel cycle steps will not be subject of a PR&PP analysis.

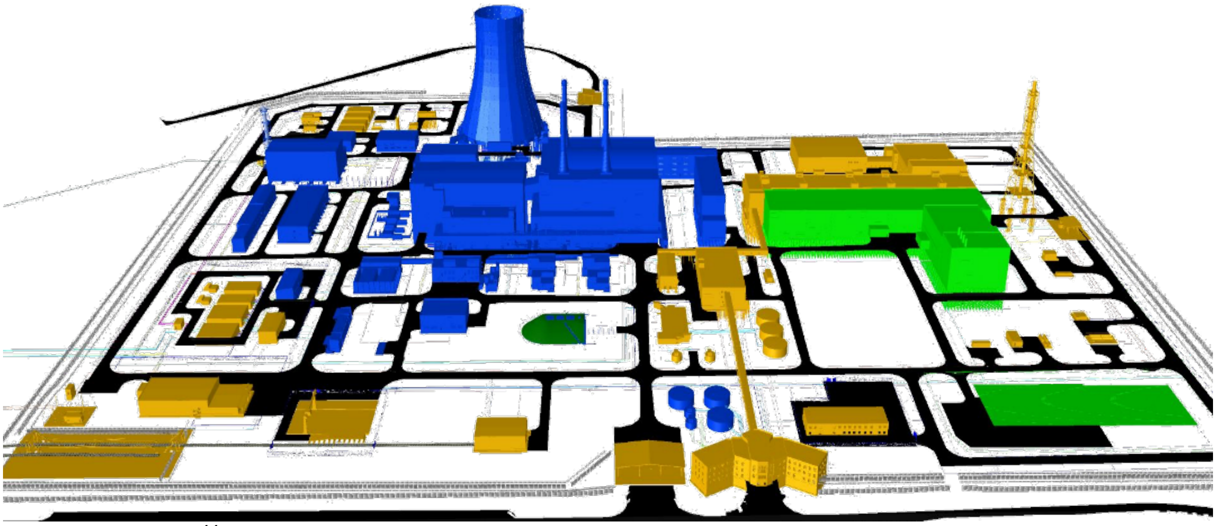

Figure 9: PDEC[11] general layout: yellow – fabrication/re-fabrication facility, blue – BREST-OD-300 reactor unit, green – SF recycling and RAW handling facility.[12]

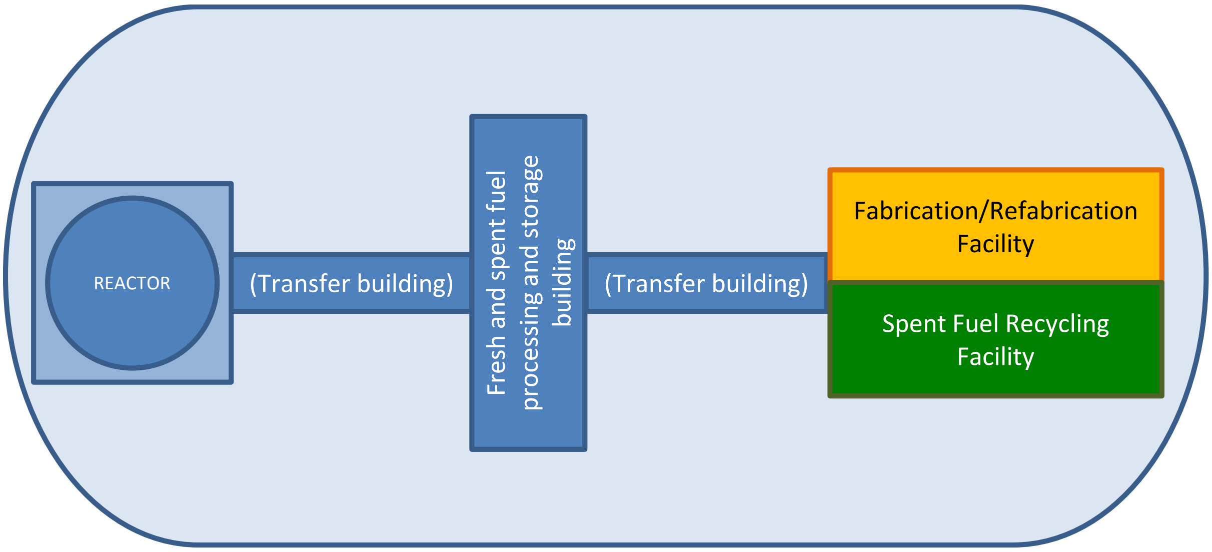

Figure 10: PDEC site PR relevant system elements.

[11] Pilot and Demonstration Energy Complex.
[12] Reproduced from Alemberti, A., and Tucek, K., "Report on the Status of GIF Lead-cooled Fast Reactor", 45th Policy Group Meeting, Sun Valley Lodge, Idaho, USA, 17-18 May 2018. Original image source: [37].

Table 9: BREST BREST-OD-300 site system elements, PR targets and related PR strategies.

System Element	***PR Target***	***NM Quality***	***Potential PR Strategy***
Reactor core	Fresh fuel	RG-Pu	Diversion
	Spent fuel	RG-Pu	Diversion
	Undeclared irradiation	WG-Pu	Misuse, breakout
Fresh and spent fuel transfer and storage building	Fresh fuel	RG-Pu	Diversion
	Spent fuel	RG-Pu	Diversion

Table 10 reports some PR-relevant characteristics of the BREST OD-300 reactor system element.

Table 10: BREST OD-300 reactor system element relevant characteristics.

Total fuel inventory	19.7 t_{HM}
No. of assemblies per core	169
Weight of fuel assembly	< 260 kg
Pu enrichment	< 15 wt.% (12-15 %) [13]
No. of assemblies to reach 1SQ of Pu	~ 0.4
Residence time	5 y
Fuel assembly burn-up	up to 10 % h.a.

3.3 SSTAR System Elements and PR Targets

Table 11 illustrates the main PR relevant system elements and PR targets of the SSTAR reactor.

Table 11: Main PR relevant system elements and PR targets of the SSTAR reactor.

System Element	***PR Target***	***NM Quality***	***Potential PR Strategy***
Reactor core	Fresh fuel	RG-Pu	Diversion[14]

SSTAR fuel will not normally be accessible outside of the reactor; and with either the highly infrequent full-core cassette replacement conducted by the reactor supplier or the sealed reactor replacement with spent fuel remaining in the sealed vessel, the hypothetical target for diversion would be the entire core or the entire reactor, both implausible targets for concealed actions. Similar to ELFR, no dismantlement or fuel handling activities are anticipated at the reactor site, and furthermore the specialized equipment and trained staff required for refuelling would be retained by the reactor supplier organization and would not be present at the reactor location during normal operations.

In SSTAR the fuel pins are permanently attached to an underlying support plate. This configuration restricts access to fuel and eliminates fuel assembly blockage accident initiators. The compact active core is removed as a single cassette during refuelling and replaced by a fresh core. Fresh or spent fuel storage is not envisioned as part of the normal operations, and full cassette core replacement would take place only at end of core life (15-30 years) and would be carried out by the reactor supplier.

Because SSTAR is in the preliminary conceptual design stage, a plant layout is not presently available. However, due to the sealed nature of the reactor system and the lack of ancillary equipment for refuelling or otherwise accessing the active core, other parts of the plan layout

[13] Depending on the isotopics of the plutonium used to produce the re-fabricated fresh fuel assemblies (in particular the amount of ^{239}Pu), the enrichment of the fuel assemblies may vary from 12% to 15%.
[14] Given the design of the reactor (no refuelling, single sealed cassette core), the only diversion strategy would be the diversion of the entire reactor core.

(besides the primary system itself) should not be considered as potential PR targets.

Regarding the materials present in the primary system, Table 12 provides the key characteristics.

Table 12: SSTAR nuclear material relevant characteristics[15].

Total fuel inventory	4.4 tHM including 750 kg fissile TRU (ca. 94 SQ of Pu[16])
Fuel fissile enrichment	1.7/3.5/17.2/19.0/20.7 TRU/HM, 5 Radial Zones
Residence time	30 y
Fuel assembly burn-up	~ 81 MWd/kg_{HM} average with peak of ~ 131 MWd/kg_{HM}

3.4 System Elements and PR Target Summary

Table 13 provides an overview of the main characteristics of the three reference reactor systems' fuel assemblies.

Table 13: Main characteristics of the ELFR, BREST-OD-300 and SSTAR fuel assemblies.

	ELFR	***BREST-OD-300***	***SSTAR***
No. Of FA per core	427	169	1
Weight of FA	~ 850 kg	< 260 kg	4421 kg
Pu enrichment	~ 18.3 wt.%	< 15 wt.%	1.7/3.5/17.2/19.0/20.7 TRU/HM, 5 Radial Zones
No of FA to reach 1 SQ of Pu	0.4/0.5 (outer and inner core assembly, respectively). Same amount for both fresh and spent fuel assemblies	~ 0.4	~.01[17]
Residence Time	6 y	5 y	15-30 y
FA burn-up	~ 61 MWd/kgHM, with peaks of ~ 104 MWd/kgHM	up to 10 % h.a.	~ 81 MWd/kgHM average, with peaks of ~ 131 MWd/kgHM

3.5 Safeguards Considerations

The LFR reactor technology would require a dedicated safeguards approach that might have several points in common with the one already developed for the SFR reactor technology. A possible example of a safeguards system for SFR that might be suited for this study is the one prepared for the GIF Example Sodium Fast Reactor (ESFR) used for the GIF PRPP Case study[18] [30] and reported in Figure 11.

[15] Given the design of the reactor (no refuelling, single sealed cassette core), the number of assemblies to divert for obtaining 1 SQ of special fissile material is not reported in the Table.
[16] Being the core a single sealed cassette core, a potential proliferator would have to divert the entire nuclear material inventory.
[17] Based on 750kg of TRU (assume Pu) in whole core assembly, and 8kg Significant Quantity value for Pu.
[18] As a reference, a facility in a State under a Comprehensive Safeguards Agreement [39] has been postulated.

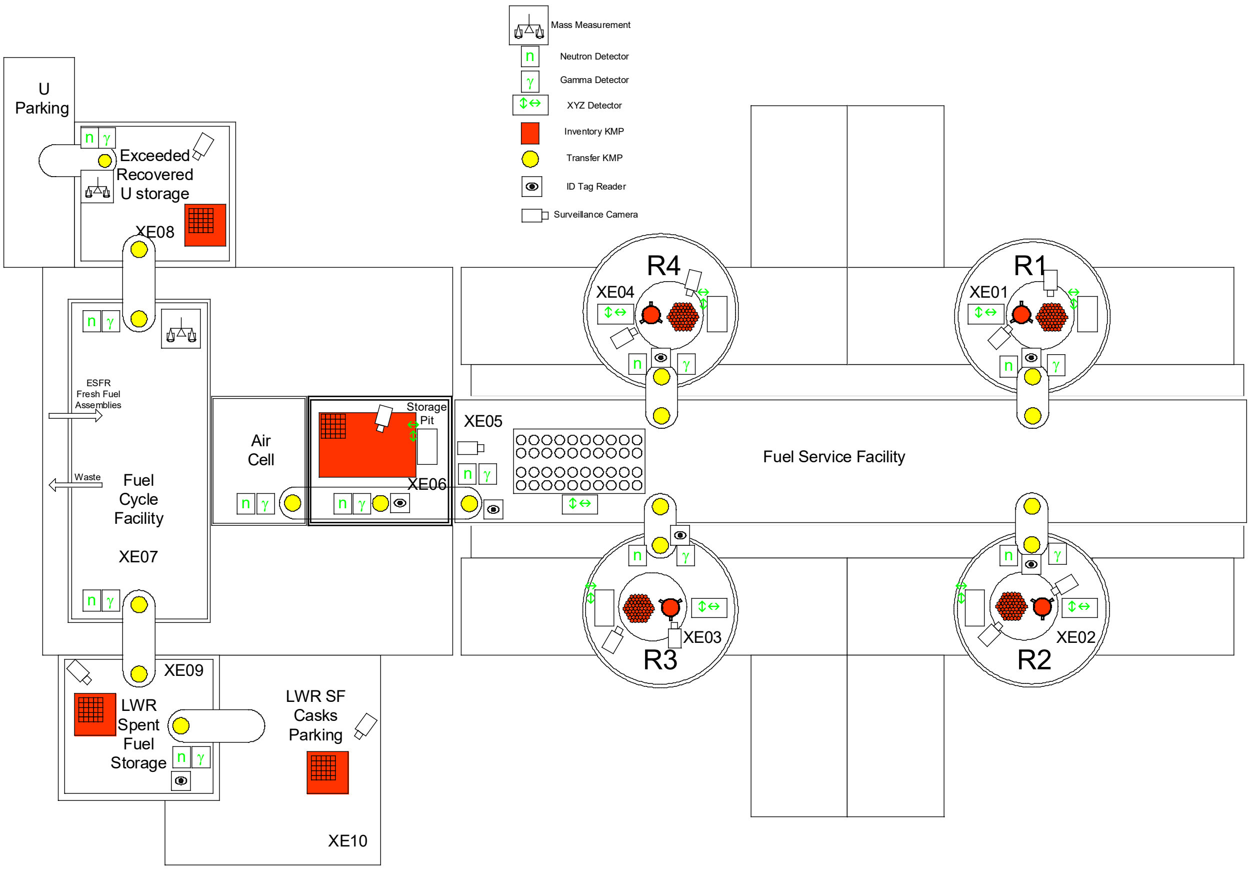

Figure 11: Safeguards system developed for the Example Sodium Fast Reactor (ESFR), object of the GIF PRPP Case Study [30]. The site co-locates 4 nuclear power units, a recycling facility and related fresh and spent fuel storage.

3.5.1 ELFR

For this study, it is assumed that a safeguards approach developed for a SFR, with appropriate adjustments, would be able to meet safeguards objectives for the ELFR reactor site.

Different technical options are under investigation in order to design the monolithic active part of the ELFR fuel assemblies with built-in features for identification (numbering, etc.). In the reactor, the upper part of the fuel assemblies extends above the level of the coolant and can be continuously monitored by cameras, a relevant feature for implementation of safeguards.

ELFR design features facilitating the application of containment & surveillance (C/S) measures are as follows [31]: a high level of automation; remote handling of both fresh and spent fuel; and standardization of items in transfer in the facility (i.e., an entire fuel assembly or its active part). Surveillance is facilitated by the possibility of visual inspection of the FA inside the four system elements. Figure 5 shows three storage/transfer area operations and the reactor, which additionally includes monitoring of the FA inside the core during operation. Monitoring of neighbouring areas is standard safeguards practice.

3.5.2 BREST OD-300

As for ELFR, it is assumed that the safeguards approaches implemented for SFR systems would be able to be effectively applied to the BREST OD-300 nuclear reactor, as well as to the fresh and spent fuel transfer and storage system elements.

A key element of ongoing consideration relates to the specific additional safeguards activities that will be implemented as a result of the co-located fuel processing and fabrication facilities. The details of the fabrication/refabrication facilities and processes are outside the scope of this white paper since the co-location of these facilities on the reactor site is not an intrinsic requirement and these processes could as easily take place outside the plant boundary. Nevertheless, it is noted that the standards applied to existing safeguarded fuel fabrication facilities would be implemented, and safeguards-relevant aspects of such existing facilities will be drawn on in the development of the approach for BREST.

It is further noted that the handling of nitride fuel must take place in an inert atmosphere, and the fuel cycle facility is inaccessible to personnel due to the very high radioactivity of spent and reprocessed fuel. All the fuel cycle processes and performed operations are fully automated.

3.5.3 SSTAR

During normal operation the SSTAR site does not provide any accountancy issue, as no movement of nuclear material is foreseen for the entire duration of the core life. Most likely the safeguards objectives would be met by a rather straightforward containment & surveillance (C/S) able to provide an adequate level of reliability (e.g., a dual C/S system on the system's core, using seals and/or surveillance cameras).

A potential significant safeguardability challenge for the SSTAR site could arise from the highly unlikely event of a nuclear material inventory re-verification needed due to a loss of continuity of knowledge rendering the nuclear material inaccessible to the inspectors. In this situation, unlikely as it may be, it is unclear how an onsite reverification could be carried out. This remains an issue for future consideration and a key issue with transportable reactor types.

3.6 Physical Protection Systems Considerations

In principle, ELFR and BREST OD-300 should not have major departures from the physical protection considerations of existing thermal and fast commercial reactors.

Additionally, SSTAR does not present any reasonable theft targets, and its Physical Protection Systems (PPSs) can be considered at least on par with existing thermal and fast commercial reactors. Its underground siting and sealed core configuration contribute positively to PP. The layout and physical characteristics of its supercritical CO_2 power conversion system have not been analysed from a PP perspective, but a comparative analysis with the more common Rankine steam cycle systems should be carried out in the future.

In view of the present efforts to take into consideration SMR in the development of GIF activities and the interest shown by many industrial stakeholders, specific considerations for such designs should be made for Physical Protection in view of the number of installations and related issues.

4 Proliferation Resistance Considerations Incorporated into Design

The following considerations related to LFRs are based on the hypothesis of a central reprocessing and fuel fabrication plant physically separated from the reactor for both ELFR and SSTAR, closed fuel cycle for BREST OD-300 with fuel fabrication and recycle facilities that could be either co-located or remote from the reactor site.
A significant difference between SSTAR on the one hand and ELFR and BREST on the other is that SSTAR foresees the supply and replacement of the entire core, whereas ELFR and BREST foresee quite standard operational practices with periodic access to the core for fuel handling and partial replacement of the core.

SSTAR and BREST fuel cycles are based on nitride fuels, whereas ELFR foresees MOX fuel (nitride fuel is considered only as prospective option) and associated processing technologies similar to those of a large SFR. All three reference systems, at least in principle, could accept MA-bearing fresh fuel.

4.1 Concealed diversion or undeclared production of material

Concealed diversion of material from the reactor site can be deterred and detected by the application of international safeguards. Material balance areas are not yet defined or not openly available. Accountancy is limited to materials in item form for both ELFR and SSTAR, and also for the BREST reactor system. Bulk form accountancy for BREST would be introduced if the potentially co-located (as an option) fuel cycle operations were to be considered within the system boundary.

Undeclared production of nuclear material usually has two main possible goals: the production of nuclear material that is more attractive than normally available, or the production of a greater quantity of already-available nuclear material. The long-lasting sealed core attribute of SSTAR is likely a positive factor in averting concealed or undeclared production of material.

4.1.1 ELFR

The ELFR fresh and spent fuel assemblies contain more than 1 SQ of plutonium per item, making the diversion of just one fuel assembly enough for the acquisition of the special fissionable material needed for one nuclear explosive device. The plutonium vector of both fresh and spent fuel assemblies is roughly comparable and characterized as reactor-grade plutonium – i.e., not the best choice for a potential nuclear proliferator, but still considered to be (theoretically) weapons-usable. Fresh fuel elements would be too highly radioactive to be handled without proper radiation shielding and other protective measures due to the presence of plutonium and MA; the radiation barrier would nonetheless be substantially lower than that provided by the presence of fission products in the spent fuel elements, making fresh fuel assemblies the more accessible nuclear material diversion target in item form containing weapons-usable material [32]. Being an off-load reactor, the diversion of fresh/spent fuel assemblies from the ELFR reactor core would only be possible during refuelling activities, and since there is no direct exit from the reactor core area, the elements would have to leave the site through the fresh and spent fuel processing and storage facility. Nuclear safeguards C/S systems would most likely be able to detect any abnormal movement of fuel assemblies in the reactor.

Transport of fresh and spent fuel to/from ELFR will include procedures similar to those of other reactors, and similar monitoring and surveillance controls are envisioned. The design of ELFR incorporates the use of standard FAs, and this will facilitate monitoring and material balance activities

The fresh and spent fuel items available in the site's storage facilities (fresh and spent fuel assemblies at the Fresh and Spent Fuel Processing and Storage Facility; fresh fuel assemblies at the Fresh Fuel Cask Receiving Area; and spent fuel assemblies at the Spent Fuel Cask Shipping Area) could be targets of a diversion strategy, but such action would be most likely detected by the safeguards C/S system in place and/or revealed by standard inspection activities. While the radiation barrier associated with both fresh fuel and spent fuel assemblies complicates their transport, fresh fuel assemblies are routinely moved and transported onsite. With respect to spent fuel assemblies, their diversion should require a long cooling period before being moved outside the Fresh and Spent Fuel Processing and Storage Facility. Spent fuel assemblies in the Spent Fuel Cask Shipping Area are already transportable and would not pose particular issues apart from the very easy detection by the C/S systems.

There are limited number of possible routes for concealed undeclared production inside the ELFR core; among them:

1) The substitution of a fuel element with a target uranium assembly (depleted, natural or enriched – in this case providing a similar contribution to the reactor's reactivity and the sustained chain reaction);
2) The insertion of an ad hoc Depleted Uranium (DU) reflector target replacing an existing reflector assembly[19];
3) The irradiation of a suitable number of fresh fuel assemblies where a small number of pins were replaced by pure uranium targets (either depleted, natural or enriched, depending on the level of sophistication of the strategy).

Since the nuclear material inventory onsite is already large compared to the quantities theoretically needed for a military weapons programme, all three options would likely have the objective of producing nuclear material with a more attractive isotopic composition than those already existing[20].

Option 1 would require both the diversion of a declared fuel assembly from the core inventory and the concealed insertion of the target assembly (i.e., introduction on the site, transport to the reactor core, insertion in a core position). Such activity would hardly escape detection from the standard safeguards C/S systems. The thermal hydraulic design features of the core would signal a dummy fuel assembly filled with a fertile material instead of a fuel assembly. Indeed, the significant variation of the power generated from such an assembly between the beginning and the end of its irradiation would result in relevant changes of the outlet temperature, which is continuously monitored. Should the insertion of such an assembly be realized without temperature monitoring provisions, this would result in an alert signal registered by the control system. Although not a current common practice, signals coming from in-core instrumentation could be made available to inspectors to detect anomalies resulting from design and operations modifications.

Option 2 would require the production and insertion of an assembly which has a completely different geometry from a normal fuel assembly. Since reflector assemblies are not normally moved around, such an activity would be difficult to conceal and would raise follow-up questions and actions by the inspectors reviewing the C/S data. Consequently, this option will not be further elaborated.

Option 3 is the most complicated strategy but also the most difficult to be detected at the ELFR reactor site. It would imply the substitution of a small number of original pins in the central fuel fabrication facility, the irradiation of the fuel assemblies in the ELFR reactor, the shipment of the assemblies to the central reprocessing facility and the subsequent diversion of the irradiated target pins a the reprocessing facility. At the ELFR nuclear reactor site the modified assemblies would

[19] It is worth recalling that no blanket is foreseen in the ELFR design.

[20] As stated before in the document, the ELFR site does not foresee the presence of separated weapons-usable special fisssionable material, and in general the available nuclear material composition is far from optimal for usage in a nuclear weapons programme.

follow a normal process flow and would likely leave the site without generating any observable anomaly. The time to accomplish this proliferation step might be quite long due to the long residence time of the assemblies onsite (i.e., irradiation time and cooling time) and would make this scenario unlikely. Any effort to reduce the assemblies' residence time (e.g., unloading of the assemblies after a single shift and early shipment of the assemblies back to the reprocessing facility) might be accomplished by claiming, for example, that a damaged assembly required early removal. Although there are means under investigation to detect missing pins in a fuel assembly, the detection of a very small number of missing pins in a fuel assembly is currently challenging. On the other hand, the substitution of even few pins at the (offsite) central fabrication facility and the diversion of the irradiated target pins at the (offsite) central reprocessing facility would very likely be detected by routine inspection verification activities. It should be noted however that, in order to achieve one SQ in the 6 years of in-pile residence time, about 60% of the pins in the fuel assembly located in the position of highest flux would need to be replaced by pure uranium targets. This condition would most likely fall under Option 1, as the large discrepancy in the power produced by such a modified assembly would be detectable by in-core instrumentation. Conversely, and since already 10% variations in the coolant outlet temperatures would be registered as anomalies by the reactor control system, for the replacement of fuel pins with uranium targets to not be detectable, more than 6 fuel assemblies would need to be involved in this option for diversion strategy, easing the C/S task.

Finally, it is worth mentioning that, whatever the option, the concealed irradiation of depleted uranium targets in the ELFR, following a standard cycle, due to the very long in-pile residence, would generate plutonium containing approximatively 67% of Pu-239 and about 28% of Pu-240, falling well into the PRPPEM definition of reactor grade plutonium [26].

4.1.2 BREST OD-300

As for ELFR, the BREST OD-300 fresh and spent fuel assemblies contain more than 1 SQ of plutonium (reactor-grade Pu and therefore far from being a preferred choice for a military programme) per item, making the diversion of just one fuel assembly enough for the acquisition of the special fissionable material needed for one nuclear explosive device. The considerations related to the diversion of fresh or spent fuel assemblies from the ELFR reactor core are valid also for the BREST OD-300 reactor core and building.

The fresh and spent fuel items available in the reactor and related fuel storage facilities (i.e., fresh and spent fuel assemblies at the Fresh and Spent Fuel Processing Facility) could be targets of a diversion strategy, and the considerations identified for the ELFR targets are similar also for the BREST OD-300 assemblies.

Undeclared production of nuclear material in a complex site like the one postulated for the BREST OD-300 might enable several theoretically possible proliferation actions, even though most of them would either be unattractive or easily detectable. As for ELFR, the potential undeclared production strategies considered in this paragraph are targeted at obtaining a more attractive isotopic mix than is normally available in nuclear material produced in nominal commercial nuclear fuel cycle operations[21].

For the undeclared production of special fissionable material in the reactor core, options 1 and 3 considered for the ELFR are potentially carried out also in the BREST OD-300 nuclear reactor, with comparable difficulties and low detection likelihood. Not having a steel reflector, option 2 is probably not a viable option. The BREST reactor core operates with a very small reactivity margin

[21] Similarly to the ELFR site, the BREST OD-300 site does not foresee the presence of separated weapons usable special fissile material, and in general the available nuclear material composition is far from optimal for usage in a nuclear weapons programme.

in the power range of 30-100% ($\Delta\rho$~ 0,65 β_{eff} with a core reproduction coefficient (no blanket) ~ 1). The replacement of several FAs (or radial lead reflector blocks) by target materials will decrease the core reactivity below the level of criticality, making reactor operation at power impossible. Given this limitation, the only viable scenario would seem to be the replacement of just one fuel assembly with a depleted uranium target. Consequently, this strategy would lead to a fairly long proliferation time, here defined as the time to obtain one SQ of Pu.

4.1.3 SSTAR

The SSTAR concept entails a sealed primary system with a very long fuel lifetime to virtually eliminate the possibility of diversion or misuse during the 15- to 30-year operational life of the reactor core. The sealed nature of the primary system is intended to prevent the possibility of access by the operator, and the absence on the site or equipment needed to access the core (as in refuelling equipment) further assures that there is essentially no possibility of concealed action for diversion or misuse. At the end of the core life, two possibilities are envisioned: (1) that the entire reactor would be shut down and removed from the site by the supplier without any unsealing of the primary system; and (2) as an alternative possibility, that the supplier would refuel the reactor by replacing the whole core cassette assembly. In the former case, the reactor with intact core would be removed to a centralized supplier location where dismantlement and recycle activities would take place under appropriate oversight and control. In the latter case, the on-site cassette refuelling operation would be conducted by the supplier under appropriate oversight and control using special equipment brought from off-site to complete this operation. The whole core cassette would then be removed from the site and transported to a centralized supplier location where spent core dismantlement and recycle activities would take place. The potential diversion targets would therefore be either the entire reactor or the whole cassette core, and diversion of either of these items cannot go undetected by safeguards measures put in place by safeguards inspectorate.

On the basis on the system's available information, the SSTAR design does not lend itself to any plausible undeclared production scenario.

4.2 Breakout

Breakout scenarios imply the decision to pursue a nuclear weapons programme without the boundary condition of keeping it concealed. In such scenarios institutional (extrinsic) barriers such as nuclear safeguards are ineffective, and the only barriers to proliferation are the intrinsic PR features of the available nuclear fuel cycle. Usually, a breakout scenario has the objective of minimizing proliferation time, while aiming at the best possible special fissionable material isotopic composition.

4.2.1 ELFR

In a global system architecture, the ELFR reactor would be deployable in full fuel cycle states as well in reactor states.

In a breakout scenario aimed at diverting already-available nuclear material, the ELFR site would guarantee the availability of a considerable amount of SQs of theoretically weapons-usable nuclear material (although far from optimal composition) readily available for diversion, with a preference for fresh fuel assemblies.

In a breakout scenario aimed at producing weapons grade nuclear material, depleted uranium fertile targets could be inserted either in the reactor core or in the steel reflector (analogous to the undeclared production scenario 1, except from the need of remaining undetected). Even in a breakout situation the above-mentioned thermal hydraulic/design constraints would require the dispersal of fertile pins among several fuel assemblies. Such a scenario would be compatible with the safe operation of the reactor, upon acceptance of lower overall power and lower thermal efficiency.

4.2.2 BREST OD-300

As in the case of the ELFR nuclear reactor site, the BREST OD-300 nuclear site would host large amounts of far from optimal but theoretically weapons usable nuclear material readily available for diversion. The BREST OD-300 demonstrator nuclear site offers the necessary infrastructure to operate a closed fuel cycle, but further considerations on these scenarios are out of the scope of this paper.

In a scenario aimed at producing weapons-grade plutonium, the potential proliferation would have to find a way to go around the BREST-OD-300 core's design feature of having a very small reactivity margin to be able to insert more than one uranium target assembly per irradiation cycle.

4.2.3 SSTAR

In the case of SSTAR, a breakout scenario would seem plausible principally at the beginning of the fuel cycle, as the long lasting core and related high burnup of the in-core isotopic mixture toward the middle and end period of the irradiation cycle would make the nuclear material available onsite a very unattractive target, to be considered only in cases where no other potential nuclear material target or nuclear material production capability exists in a state.

4.3 Production in clandestine facilities

None of the considered reactors and related co-located processes are directly suited for clandestine replication in a nuclear weapons programme. Nonetheless, the operation of a nuclear energy system usually requires the acquisition of a number of skills, competences, technological know-how and operations experience that might play a role in the development of a clandestine nuclear programme.

4.3.1 ELFR and BREST-OD-300

Simpler thermal reactor facilities could be easier to adopt for dedicated clandestine production of Pu, and it does not seem plausible to consider a direct replication of the ELFR or the BREST-OD-300 design on a clandestine site.

The operation of a commercial nuclear power reactor implies the availability of a complex structure providing a wide range of skills, competences and technological know-how, indispensable to ensure the safe and efficient operation of the power plant (see, e.g., [33]). The needed structure includes several national organizations and authorities with competences in a wide range of domain areas, a technical industry able to support the life cycle of the nuclear facility, and highly trained human resources (*i.e.,* engineers, physicists, radioprotection experts, highly specialized workers) that might form the basis for a transfer of knowledge and expertise to a clandestine military nuclear programme.

4.3.2 SSTAR

The peculiar design of the SSTAR nuclear reactor makes it more similar to a "nuclear battery" than to a conventional nuclear power reactor. Since all the nuclear-related technologies, processes and materials are out of any planned or likely operator's action, the operation of the SSTAR nuclear reactor does not (at least in principle) require the depth of particular skills and competences in the nuclear engineering/physics/chemistry domain that are required for other thermal or fast reactor system concepts.

5 Physical Protection Considerations Incorporated into Design

This section analyses the main physical protection aspects related to the three reference designs. While section 4 concentrated on nuclear proliferation actions in which the potential proliferator is the State and the entity enforcing the institutional extrinsic safeguards measures is a supranational nuclear safeguards inspectorate (typically the IAEA), in this section the theft of nuclear material or a sabotage attack is postulated to be carried out by a sub-national group. Generally the ultimate responsibility in terms of nuclear security lies with the States, which is called upon ensuring that an appropriate level of nuclear security is ensured in every nuclear site.

Both the ELFR and BREST NPPs will likely be hosted on sites of comparable size as that of existing Gen II and III power plants. To protect such sites and related facilities, the implementation of physical protection systems (PPSs) similar to those of existing reactors is foreseen. In the case of SSTAR, a substantially smaller site footprint is envisioned with the possibility of partial undergrounding. However, for all of these concepts, complete PPS designs have yet to be fully developed. The next sections provide a high-level, qualitative overview of those elements of the LFR system design that create potential benefits or issues for potential sub-national threats.

5.1 Theft of material for nuclear explosives

Potential targets for diversion of fissile materials have been identified in Sections 3 and 4 for state actors[22] and apply to theft by sub-national actors as well. The ELFR and SSTAR sites and the reactor portion of the BREST OD-300 site would host only item material. In terms of potential attractiveness of the nuclear material available for potential theft, the following paragraphs will categorize the potential theft targets according to the guidelines of IAEA INFCIRC/225 [34]. IAEA INFCIRC/225 provides guidelines for the physical protection of nuclear material, including its physical protection during transport, and of nuclear facilities against malicious acts" ([34], 1.12). In particular, it considers three types of risks: "unauthorized removal of nuclear material] with the intent to construct a nuclear explosive device; [...] unauthorized removal which could lead to subsequent dispersal; [...] sabotage" ([34], 1.14). Category I material is considered to be unirradiated plutonium (2kg or more), unirradiated uranium (U-235 isotope) enriched above natural (5kg or more), unirradiated uranium (U-233 isotope) (2kg or more). Category II material is considered to be unirradiated plutonium (more than 0.5kg but less than 2kg), unirradiated U-235 (enriched to 20% or more) (more than 1kg but less than 5kg), unirradiated U-235 (enriched between 10% and 20%) (10kg or more), U-233 (more than 0.5kg but less than 2kg), irradiated fuel (depleted or natural uranium, thorium or low enriched fuel - less than 10% fissile content) (see [34], Table 1). For any irradiated fuel that was category I before irradiation, the operator might decide to keep the category I classification even after irradiation. Among the various differences between category I and II nuclear material, there is the obligation for category I material to be kept in an inner area inside a protected area (the latter is enough for category II material). Inside the inner area, the nuclear material should be kept in a nuclear materials storage vault.

5.1.1 ELFR

Table 14 provides an overview of the ELFR site potential theft targets and their classification according to Table 1 of INFCIRC/225/Rev.5 [34]

[22] The diversion paths in Sections 3 and 4 refer to scenarios where the State is the diverter and are framed in a nuclear safeguards context. The malevolent actor in a security scenario is not the State but a sub-national group.

Table 14: ELFR site potential theft targets and their classification according to [34].

System Element	***Theft Target***	***NM Categorization*** [34]
Fresh fuel cask receiving area	Fresh fuel	Category I
Fresh and spent fuel processing and storage facility	Fresh fuel	Category I
	Spent fuel	Category II
Reactor core	Fresh fuel	Category I
	Spent fuel	Category II
Spent fuel cask shipping area	Spent fuel	Category II

The ELFR fuel assemblies can be handled only with the availability of dedicated specialized plant equipment and require a high level of operator skill and training. All operations are performed remotely because of the high radiation level around the fuel elements handled – within transfer flasks for passive cooling – in gas environment. Moreover, no equipment is available on site for disassembling the active part of the fuel assemblies. As far as MA-bearing fuel, the radioactivity level is so high to require remote handling using methods and locations that create a substantial barrier for access by non-state actors.

5.1.2 BREST OD-300

Table 15 provides an overview of the BREST OD-300 site potential theft targets and their classification according to Table 1 of INFCIRC/225/Rev.5 [34].

Table 15: BREST OD-300 site potential theft targets and their classification according to [34].

System Element	***Theft Target***	***NM Categorization*** [34]
Reactor core	Fresh fuel	Category II
	Spent fuel	Category II
Fresh and spent fuel processing and storage building	Fresh fuel	Category II
	Spent fuel	Category II

The considerations made for the ELFR item targets are valid also for BREST OD-300.

5.1.3 SSTAR

In the case of SSTAR, by design there would be no access to fresh or spent fuel during refuelling operations since the plant operates without refuelling for extended periods of time (15-30 years). Refuelling operations at the end of core life would be conducted by the reactor supplier, and the refuelling approach involves the removal and replacement of the complete core as a cassette unit. Table 16 provides an overview of the SSTAR site potential theft targets and their classification according to Table 1 of INFCIRC/225/Rev.5 [34]

Table 16: SSTAR site potential theft targets and their classification according to [34].

System Element	***Theft Target***	***NM Categorization*** [34]
Reactor sealed primary system or cassette core	Fresh reactor/cassette core	Category I
	Spent reactor/cassette core	Category II

5.2 Radiological sabotage[23]

For SSTAR, the reactor itself is the system element of concern. Key protective features include (1)

[23] This section did not see relevant changes since the previous White Paper [4], as SSTAR and ELFR did not see advancements impacting on this section, and BREST-OD-300 is foreseen to be built in the near future and therefore left out from the analysis for sensitivity reasons.

the sealed nature of the reactor vessel, (2) the absence of stored fuel or other radioactive materials, and (3) the potential for sub-grade or underground siting.

For ELFR, a sabotage incident yielding potential radiological consequences would imply a direct attack on the following system elements:

- Fresh fuel storage area;
- Reactor;
- Spent fuel storage area at fuel building;
- The Independent Spent Fuel Storage;
- Fuel shipping neighbouring areas (during arrival of fresh fuel and dispatching of spent fuel).

For BREST OD-300, a similar set of system elements can be identified:

- Fresh fuel transfer and storage facilities;
- Reactor;
- Spent fuel transfer and storage facilities.

All these system elements need to be protected from *direct attack*. The reactor containment building will be designed on the basis of the experience on SFRs and LWRs, as well as guidance from other guidance documents such as the European Utility Requirements document for LWRs [35], and will include limited access to withstand external attack and take into account evolution in terms of threat definition. It is worth mentioning that all LFR designs here considered leverage on the inherent lead properties in implementing passive safety provisions to accidents – aligning with the post-Fukushima lessons learned –, which also enhance the resilience of the plant to attacks, including cyber attacks. The capability of the reactor to ultimately spontaneously respond to initiators without operators' intervention is in fact also beneficial in case of sabotage to the control (direct attack) or monitoring (indirect attack) systems.

In case of a successful direct attack to the reactor with the defeat of all physical protection barriers finally yielding to severe core damage, the inherent characteristics of an LFR can mitigate the consequences by the scavenging effect of lead with respect to most fission products and the fact that the coolant itself does not contribute to dispersion of the radioactivity. Moreover, the tendency toward dispersion of melted fuel in lead (because of their similar density) would make the creation of a new critical fuel configuration very unlikely.

The chemical stability of lead prevents fires and a simple intervention with water would allow the cooling of the bulk lead, with the formation of a solidified outer protective layer.

The reactor can theoretically be *indirectly sabotaged* through an attack on the shut-down systems. Shut down by the operator can be also compromised by outsiders with the help of insiders.

Failure of all the shut-down systems could theoretically be an initiator of severe accidents. As an example, negative reactivity feedbacks and operation of the decay heat removal (DHR) system will intervene inherently shutting the reactor down, thereby limiting the core outlet temperature of ELFR in the range 700°C-800°C in case of Unprotected Loss of Heat Sink or in case of Unprotected Loss of Heat Sink + Unprotected Loss of Flow.

The application of the principle of defence-in-depth for the shut-down function as required by the safety analysis will provide protection also against acts of sabotage. Diversified automatic systems would be more difficult to sabotage.

To enhance safety robustness and resilience to sabotage, a passive shutdown system is considered, conceived to be mostly effective in case of Unprotected Loss of Flow and in case of Unprotected Loss of Heat Sink. The reinforcement of resistance against sabotage is mainly due to the fact these systems could be impaired only with direct attack.

The reactor can be sabotaged indirectly also through an attack to the DHR systems.

For the LFR the grace period for DHR function is relatively long thanks to the large thermal capacity of a pool type reactor. Hence, the ELFR, the primary coolant temperature will need more than 2 hours to increase by 200°C.

For the ELFR, the two independent, redundant, and diversified DHR systems use water stored inside the reactor building (DRC System) or outside the reactor building (Condenser Loop System branched from the steam-water loops). Water/steam circulation is by gravity. Actuation will require opening of valves, which can be performed manually or automatically. This diversity of DHR systems mitigates against any single system being disabled by an attack. Freezing of the dip coolers or of the steam generator (SG) connected to the steam condenser does not hamper the circulation through the remaining SGs and through the core.

For both DHR systems, the storage water pools are protected against sabotage, and steam venting to the atmosphere can take place via small ducts without intrusion possibilities and with multiple outlets to prevent risk of intentional plugging.

Also, the spent fuel storage area can be indirectly sabotaged through attack to its cooling systems with air as coolant. In both cases the grace time to recover the cooling function of the spent fuel is significantly longer than in case of the reactor core.

It has to be remarked that: not all design solutions improving safety and reliability will necessarily improve robustness against acts of sabotage, actually it might be the other way round; hence, any design solutions must balance the trade-off for the different objectives and goals as well as take into account economical aspects.

6 PR&PP Issues, Concerns, and Benefits

All proposed reference LFR systems have inherent and design features favourable to PR&PP; these include the following:

- Simple, compact core;
- Low pressure operation;
- Integral power conversion equipment;
- No intermediate cooling system;
- Lead coolant that is chemically relatively inert and has a high margin to boiling;
- Fast spectrum that offers fuel cycle and materials management flexibility;
- Minor actinide fuel;
- Natural circulation DHR.

In addition, SSTAR is a Gen-IV system specifically designed to minimize proliferation risk through its very long core life and deployment as a sealed system, eliminating access to fresh or spent fuel during the reactor life. In addition, its small size enables a small operational and security footprint.

From the pSSCs point of view, the following aspects of the design choices present proliferation resistance challenges and advantages for PR threats:

- The use of a MOX or mixed nitride fuel containing MA might increase PR;
- The long-life sealed core eliminates possibility of access by the operator, and the large size of the fuel assemblies can be handled only with the availability of dedicated specialized plant equipment and requires a high level of operator skill and training; All operations are performed remotely because of the high radiation level around the fuel elements that create a substantial barrier for access by non-state actors.

A summary of the main PR relevant intrinsic design features of the three reference designs is presented: in Appendix 1 according to the IAEA document Proliferation Resistance Fundamentals for Future Nuclear Energy Systems (IAEA-STR-332 [36]).

From the viewpoint of the pSSC, the following aspects of the design choices present physical protection challenges and advantages for PP threats. Advantages include

- System simplification;
- The use of a coolant chemically compatible with air and water and operating at ambient pressure;
- Reduced need for robust protection against the risk of catastrophic events, initiated by acts of sabotage because there is a little risk of fire propagation;
- No credible scenarios of significant containment pressurization due to design features of the steam generators that limit maximum flow rates;
- Low pressure of the primary system;
- Passive decay heat removal; and
- Compact security footprint.

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APPENDIX 1: Summary of PR relevant intrinsic design features. Reference IAEA-STR-332. Please refer to IAEA-STR-332 [36], for full explanations and complete definitions of terms and concepts.

Summary of PR relevant Intrinsic design features	ELFR	BREST-OD-300	SSTAR
Features reducing the attractiveness of the technology for nuclear weapons programmes			
1. The Reactor Technology needs an enrichment Fuel Cycle phase	No	No	No
2. The Reactor Technology produces SF with low % of fissile plutonium	The ELFR works as an adiabatic core, keeping the % of fissile Pu constant throughout the fuel assembly lifetime	The BREST reactor works as an equilibrium core, keeping the % of fissile Pu almost constant throughout the fuel assembly lifetime. Full fuel reproduction in the core, core reproduction coefficient (no blanket) is ~ 1.	The SSTAR operates with a long-life core designed to maintain a relatively constant fissile content by virtue of its conversion ratio at only slightly above 1.
3. Fissile material recycling performed without full separation from fission products	The closed fuel cycle option foresees a centralized reprocessing phase with homogeneous recycling of all actinides (i.e., no separation between U, Pu and MAs), in principle not posing a priori constraints	The closed fuel cycle option foresees on-site reprocessing phase with homogeneous recycling of Am and Np (i.e., no separation between U, Pu and MAs).	The long-life core/reactor life precludes the recycling of fuel except at supplier facilities under international control
Features preventing or inhibiting diversion of nuclear material			
4. Fuel assemblies are large & difficult to dismantle	The FAs are larger than those of SFR. No assembly disassembling is foreseen onsite (no fuel pin replacement)	The FAs are large (by size similar to SFR and VVER). Unauthorized dismantling is very difficult.	The SSTAR fuel assembly is essentially the full core. By design, the reactor vessel is sealed, and no fuel removal is envisioned while at the operator site
5. Fissile material in fuel is difficult to extract	The fuel is MOX, and as such does not pose particular separation technological challenges	Handling the nitride fuel must take place in an inert atmosphere only. Standard equipment does not allow Pu extraction during SNF reprocessing (no separation between U and Pu).	Fuel is nitride. Access to fresh or spent fuel is highly restricted due to the sealed core design and lack of refueling equipment on site. Access to fuel is limited to supplier facilities operating under international control.
6. Fuel cycle facilities have few points of access to nuclear material, especially in separated form	The closed fuel cycle option foresees a centralized reprocessing phase, in principle not posing a priori constraints	Closed fuel cycle facilities well insulated from people due to very high radioactivity of SF and reprocessed fuel. Closed fuel cycle processes and all performed operations are fully automated.	Access to fuel is limited to supplier facilities operating under international control.
7. Fuel cycle facilities can only be operated to process declared feed materials in declared quantities	The closed fuel cycle option foresees a centralized reprocessing phase, in principle not posing a priori constraints	Closed fuel cycle facilities can work only according to the strictly defined scenario. Any unauthorized interference, changing of the scenario will violate and stop the process.	Access to fuel is limited to supplier facilities operating under international control.

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	No radial nor axial blanket foreseen. Fertile targets, at least, theoretically, could however be irradiated in an in-core or reflector position	No radial nor axial blanket foreseen. BREST reactor core operates with very small reactivity margin in the power range 30-100% ($\Delta\rho$ ~ 0,65 β_{eff}) with core reproduction coefficient (no blanket) is ~ 1. Changing of several FAs (or radial lead reflector blocks) to target materials will fall core reactivity below the level of criticality – reactor operation at power will be impossible.	Access to the core precluded by design of a sealed-core system
9. The core prevents operation of the reactor with undeclared target materials (e.g. small reactivity margins)	The ELFR is a 1,500MWth reactor, it is unlikely that the insertion of one target assembly would imply sub-criticality	*See the previous answer.* Exclusion of one or several FAs from the core is impossible by the same reasons – reactor operation at power will be impossible.	Access to the core precluded by design of a sealed-core system
10. Facilities are difficult to modify for undeclared production of nuclear material	There is no foreseen disassembly of fuel assemblies on the reactor site. Being the ELFR conceived to operate in a closed fuel cycle configuration, the assemblies are designed to be eventually dismantled at the central reprocessing facility	Unauthorized FA dismantling is very difficult. Unauthorized modification of the closed fuel cycle facilities almost impossible.	Access to the core precluded by design of a sealed-core system
11. The core is not accessible during reactor operation	The ELFR foresees off-load refueling. During operation the core is closed and sealed, and the FAs continuously safeguarded thanks to their extension above the lead free level.	BREST foresees off-load refueling. At refueling will be performed automatic registration and recording of unloaded and loaded FAs. Refueling process is quite complicated and evident. Hidden unauthorized refueling is impossible. During operation the core is closed and sealed, and the FAs continuously safeguarded.	Access to the core precluded by design of a sealed-core system
12. Uranium enrichment plants (if needed) cannot be used to produce HEU	The ELFR fuel cycle does not require an enrichment step	No uranium enrichment required. Use of VVER Pu for starting loading.	No uranium enrichment required

Features facilitating verification, including continuity of knowledge			
13. The system allows for unambiguous Design Information Verification (DIV) throughout life cycle	While no DIV issues are foreseeable for the construction phase, the current level of design development does not allow performing a detailed assessment of the ease of performing DIV during the operation and decommissioning phases.	Continuous monitoring, accounting and logging of all reactor parameters is assumed.	Unambiguous Design Information Verification (DIV) throughout life cycle would be implemented
14. The inventory and flow of nuclear material can be specified and accounted for in the clearest possible manner	The system should be at least comparable with SFR designs. No issues are foreseen from the operator's side.	Continuous monitoring, accounting and logging of all parameters of the closed fuel cycle facilities. All processes are automated, no manual operations, human error or intentional concealment excluded.	Long-life core/sealed reactor system allows for full and clear accountability
15. Nuclear materials remain accessible for verification to the greatest practical extent	During operations the core is sealed and not accessible, and the FAs continuously safeguarded thanks to their extension above the lead free level. The fresh and spent fuel storages are still to be designed.	During operations the core is sealed and not accessible, and the FAs continuously safeguarded. The fresh and spent fuel storages are available for inspections.	Long-life core/sealed reactor system allows for full and clear verification
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	The current level of design development does not allow performance of a detailed assessment	Approaches and requirements are similar to those currently used in existing NPPs with VVER and SFR reactors.	The current level of design development does not allow performance of a detailed assessment
17. The system provides for the installation of measurement instruments, surveillance equipment and supporting infrastructure likely to be needed for verification	The current level of design development does not allow performance of a detailed assessment	This has to be done in accordance with the necessary requirements: rules, laws, agreements, etc. Additional up building of measurement instruments, equipment, supporting infrastructure is possible but leads to an increase in cost of NPP.	The current level of design development does not allow performance of a detailed assessment

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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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