

Politecnico di Torino



UPPSALA UNIVERSITET

Politecnico di Torino

Ingegneria Energetica e Nucleare

A.Y. 2025/2026

Graduation Session September 2026

Modeling the Impact of Flexible SMR Operation on Spent Nuclear Fuel for Safeguards Verification Purposes

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Ai miei genitori
per la loro fiducia in me
che ha reso questo lavoro possibile.

Abstract

The increasing integration of variable renewable energy sources is driving interest in load-following operation of small modular reactors (SMRs). However, the impact of such operating modes on nuclear material inventories and spent fuel composition remains to be investigated from a safeguards verification perspective.

This work evaluates the impact of load-following operation on spent nuclear fuel composition and nuclear material inventories in a soluble-boron-free pressurized water reactor SMR, the Autonomous Transportable On-demand reactor Module (ATOM). This is achieved using Serpent depletion calculations under dynamic operating conditions applied to the ATOM fuel assembly model. Of particular interest are differences in overall fissile material content, plutonium vectors, minor actinides and selected radionuclides relevant for non-destructive assay (NDA) spent fuel verification.

The results provide insight into whether load-following operation in boron-free SMRs leads to measurable deviations in spent fuel composition compared to baseload operations, and allow for investigations on the suitability of existing verification techniques and tools to verify spent nuclear fuel irradiated under non-steady-state conditions.

Acknowledgements

I would like to thank my supervisor at Politecnico di Torino, Prof. Raffaella Testoni, for her presence and availability throughout every stage of this project. I will always be grateful for her support in finding a way to deepen my knowledge of the topics I am the most interested in.

My gratitude also goes to my supervisors at Uppsala University for their guidance and inclusiveness, which gave me the chance to grow and expand my network of knowledge. I would like to thank Prof. Sophie Grape for giving me the opportunity to join her research group and undertake this project, as well as for her close supervision throughout it. I am grateful to Dr. Erik Branger and Claudia Olaru, who never hesitated to offer their valuable advice and constant support.

A special thanks goes to my colleagues at Uppsala University, who made my stay in Sweden fun and gave me the opportunity to explore many new aspects of physics that I now find fascinating.

Inoltre vorrei ringraziare tutte le persone che anche senza partecipare attivamente a questo progetto lo hanno reso possibile.

I miei genitori, che mi hanno dato non solo i mezzi, ma anche la fiducia per affrontare questo percorso, e i miei fratelli, con cui continuo a crescere e che hanno attraversato con me questi anni di studio. Grazie per aver festeggiato ogni mio traguardo e per aver alleggerito i momenti più faticosi.

Gabri, per la sua infinita pazienza e comprensione, per l'impegno e l'amore che ha dimostrato sempre. Nei momenti in cui mettevo tutto in dubbio, è rimasto sempre la mia certezza.

Elisabetta e Pierpaolo, che con la loro gentilezza, apertura e generosità, mi hanno regalato un luogo sereno in cui sentirmi sempre accolta.

Tutti i miei amici e colleghi, che sono stati insostituibili e hanno reso questi anni speciali. Una frase non basta per racchiudere quello che la vostra amicizia ha significato per me. Se è vero che gli amici sono la famiglia che ci scegliamo, siete la migliore scelta che ho fatto fino ad ora.

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Acronyms

AMR	Advanced Modular Reactor
AP	Additional Protocol
ATOM	Autonomous Transportable On-demand reactor Module
BA	Burnable Absorber
BL	Base-Load
BU	Burnup
BWR	Boiling Water Reactor
C/S	Containment Surveillance
CR	Control Rod
CSA	Comprehensive Safeguards Agreement
CSBA	Centrally Shielded Burnable Absorber
CT	Cooling Time
CVCS	Chemical and Volume Control System
CVD	Cerenkov Viewing Device
CZT	cadmium zinc telluride
DBA	Design Basis Accident
DCVD	Digital Cerenkov Viewing Device
EUR	European Utility Requirements
FA	Fuel Assembly
FDET	Fork Detector
FOAK	First Of A Kind
FP	Fission Product
GFR	Gas-cooled Fast Reactor
HALEU	High Assay Low-Enriched Uranium
HTGR	High Temperature Gas-cooled Reactors
HWR	Heavy Water Reactor
IAEA	International Atomic Energy Agency
ICVD	Improved Cerenkov Viewing Device
IE	Initial Enrichment

iPWR	Integral pressurized water reactor
IRAT	Irradiated Fuel Attribute Tester
LEU	Low Enriched Uranium
LF	Load-Following
LMFR	Liquid Metal-cooled Fast Reactors
LWR	Light Water Reactor
MSR	Molten Salt reactors
MTC	Moderator Temperature Coefficient
NDA	Non-Destructive Assay
NNWS	Non-Nuclear-Weapon State
NPP	Nuclear Power Plant
NPT	Treaty on the Non-Proliferation of Nuclear Weapons
NRC	Nuclear Regulatory Commission
NWFZ	Nuclear Weapon Free Zones
NWS	Nuclear-Weapon State
OCC	Overnight Capital Costs
O&M	Operation and Maintenance
PGET	Passive Gamma Emission Tomography
PWR	Pressurized Water Reactor
SBF	Soluble Boron Free
SF	Spontneous Fission
SFAT	Spent Fuel Attribute Tester
SMR	Small Modular Reactors
SNF	Spent Nuclear Fuel
TF	Thermal-Fission
TOP	Truly-Optimized PWR Lattice
TRU	Transuranic element
VOA	Voluntary Offer Agreement

Introduction

The increasing demand for electricity, together with the growing integration of variable renewable energy sources, is changing the operational requirements of electricity systems. As electricity generation from renewable sources depends on variable environmental and meteorological conditions, other generating units must compensate for variations in their output to maintain the balance between electricity production and demand. In this context, although nuclear power plants have traditionally operated under steady-state conditions, their participation in load-following operation is become increasingly valuable in low-carbon energy systems.

Small modular reactors (SMRs) are particularly relevant in this context because their reduced power output, modular deployment and enhanced manoeuvring capabilities may facilitate their integration into grids with a high share of renewable generation. Nevertheless, operating a reactor under variable-power conditions changes the relationship between irradiation time and energy production. Even when the same final burnup is reached, differences in neutron flux and irradiation history may affect the evolution of the fuel composition and the accumulation of radioactive materials.

From a nuclear safeguards perspective, the composition of spent nuclear fuel (SNF) is of vital importance. Verification methods rely largely on gamma-ray and neutron signatures associated with specific radionuclides to confirm declared fuel parameters and detect possible diversion. For this reason, changes in fuel inventories caused by unsteady operation could influence the relationship between measured signatures and fuel characteristics weakening the reliability of these signatures as indicators of fuel characteristics.

Consequently, assessing the influence of load-following operation of SNF composition is necessary to determine whether established safeguards approaches remain applicable to spent fuel discharged from flexibly operated SMRs.

Goal

The purpose of this work is to evaluate of the impact of load-following operation on SNF composition and nuclear material inventories in a small modular boron-free pressurized water reactor (PWR) knows as Autonomous Transportable On-demand reactor Module (ATOM). This is achieved using Monte Carlo simulations and the Serpent2 code for depletion calculations under dynamic operating conditions applied to the ATOM fuel assembly (FA) model. Of particular interest are differences in overall fissile material content, plutonium vectors, minor actinides and selected radionuclides relevant for non-destructive assay (NDA) spent fuel verification.
The obtained results will provide a comparison of the spent fuel composition in the case of seasonal load-following (LF) operation and base-load (BL) operation. In addition, investigations on the suitability of existing verification techniques and tools to verify SNF irradiated under non-steady-state conditions are possible.

Thesis structure

This work is structured as described here. In Chapter 1, an introduction to SMRs is provided in order to present their main characteristics, potential applications, and relevance in the current nuclear energy context. The reference SMR design used for this work is described in Chapter 2. Then an overview of nuclear safeguards follows in Chapter 3, with particular attention to their role in the verification of SNF and to NDA methods in particular. After this background, Chapter 4 describes the methodology adopted to obtain the results of the depletion calculation with Serpent2. This includes the assumptions, data, and procedures used throughout the analysis. Finally, the results are presented and discussed in Chapter 5, highlighting the main findings and their implications for safeguards verification in the context of SMR operation. The most important conclusions of this work are then summarized in the conclusion.

AI Use Disclosure

Generative AI tools were used exclusively for editorial support, such as grammar correction, translations, paraphrasing suggestions, or improving readability.

Chapter 1

Small Modular Reactors

SMRs are defined by the International Atomic Energy Agency (IAEA) as advanced nuclear reactors with a power capacity of up to 300 MWe per unit, representing approximately one third of the generating capacity of traditional nuclear power plants (NPPs) [1]. These reactors are physically smaller than conventional models and feature a modular design that allows systems and components to be factory assembled and transported as a single unit for installation. They represent a wide category of reactors and many characteristics may very among it. One example is the power output, which is very variable, with units typically classified as microreactors when their capacity falls below 10 MWe. Another distinction applies to reactor technology: while SMRs are often derived from established light water technology systems in the form of Generation III/III+ designs through the down scaling of large-scale reactors, others are based on advanced Generation IV concepts that employ a totally different technology. Many Generation IV concepts are also still under development, including molten salt reactors and lead-cooled reactors. SMRs employing Generation IV technology are sometimes referred to as advanced modular reactors (AMRs), to clarify the distinction from light-water SMRs.

SMRs offer a versatile solution for locations that cannot accommodate larger plants due to geographical or infrastructure constraints as well as for sites where the energy demand is less than a large scale reactor and the output of an SMR is more appropriate. The ability to manufacture prefabricated units can lead to significant savings in construction time and costs, as these units avoid the complexities often associated with custom designed large scale reactors. Furthermore, SMRs can be deployed incrementally to match rising energy demand, making them a flexible option for expanding energy grids. In regions with limited grid capacity, SMRs provide a viable path for providing low-carbon power to industry and populations. SMRs are also expected to provide different forms of power outputs to meet specific load demands, such as electricity generation, propulsion of marine vessels, and powering spacecrafts.

Safety is a hallmark of proposed SMR designs, which generally rely on passive systems and inherent characteristics such as low operating pressure and natural physical phenomena like gravity and convection to shut down without human intervention. These increased safety margins significantly reduce the potential for accidental radioactive releases into the environment.

Additionally, some SMRs have longer operating cycles, reducing the need for frequent refuelling. Some designs require refueling only every 3 to 7 years, compared to the 1 to 2 year cycles of conventional plants. Beyond electricity generation, many SMRs are also designed to support various applications including district heating, water desalination, and industrial steam, and they can be paired with variable renewable energy sources in hybrid systems to ensure a stable and efficient clean energy transition.

While SMRs offer significant potential benefits, their development and deployment still face several important challenges. These include obtaining regulatory approvals, strengthening public acceptance, and demonstrating the cost and performance advantages of SMR designs. Nevertheless, the global interest in this technology is clearly evident. Currently, over 100 commercial SMR designs are under development worldwide, with the United States leading by a wide margin with 45 designs, followed by Russia with 18, China and France with 12 each, and Japan with 10. SMRs are also being developed in Canada, South Korea, and the United Kingdom [2]. This broad international engagement highlights the strong potential for cooperation in SMR development, as it could accelerate technology transfer, support the exchange of best practices, and allow countries to combine resources for joint research and development efforts.

This chapter introduces the main characteristics of SMRs in order to provide the background for the analysis carried out in this work. Particular attention is given to aspects that are relevant to flexible operation, since SMRs are often proposed not only for BL electricity generation, but also for applications in which reactor power may vary in response to changing demand or renewable energy production. This is important because flexible operation can influence the irradiation history of the fuel. Variations in power history may lead to differences in burnup evolution, isotopic composition, and radiation-related quantities after discharge. These quantities are important for safeguards verification, where spent fuel measurements can be used to support the verification of declared reactor operation and fuel characteristics.

1.1 SMR Technical Characteristics

SMRs are being developed by many different entities around the world, and several different types have been proposed. As mentioned above, SMRs can be classified according to their power output or generation, as well as by coolant type, neutron

spectrum, or deployment strategy. These classification criteria are largely the same as those used for conventional large-scale reactors, since SMRs are often based on similar reactor physics principles and technological categories, although applied to smaller and more modular designs.

In addition to these classification criteria, safety features represent another important aspect for understanding SMR technology. Enhanced safety is one of the main motivations behind many SMR designs, especially through compact and integrated layouts, passive safety systems, and reduced dependence on active components or operator intervention. For this reason, the following sections discuss not only the main technological characteristics used to classify SMRs, but also the safety features that distinguish them from conventional large-scale reactors.

1.1.1 Reactor Technologies

Among the different classification criteria, reactor technology and coolant type are the most widely used, since they describe the basic operating principles of the reactor and the way heat is removed from the core [3]. Following this classification, reactors and, as a consequence, SMRs can be divided into:

- Light Water Reactors (LWRs)
- Heavy Water Reactors (HWRs)
- Gas-cooled fast reactors (GFRs) and High Temperature Gas-cooled Reactors (HTGRs)
- Liquid Metal-cooled Fast Reactors (LMFRs)
- Molten Salt reactors (MSR)

LWRs come in two types: PWRs and boiling water reactors (BWRs). As documented by the SMR database [2], 26 proposed SMRs are PWR type designs, and overall 47 proposed designs are cooled and moderated by water.

A PWR has a pressurized primary circuit and a steam generator working as a heat exchanger between the primary loop and the secondary turbine loop. In fact, the power generated from the fission reaction is transferred to the turbine through a heat transport system, without direct contact between the reactor coolant and the turbine. All current PWRs use FAs filled with uranium dioxide (UO_2) fuel rods. Natural uranium is mostly composed of ^{238}U and an enrichment process is needed to increase the percentage of the fissile uranium isotope ^{235}U from 0.7% to a value that usually does not exceed 5%. In the case of PWR SMRs, the physical size of the core is smaller than that of a large-scale NPP, so the active fuel rods my be shorter in length. Higher initial enrichments (IEs) are often proposed, especially for AMRs, to compensate for the smaller core size, longer fuel cycles, and higher neutron leakage, while maintaining sufficient reactivity throughout operation.

Compared to PWRs, the pressure in BWRs is much lower, making two phase flow possible. BWRs can extract the heat from the fuel through a unique structure

that connects directly the core and the turbine, enabling special power control by flow rate and pressure manipulation. An aspect to consider when using BWRs is the fact that liquid water has a better moderation capability compared to steam, thus neutrons will be moderated more aggressively in the environment with more liquid water. As a result, the reactivity can drop during an increase in the void ratio.

The other reactor types mentioned above belong to the Generation IV category and fall outside the scope of this work, which focuses on LWR technology. A brief overview of AMR concepts is provided in Appendix A for interested readers.

1.1.2 Safety Features

When discussing nuclear reactors, safety is the fundamental constraint that influences every aspect of design. Traditionally, NPPs have relied mainly on active safety systems, while many SMRs are being designed to rely more extensively on inherent and passive safety features [4].

Nuclear reactor safety systems are designed to protect people and the environment by preventing accidents, responding quickly and reliably during abnormal conditions, and limiting the consequences of accidents if they occur. Many accident scenarios that may occur during the reactor lifetime are considered at the design stage, and specific safety systems are developed to address them. However, not all possible accidents can be explicitly accounted for, since very low-probability events may not be practical or reasonable to include in the design basis. After a cost-benefit analysis, accidents are therefore classified either as design basis accidents (DBAs) or as beyond DBAs, which are not specifically considered in the original design.

At the core of nuclear safety are three essential functions: controlling the nuclear chain reaction, removing heat from the reactor core, and ensuring the containment of radioactive materials [4]. To achieve these objectives, reactors are designed according to a defense-in-depth approach, which relies on multiple independent layers of protection, including engineered safety systems such as the emergency core cooling system and physical barriers such as the fuel cladding and the containment building. The purpose of this approach is to ensure that, even if one system fails, other independent systems and barriers continue to provide protection.

Another key aspect of nuclear safety design is the use of redundancy and diversity [4]. Redundancy means that multiple systems are available to perform the same safety function, while diversity means that these systems may rely on different technologies or operating principles. Together, these principles reduce the probability that a single failure could lead to dangerous consequences.

Overall, nuclear reactor safety systems combine robust engineering, multiple protective barriers, and careful risk management to ensure that the probability

of accidents remains extremely low and that any potential consequences are kept within acceptable safety limits [4].

Safety Approach in SMRs

When considering SMRs, a tiered approach to nuclear safety is being developed. This approach begins with safety-by-design, followed by passive safety systems, and finally active safety systems [4].

Safety-by-design aims to eliminate or substantially reduce the probability of DBAs. In fact, SMRs are designed to eliminate entire classes of accidents, so that the corresponding safety systems can be simplified or may no longer be required. Accident scenarios that cannot be eliminated are expected to have a significantly reduced frequency and severity.

Modern SMR designs improve safety primarily through simplified and integrated engineering. By placing key components, such as steam generators, inside the reactor pressure vessel, large external piping can be eliminated, significantly reducing the risk of major loss of coolant accidents [4]. Integral pressurized water reactors (iPWRs) also have a larger water inventory, which enhances heat absorption and provides more time to manage cooling after shutdown. Additional design improvements, such as internal control rod (CR) drive mechanisms and fewer vessel penetrations, further reduce potential failure points and contribute to a more compact and robust reactor configuration.

These new designs often rely on passive safety systems, which represent the second safety tier and use natural forces to mitigate or prevent the consequences of an accident. This differs from active safety systems, which require operator action, external power, or mechanical actuation.

SMRs differ from conventional reactors because of their enhanced use of passive safety. Also new large-scale reactors rely deeply on passive safety systems, but the smaller size of SMRs makes them a lot more effective. One system that shows this difference clearly is the decay heat removal system [4]. Its purpose is the removal of decay heat after reactor shutdown as well as important safety functions during accident conditions. In traditional reactors it may require an electric actuation signal, a motor-driven or manually operated valve, a pump to establish coolant flow, and an external power supply. These requirements can create accident vulnerabilities, which are usually mitigated by using multiple redundant and diverse external power sources. In SMRs, by contrast, passive decay heat removal systems can rely on natural convection and gravity-driven circulation of coolant water. Since coolant pumps and external actuation systems are not required, these systems are generally considered more reliable and robust.

Overall, these reactors also contain a smaller inventory of radioactive material and are designed to maintain safe conditions for extended periods without active

intervention, thereby improving safety and reducing the potential consequences of an accident.

1.1.3 Cost

The SMR economic performance is related to technical features, including reactor size, fuel-cycle duration, operating strategy, and fuel inventory evolution. Although this work does not perform a detailed economic assessment, SMR costs are relevant to the broader context of their deployment. For this reason, a brief overview of the main cost components and economic challenges of SMRs is included.

The assessment of the economic viability of SMRs is challenging, largely because most designs are still in the development, demonstration, or prototyping phase. Economic assessments rely on projected life-cycle costs and the cost estimates are therefore significantly uncertain, particularly for first-of-a-kind (FOAK) projects. Capital expenditure is a major component of life cycle costs and includes both overnight capital costs (OCC) and financing costs. OCC covers engineering, procurement, construction, land acquisition, site preparation, project management, licensing, and permitting. Financing costs consist of the interest accumulated during construction and depend strongly on the project's cost of capital. Overall, OCCs are the main driver of NPP economics, accounting for approximately 50-75% of total life-cycle costs [5]. Operation and maintenance (O&M) costs include all non-fuel operating expenses. These costs are expected for SMRs to be higher per unit of electricity compared to large NPPs, but overall they affect nuclear costs less than capital expenditure. Decommissioning must also be considered when evaluating life cycle costs, however, reliable data for SMRs remain limited since no commercial SMR project has yet completed the decommissioning process.

Fuel costs for SMRs may differ from those of conventional reactors because of variations in design, fuel type, and fuel cycle duration. Light water SMRs using low-enriched uranium (LEU) are expected to have fuel costs broadly comparable to existing large light water reactors. However, several advanced SMR concepts require high-assay low-enriched uranium (HALEU), which is currently produced commercially only in Russia and China. As a result, supply chain constraints and fuel availability are likely to influence future fuel costs. Moreover, many SMRs are designed with longer fuel cycles, requiring refueling every three to seven years instead of every one to two years for conventional reactors, leading to different fuel cost structures.[5]

A further limitation is the reduced economy of scale. Compared with SMRs, conventional large nuclear reactors can distribute costs across a greater electricity output. Studies indicate that reactor costs decline by approximately 20-35% as reactor size doubles, while SMR capital costs may be up to 70% higher on a per-unit output basis than those of reactors using similar technologies [5].

Despite these challenges, several characteristics of SMRs could potentially offset their higher unit costs. One of the most frequently cited advantages is modularization, which involves standardized factory fabrication of reactor components that can later be transported and assembled on-site. This construction strategy is expected to simplify installation and reduce project complexity, although estimates suggest that approximately 60% modularity may be necessary to achieve meaningful cost savings[5]. Cost reductions may also arise through learning effects, whereby efficiencies improve as experience accumulates from repeated manufacturing and deployment of reactor units. Such learning effects could potentially reduce costs by approximately 5-10% over time [5]. In addition, the smaller size and simplified design of SMRs are expected to shorten construction schedules relative to conventional large reactors. This may produce capital cost savings in the range of 6-20% by reducing financing expenses associated with long construction periods[5].

Another potential economic advantage is co-siting, which involves deploying multiple reactor units at a single location. Shared infrastructure, staffing, and operational systems could lower overall capital costs by approximately 10-25% per unit [5].

Overall, while SMRs may not always represent the lowest-cost option in highly competitive electricity markets, they may still offer a promising alternative within broader energy systems when factors such as reliability, grid stability, and flexibility are considered. Nevertheless, significant uncertainty remains regarding future SMR costs, particularly for FOAK projects, which are expected to incur substantially higher expenses before cost reductions can be achieved through accumulated industrial experience.

1.1.4 Regulation and Licencing

The development of a civilian nuclear power program requires a comprehensive national legal framework consistent with international principles on nuclear safety, security, safeguards, and liability. A key element of this framework is the establishment of an independent nuclear regulatory authority responsible for overseeing the sector. Its responsibilities typically include developing regulatory standards, licensing and authorizing nuclear installations, conducting inspections, and carrying out enforcement activities. The regulator must also ensure that all stages of the nuclear facility life-cycle, including design, site selection, construction, operation, waste management, decommissioning, and emergency response, comply with the established regulatory framework.

Compliance with international conventions on nuclear safety, security, safeguards, and liability is essential for all nuclear installations, including SMRs. These frameworks aim to protect nuclear facilities from both internal and external threats, and some of them are shown in Table 1.1. Since SMRs pose many of the same

risks as conventional NPPs, the core legal and regulatory principles applied to large reactors are generally also applicable to SMRs. However, some advanced SMR concepts create regulatory uncertainties that require further consideration within national legislation. In particular, floating NPPs raise unresolved questions concerning, for example, maritime law and nuclear security [6].

Table 1.1: Selected Examples of International Instruments on Nuclear Safety, Security, Safeguards and Liability [5]

Nuclear Safety and Security	• Convention on Nuclear Safety (1994) • Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (1997) • Convention on Early Notification of a Nuclear Accident (1986) • Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (1986) • Convention on the Physical Protection of Nuclear Material (1979) • Amendment to the Convention on the Physical Protection of Nuclear Material (2005)
Safeguards	• Treaty on the Non-Proliferation of Nuclear Weapons (NPT) • Comprehensive Safeguards Agreements (CSA) with the IAEA • Zangger Committee • Wassenaar Arrangement • Nuclear Suppliers Group
Nuclear Liability	• 1963 Vienna Convention on Civil Liability for Nuclear Damage (IAEA, 1963) • Protocol to Amend the Vienna Convention on Civil Liability for Nuclear Damage (IAEA, 1997) • Paris Convention on Third Party Liability in the Field of Nuclear Energy of 2004 (OECD, 2004) and the Brussels Convention on Civil Liability for Nuclear Damage (OECD, 2004) • Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention (1988) • Convention on Supplementary Compensation for Nuclear Damage (IAEA, 1997).

For countries with limited experience in nuclear governance, international assistance mechanisms can support the development of legal, regulatory, and institutional capacity. The IAEA Milestone Approach provides a structured pathway for states seeking to establish a nuclear power program, while organizations such as the Western European Nuclear Regulators Association (WENRA) facilitate cooperation and the exchange of regulatory expertise, including on SMR-related issues. Established regulators in countries such as the United States, South Korea, Canada, and the United Kingdom also support nuclear capacity-building through bilateral partnerships and technical collaboration.

Licensing is one of the core responsibilities of a nuclear regulatory authority. In general, SMR licensing follows the same fundamental procedures applied to conventional large-scale NPPs, although adjustments may be required to account for innovative technologies and operational characteristics. In several countries, the formal licensing process may be preceded by a pre-application or pre-licensing engagement phase, during which the regulator and developer exchange information on technical specifications, safety considerations, and regulatory expectations. Although this practice is not universal, it has become increasingly common because it helps developers identify potential regulatory issues before formal submission and allows regulators to become familiar with emerging technologies, possible regulatory gaps, and resource needs for later licensing stages [6].

After the pre-application phase, SMR projects proceed to the formal licensing process, which differs across national regulatory systems. In the United States, the Nuclear Regulatory Commission (NRC) is the independent regulator for civilian nuclear power and is widely regarded as one of the most influential nuclear regulatory authorities: certification of an SMR design by the NRC provides an important degree of confidence regarding reactor safety. In countries with more limited nuclear regulatory experience, construction and operation licenses are generally issued separately. Regulatory oversight then continues throughout construction, operation, and decommissioning, through inspections, measurements, and performance assessments aimed at verifying compliance with the approved license and regulatory requirements.

Therefore, the smaller physical size of SMRs does not necessarily imply a simpler licensing process under existing regulatory systems. Many SMR concepts include novel technological features that are not fully addressed by current regulations, requiring designers to justify these innovations and potentially extending licensing timelines. Non-light water reactor designs tend to face greater scrutiny because of limited operational experience and supporting regulatory studies. In addition, the increasing use of digital automation and remote-control systems introduces cybersecurity considerations, requiring regulators to develop a more advanced understanding of cyber risks and digital system vulnerabilities [5].

1.2 Expected Use of SMRs

Interest in SMRs is growing among energy-intensive users because, beyond their potential to decarbonize electricity production, they may also provide both high-grade and low-grade heat. This makes them versatile and applicable in different scenarios, including electricity generation, marine propulsion, spacecraft propulsion, and heat generation. Each of these applications involves different operating conditions and design requirements.

In electricity generation, SMRs can be integrated with renewable energy sources, especially in stand-alone microgrids or remote communities. In these systems, an SMR may act as the main dispatchable power source, supporting the grid when renewable sources such as photovoltaic panels or wind turbines are not producing enough electricity. This differs from large NPPs connected to strong grids, where other power sources, such as hydro or gas plants, often help maintain the balance between supply and demand. [3]

One possible application is cogeneration. In this case, SMRs could improve overall efficiency by using waste heat from electricity production for additional processes, such as carbon capture and storage. SMRs could also provide the electricity and heat required for hydrogen production. Hydrogen can be considered sustainable when produced without CO_2 emissions, and synthetic fuels derived from hydrogen can mimic the performance of fossil fuels, making them a promising alternative for carbon-intensive sectors such as aviation. [7]

Another relevant application is the supply of energy to data centers. These large facilities are responsible for the storage, processing, and distribution of data, and they may require approximately 100 MW of power, which could potentially be provided by a single SMR. Currently, data centers mainly rely on fossil fuels for their energy supply. [7]

SMRs are also suitable for heat-generation applications because nuclear reactors are inherently thermal energy sources. Their small size and modularity make them potentially useful for supplying heat to remote industrial sites or communities. Since different industrial processes require different temperature levels, the choice of reactor type depends strongly on the intended application. HTGR are often considered especially promising for industrial thermal applications. Some SMRs may also be designed specifically to supply heat for industrial processes, such as steam production for mining sites or other remote facilities. [7]

SMRs could also be used to provide propulsion and onboard power for ships. In the past, nuclear reactors have already been used in submarines, aircraft carriers, icebreakers, and merchant ships. These marine reactors had power outputs and physical dimensions comparable to land-based SMRs, but their design requirements are differ because they operated in a marine environment and had to support both propulsion and shipboard systems. Marine reactors could benefit from the ocean as a large heat sink, and highly enriched uranium fuel, allowing compact cores and long core lifetimes. [3]

Nuclear reactors are also considered useful for spacecraft applications because they can provide a large amount of energy relative to their mass. Some SMR concepts have been proposed for spacecraft or extraterrestrial base applications, with nominal power levels ranging from tens to thousands of kilowatts. In space applications, nuclear reactors may be used to generate electricity when solar intensity is low, to provide propulsion, or to support both functions in dual-mode

systems. [3]

Finally, heat produced by SMRs could also be used for water desalination, enabling the conversion of seawater into potable water.

1.2.1 SMR LF Operation

For most of the past 70 years, NPPs have been mainly BL source of electricity. As a result, most nuclear reactors used for electricity generation were designed to meet stable BL demand, with limited attention given to LF operation. However, in countries where nuclear power became a major part of the national electricity mix, utilities had to introduce or improve the maneuverability of their NPPs so that electricity supply could adapt to daily and seasonal variations in demand. [3]

This situation has changed further with the growing integration of renewable energy sources. In hybrid energy systems, especially off-grid systems that combine nuclear power with non-dispatchable energy resources, reactors must be able to respond to operational commands through rapid power ramp-up and flexible output control. For this reason, many recent SMR designs include the capability to vary their power level. Today, LF capability is therefore one of the key features that distinguishes SMRs from traditional large-scale reactors in existing NPPs. However, in France and previously in Germany, NPPs already operate in LF mode, participating in primary and secondary frequency control. Primary frequency control automatically adjusts plant power within seconds to limit grid-frequency deviations, while secondary frequency control acts over seconds to minutes to restore the target frequency and scheduled power exchanges. Some units also follow variable load programs with one or two large power changes per day [8].

In practice, LF can be achieved through two main approaches [3]. The first is a direct approach, based on reactivity control and/or turbine control. The second is an indirect approach, based on the use of flexible loads and/or energy storage. These techniques are necessary, but they also have important limitations. Varying the core power can contribute to corrosion, thermal fatigue, and wear in some reactor components. Turbine bypass can reduce output power rapidly, but it causes unnecessary energy losses and creates operational challenges for the condenser and cooling system.

The LF requirements of SMRs depend strongly on the application. In electricity generation, especially in stand-alone microgrids with renewable energy sources, the SMR may need to adjust its power output to compensate for variations in renewable production and changes in electricity demand. These demand changes can be predictable, such as daily or seasonal cycles, or more random, such as sudden changes caused by variable photovoltaic generation on cloudy days [3]. Modern NPPs equipped with light-water reactors are designed to have strong maneuvering capabilities. The minimum maneuverability requirements of modern reactors are

defined by utility requirements, which are based on the needs of grid operators. For example, according to the European Utility Requirements (EUR), a NPP must be capable of daily load cycling between 50% and 100% of its rated power, P_r, with a rate of change of electric output of 3-5% of P_r per minute [9]. In practice, the daily variation of nuclear generation is typically lower. In France, it is usually below 5-10% of total nuclear generation, with an average daily variation of about 6.7% in 2010. However, during some periods, the daily variation can exceed 20% [8]. During warmer periods of the year, nuclear generation is lower because of the higher cooling-water temperature, and the relative daily variation is therefore higher.

In marine propulsion, LF is especially important because ships frequently change speed and may need to reverse direction quickly. As a result, marine reactors can require much faster power maneuvering than land-based SMRs used with photovoltaic generation [3].

In spacecraft propulsion, the requirements can be even more severe, since the reactor may need to change its output rapidly to adjust thrust or respond to changes in electrical load between electricity generation and propulsion [3].

In heat-generation applications, LF is usually less rapid than in propulsion or electricity generation, but it is still necessary because industrial heat demand can vary depending on process needs, time of day, weather conditions, start-up, shutdown, or process outages [3].

1.2.2 Deployment

The global development of SMRs has accelerated significantly in recent years. Around 130 SMR technologies are currently being pursued worldwide, with several designs already moving beyond the conceptual stage and entering pre-licensing, licensing, or construction phases. Regulatory activity is particularly visible in countries such as the United States, the United Kingdom, Canada, and South Korea, where different SMR designs are being assessed or approved by national safety authorities [2]. This international momentum shows that SMRs are no longer only a long-term research concept, but are increasingly becoming part of concrete national energy and industrial strategies.

At present, five SMR projects can be identified as being under construction or in an advanced construction phase [2]. One example is the ACP100, also known as Linglong One, a Chinese water-cooled thermal reactor with an output of 125 MWe. Its preliminary safety analysis report was approved in 2020, and construction has advanced rapidly in recent years. Functional tests were completed in December 2025. Commercial operation is expected in the first half of 2026, which would make it one of the first land-based commercial SMRs to enter operation.

Another project under construction is Argentina's CAREM25, a domestically

developed water-cooled thermal SMR prototype. The reactor has a small electrical output, usually reported at around 25-32 MWe depending on the source and configuration. First concrete was poured in 2014, making CAREM one of the earliest SMR construction projects in the world. However, the project has experienced repeated delays, including a halt in construction due to budgetary and workforce issues.

Beyond these SMR concepts, the global landscape is also characterized by a wide diversity of AMRs technologies. Although most AMRs designs are still at the development or licensing stage, two projects are already in operation [2]. In China, the HTR-PM at Shidao Bay in Shandong Province is the world's first modular high-temperature gas-cooled pebble-bed reactor power plant to enter commercial operation. In Russia, the Akademik Lomonosov is the world's first floating NPP. [2]

The existence of projects under construction and in operation demonstrates that the deployment of SMRs and AMRs is beginning to move from planning to implementation in a context of a wider global race for regulatory innovation, industrial supply-chain development, first-of-a-kind demonstration projects, and competition for technological leadership. [2]

In this international context, the European Commission presents the deployment of SMRs in Europe in "Strategy for the development and deployment of SMRs in Europe" [10] as both a strategic energy objective and an industrial policy priority.

The European Context

Overall, the EU deployment strategy is based on coordinated action among EU institutions, Member States, industry, and research organizations to create the financial, industrial, regulatory, and technological conditions needed for SMR deployment in Europe by the early 2030s. Instead of treating SMR development as a set of separate national initiatives, the European Commission frames it as a shared European industrial project requiring coordination in research, supply chains, licensing, skills, and financing. A central element of this strategy is the European Industrial Alliance on SMRs, established in February 2024. The Alliance brings together almost 400 organizations and aims to support the operation of the first SMRs in the EU by the early 2030s [11]. In 2024, it launched a first call to identify concrete projects, and in September 2025 it presented its 2025-2029 Strategic Action Plan. The Alliance focuses on two main technological pathways: light-water SMRs, which are closer to deployment, and AMRs, which still require further research, testing, and demonstration. The Commission also stresses that deployment will depend on the creation of a strong European supply chain. This includes reactor components, modular manufacturing capacity, and fuel-cycle services such as enrichment and conversion. In this context, the Commission

supports a fleet approach, where similar or standardized reactor designs could be deployed across several Member States, to help reduce construction times and costs.

Industrial cooperation and project consolidation are also considered essential. SMR developers, start-ups, scale-ups, utilities, end-users, and supply-chain companies are encouraged to work together, share testing facilities and resources, and cooperate on demonstration projects, even if they may later compete in the final market.

Financing remains one of the main barriers to SMR deployment. As peviously discussed, the SMR business model has some advantages compared to the large scale NPPs, however, this model has not yet been proven in practice and public support is therefore considered necessary to mobilize private investment and de-risk FOAK projects. Some of the possible instruments identified by the EU commission include guarantees, the Innovation Fund, InvestEU, Important Projects of Common European Interest, the Net-Zero Industry Act, and the Clean Industrial Deal State Aid Framework. These initiatives show that the EU strategy is not limited to reactor construction, but also includes the creation of industrial ecosystems able to support manufacturing, supply chains, and future applications. Research, innovation, and regulatory cooperation are further pillars of the deployment strategy. In 2024, five Euratom research projects related to light-water SMRs and AMRs were launched. [10]

On the regulatory side, the European Nuclear Safety Regulators' Group has created a taskforce where regulators from several Member States exchange information on SMR designs during the pre-licensing phase. This cooperation could allow countries to align licensing procedures or mutually recognize certain licensing decisions.

Finally, the Commission emphasizes the importance of public awareness and international cooperation. Public trust is considered essential for SMR deployment, and the Commission supports transparent communication on the characteristics, safety measures, and potential benefits of SMRs, especially at local level. Internationally, the EU intends to cooperate with organizations such as the IAEA and the OECD Nuclear Energy Agency, as well as with partner countries including the United States, Canada, the United Kingdom, Japan, and South Korea.

Chapter 2

ATOM SMR

This chapter focuses on the description of the ATOM SMR, which was selected as the reference reactor for this work, as it provides a real and developed SMR concept on which the study can be based, rather than a generic reactor model. Moreover, its soluble boron free (SBF) operation and load-following (LF) capability make it relevant for assessing the impact of flexible operation on spent fuel characterization and signatures. In fact, compared with current large PWRs), ATOM presents significant design differences, including a compact core, a different reactivity-control strategy, and extensive use of burnable absorbers (BAs). These features make it an interesting benchmark for assessing whether flexible operation can affect spent fuel characteristics relevant for safeguards verification [12].

Developed at the Korea Advanced Institute of Science and Technology (KAIST), the ATOM core is characterized by its compact, integrated design and innovative reactivity control mechanisms. The integration of the heat exchanger and steam generator inside the SMR pressure vessel enhances reactor safety and simplifies the overall system, potentially reducing construction time. In addition, removing complex reactivity control systems such as the Chemical and Volume Control System (CVCS) makes the reactor more compact. [12]

The particular aspect of this reactor is the absence of soluble boron, which is used in common reactors to provide homogeneous reactivity control. Soluble boron has several well known drawbacks, including the possibility of a positive moderator temperature coefficient (MTC) at high concentrations, crud accumulation, slow reactivity response, and material corrosion. Therefore, SBF operation is being developed to improve passive autonomous LF capability, including frequency control operation.

In general, a key design target in a PWR core is to maintain a small excess reactivity throughout the entire operating cycle. This is needed to reduce dependence on CRs, to improve the cold shutdown margin and to minimize power distortion. In order to achieve this small excess reactivity, the BA, which acts

as the main reactivity control mechanism replacing soluble boron, must deplete gradually during the cycle. In fact, if the depletion of the BA matches the one of the fuel, the need to make CR adjustments during operations is strongly reduced.

2.1 TOP Lattice and Neutron Economy

Despite having many advantages, the smaller core size of an SMR increases neutron leakage in both the radial and axial directions, which lowers uranium utilization compared with a typical PWR. To improve neutron economy, the ATOM core adopts an enhanced-moderation FA, called the Truly-Optimized PWR (TOP) lattice [13].

Most current SBF SMRs use the standard 17×17 lattice, originally optimized for soluble-boron operation to maintain a negative MTC throughout reactor operation. Starting from the standard 17×17 FA, neutron moderation can be enhanced in two ways: by increasing the pin pitch and FA size while keeping the same fuel radius, or by reducing the fuel radius while preserving the original FA dimensions. For the ATOM SMR, the smaller-fuel-radius TOP lattice is adopted. As it is possible to see in Figure 4.1, this TOP lattice also includes eight Er_2O_3-bearing fuel rods positioned next to the guide tubes. The addition of Er_2O_3 in the fuel is strategically utilized to reduce early excess reactivity by approximately 400 pcm and minimize local pin peaking factors during the initial stages of operation, ultimately depleting early enough to leave a negligible residual reactivity penalty after a burnup of 30 GWd/tU near the end of the cycle. [13].

Overall, the use of the TOP lattice produces a significant improvement in neutron economy, exceeding that of commercial PWRs, while providing a sufficiently negative MTC and minimizing the temperature defect.

2.2 ATOM Core Specifications and Performance

The ATOM core is designed with a thermal power of 450 MWth and is designed 69 TOP 17×17 FAs loading with an active height of 200 cm. Each FA contains 264 fuel rods loaded with the BA, 24 guide tubes, and one central tube (see Figure 4.1). Information about the reactor and its fuel characteristic are summarized in Table 2.1. The fuel inventory is reduced by about 14% compared to a core using the standard fuel radius without the TOP lattice [12]. To maximize neutron economy and optimize the core power profile, the ATOM SMR core employs an "in-then-out" fuel shuffling scheme for its two-batch design, while it utilizes a scattered shuffling scheme for the three-batch configuration to flatten the radial power distribution. This is shown in Figure 2.1. [12]

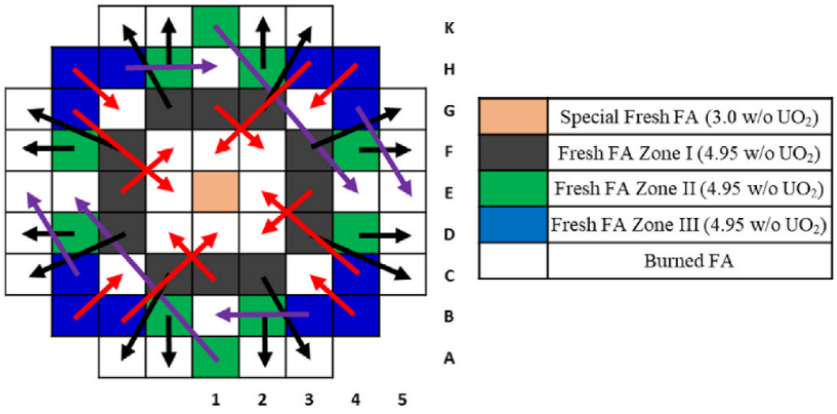


Figure 2.1: Arrangement of Zone I, II, and III of the ATOM core shown in black, green and blue, respectively. Arrows of different colors explain the two-batch reshuffling scheme. [12]

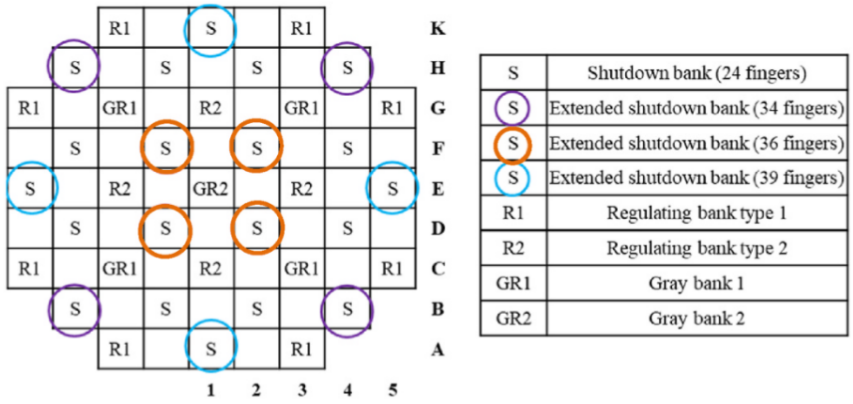


Figure 2.2: Arrangement of the shutdown (S), regulating (R1 and R2), and gray (GR1 and GR2) rods. The purple, orange, and blue coloured circles identify the extended shutdown rods, which increase the number of fingers to 34, 36, or 39, respectively. [12]

In SBF operation, achieving cold shutdown is difficult because the number of CRs is limited. This challenge is particularly significant when highly negative temperature coefficients result in a large temperature defect, defined as the reactivity difference between hot full-power and cold zero-power conditions. For this reason, the conventional checker-board CR pattern is difficult to apply, and the fraction of FAs with CR inserts must be substantially increased to ensure cold shutdown. As shown in Figure 2.2, the CR arrangement of the ATOM core consists of 20 shutdown control element assemblies, 12 regulating control element assemblies,

and five mechanical shim gray assemblies. The CR materials are defined in Table 2.1. Some shutdown control element assemblies are extended and contain either 34, 36, or 39 fingers by using positions in neighboring FAs to improve the cold shutdown margin. The gray rods are used to achieve core criticality without causing significant power distortion and the mechanical shim is intended to be very weak.

The ATOM core design features two fuel management schemes to minimize the burnup reactivity swing: a two-batch and a three-batch scheme. The ATOM SMR achieves a two-year fuel cycle by using a two-batch fuel management scheme with an in-then-out shuffling strategy, resulting in an irradiation cycle duration of approximately 725 effective full-power days. To manage the power distribution and mitigate radial peaking, the ATOM core is radially divided into three distinct zones shown in Figure 2.1. Zone I includes the innermost FA, while zone II and zone III are located in the outer core regions. Most fresh FAs are placed in the inner core positions, while once-burned assemblies are placed near the periphery to reduce neutron leakage. To improve neutron economy and to reduce central power peaking, some burned FAs are also placed in inner regions. The two-batch fuel management scheme is the one selected for this work. During refuelling, the number of inserted fresh FAs is 34, all using 4.95 % UO_2, which produces rotational symmetry in the core. The fresh FAs are divided radially into three zones, except for the central assembly.

2.3 BAs in SBF Operation

In conventional PWRs, excess reactivity is commonly controlled using BAs that are mixed into the fuel or coated onto the pellet surface to reduce fabrication costs. However, conventional BAs are designed to deplete relatively rapidly because they are intended to compensate excess reactivity over one fuel cycle. This happens because large-scale reactors generally have shorter operating cycles than the ATOM SMR: at refuelling, the depleted BAs are removed together with the discharged fuel, while a new batch of BAs is introduced with the fresh fuel. This is why, in conventional PWRs, the main excess reactivity is mostly controlled by soluble boron. In SBF operation, however, excess reactivity is mainly compensated by BAs, while the remaining reactivity is controlled by a group of weak CRs.

The SBF PWR concept was already proposed a long time ago but has not become practical mainly because of the lack of successful BA designs capable of maintaining sufficiently small excess reactivity without degrading neutronic and safety performance. To achieve a very small excess reactivity in the SBF ATOM core without compromising core performance, the design uses an innovative three-dimensional BA called the centrally shielded BA (CSBA) [12].

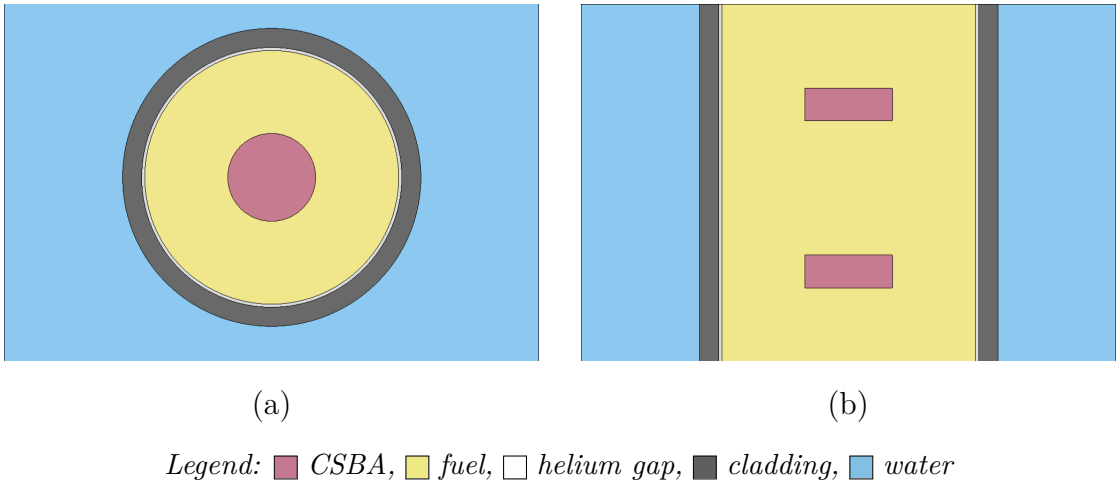


Figure 2.3: Cross section of the ATOM fuel pellet, as implemented in Serpent. Figure (a) shows the fuel rod from above, with the cylindrically shaped CSBA shown in pink, surrounded by uranium dioxide (yellow), a helium gap (white), fuel cladding (grey) and water (blue). Figure (b) shows the fuel rod from the side.

As seen in Figure 2.3, in the CSBA concept, a disk of gadolinia (Gd_2O_3), is placed in the central region of the fuel pellet. The effective stress at the interface between the CSBA disks and the fuel is acceptable, while the maximum temperature of the CSBA-loaded fuel pellet is comparable to that of a conventional pellet. The effect of asymmetric power distribution caused by neighboring fuel rods is also relatively small, corresponding to about 15 K in peak temperature, with little change in the resulting thermal expansion and stress [13].

In the ATOM SMR design, two cylindrical CSBAs per pellet are used and distributed uniformly along the fuel rod as. Gd_2O_3 is selected as the BA material because it is well established in PWR technology and is highly effective for controlling excess reactivity due to the large thermal neutron absorption cross sections of gadolinium isotopes such as Gd-155 and Gd-157. The two-batch fuel management scheme selected for this work, uses two cylindrical Gd_2O_3 CSBAs with 89% theoretical density in the monoclinic phase [12]. The cylindrical CSBA is shown to be highly effective in controlling the excess reactivity of SBF cores. The targeted irradiated scenario features two cycles, each being two-years long. The reactivity swing over the first two-years is estimated to be 1000 pcm [12].

The CSBA dimension depends on the zone of the reactor core, and the different configurations are given in Table 2.1. The largest cylindrical CSBA is placed in the inner zone (zone I) to reduce power peaking, while the smallest CSBA is loaded in the outer regions (zone II and III). The appropriate height-to-diameter ratio of the

cylindrical CSBA for the core is approximately 0.3–0.35. The central FA uses a unique cylindrical CSBA design to control the central peaking factor. In this work, zone II was used as a reference.

Table 2.1: Design Parameters and Material Specifications for the ATOM Core

Parameter	**Value**		
FA Dimensions [cm] [12]			
Fuel pellet radius	0.380		
Helium gap	0.00915		
Cladding thickness	0.0573		
Pin pitch [13]	1.2623		
FA pitch [13]	21.6038		
Core active height	200		
Control & Grey Rod Dimensions [cm] [12]			
CR radius	0.51674		
CR gap radius	0.52055		
CR clad radius	0.56754		
Grey rod radius	0.44000		
Grey rod clad thickness	0.12800		
CSBA Dimensions [12]	**Zone I** [mm]	**Zone II** [mm]	**Zone III** [mm]
CSBA Diameter	3.25	2.63	2.40
CSBA Height	0.98	0.92	0.84
Fuel & Core Materials [12]			
Fuel Pellets	UO_2 (4.95 % Enrichment, 95.5% TD)		
Erbia-enriched Fuel Pellets	UO_2 + 0.7% Er_2O_3		
Cladding [13]	Zircaloy-4 (composition from PNNL-15870, Rev. 1)		
Radial Reflectors	SS-304		
Coolant / Moderator	Light Water (0.706 g/cm^3 at 582.5 K)		
CSBA Material	Monoclinic Gd_2O_3 (89% TD)		
CR Materials [12]			
Shutdown Rods	90% B–10 B_4C		
Regulating Rods (Type 1)	50% B–10 B_4C		
Regulating Rods (Type 2)	Natural B–10 B_4C		
Grey Rod Material	Manganese		

Chapter 3

Nuclear Safeguards

Nuclear science and technology have significant potential to contribute to human health, economic development and social prosperity. At the same time, the same knowledge, materials and technologies may also provide the basis for the development of nuclear weapons through early detection. Nuclear safeguards are therefore an essential component of the international security system. The primary purpose of safeguards is to deter the diversion of nuclear material and the misuse of nuclear facilities or technology for the production of nuclear weapons. They consist of technical measures applied to nuclear material, facilities and activities, through which verification that States are respecting their agreements and that safeguarded nuclear material remains in peaceful use is possible.

The measures used for this purpose include nuclear material accountancy, on-site inspections, containment and surveillance (C/S), and the verification of facility design information. In practice, the IAEA is the international organization that verifies the correctness and completeness of the information declared by States through activities conducted at nuclear facilities and through the subsequent evaluation of safeguards-relevant information at its headquarters. The findings obtained from these activities provide the basis for drawing safeguards conclusions about the absence of diversion of nuclear material and misuse of facilities.

Nuclear material subject to IAEA safeguards includes both special fissionable material and source material. Special fissionable material comprises materials from which nuclear weapons or other nuclear explosive devices could readily be produced; some examples include ^{239}Pu, ^{233}U, and uranium enriched in the isotopes ^{235}U or ^{233}U. Source material includes natural uranium, depleted uranium and thorium. Although source material is not directly suitable for such devices, it can be transformed through technically demanding processing or irradiation into special fissionable material. For these reasons, source material is also subject to safeguards and verification. Most States are likely to possess at least some nuclear material within their territory, including material used for non-nuclear purposes.

Radioactive sources that do not contain uranium, plutonium or thorium are not subject to safeguards and therefore do not need to be reported to the IAEA under a safeguards agreement.[14]

Beyond their technical verification function, safeguards serve as confidence-building measures among States. They allow a State to demonstrate that its nuclear activities are being conducted exclusively for peaceful purposes and provide other States with corresponding assurance. The IAEA performs an essential verification role under the NPT. Safeguards also support the regional non-proliferation framework established by the five Nuclear-Weapon-Free-Zone treaties.[14]

As the number of nuclear facilities and the quantity of nuclear material subject to verification continue to grow, so does the safeguards workload. This increase is driven particularly by the continuous accumulation of SNF resulting from the ongoing operation of nuclear reactors. However, the resources available to the IAEA do not necessarily increase at the same rate. For this reason, it must continually seek improvements in the effectiveness and productivity of safeguards implementation. Enhanced cooperation with Member States and the maintenance of a qualified multidisciplinary workforce are therefore particularly important. [14]

This chapter provides a general overview of nuclear safeguards, including the role of the IAEA, the relevant legal framework and the practical implementation of safeguards. It then presents the principal nuclear verification methods, with particular emphasis on NDA measurements of SNF.

This focus is directly relevant to the present work because LF operation and non-constant irradiation histories affect the composition of SNF and may consequently influence the response of safeguards instrumentation. Particular attention is therefore given to NDA techniques, which are generally preferred for spent-fuel verification because they allow the fuel to be characterized without physically altering or consuming the measured item.

3.1 Safeguards Legal Framework

The legal framework for IAEA safeguards consists of a number of elements. These include the IAEA Statute, safeguards obligations under the NPT and treaties establishing nuclear-weapon-free zones (NWFZs), safeguards agreements concluded between individual States and the IAEA, together with their protocols and subsidiary arrangements, and the decisions of the IAEA Board of Governors. Some regions also have their own verification or control mechanisms, which may operate alongside and in cooperation with IAEA safeguards. In the following sections the most relevant agreements for the purpose of this work are explained, while Appendix B contains additional information on the topic.

3.1.1 Treaty on the Non-Proliferation of Nuclear Weapons

The NPT is the centrepiece of international efforts to prevent the further spread of nuclear weapons, promote cooperation in the peaceful uses of nuclear energy, and advance the goal of nuclear disarmament and general and complete disarmament. The Treaty was opened for signature in 1968 and entered into force on 5 March 1970. On 11 May 1995, it was extended indefinitely. With 191 States parties, it is the most widely adhered-to treaty in the fields of nuclear non-proliferation, peaceful uses of nuclear energy and nuclear disarmament. [14]

The NPT establishes a balance of rights and obligations and distinguishes between nuclear-weapon States (NWSs) and non-nuclear-weapon States (NNWSs). NWSs are defined as those that manufactured and exploded a nuclear weapon or other nuclear explosive device before 1 January 1967 [15]. These States are China, France, the Russian Federation, the United Kingdom and the United States of America. Under the Treaty, the NWS have committed not to transfer nuclear weapons or other nuclear explosive devices to any recipient and not to assist, encourage or induce a NNWS to manufacture or otherwise acquire such weapons or devices. NNWSs parties have committed not to manufacture or otherwise acquire nuclear weapons or other nuclear explosive devices.

Although the IAEA is not a party to the NPT, it is entrusted with important responsibilities under the Treaty. In particular, Article III assigns the Agency a specific verification role as the international safeguards inspectorate. Each NNWS party is required to conclude a comprehensive safeguards agreement with the IAEA. These agreements enable the Agency to verify the fulfillment of the State's obligations under the Treaty. By 31 December 2025, 183 NNWSs parties had brought the required CSAs into force, while three had not yet done so [15].

The IAEA therefore verifies the obligations assumed by NNWSs parties through the implementation of safeguards. At the same time, the Agency contributes to the objectives of Article IV of the Treaty by fostering international cooperation in the peaceful uses of nuclear energy in accordance with its Statute. [14]

The full text of the NPT articles cited above is reported in Appendix C, together with a description of NWFZs.

3.1.2 Safeguards Agreements

IAEA safeguards are implemented through legally binding agreements concluded between States and the Agency. In accordance with the IAEA Statute, a State accepts the application of safeguards by entering into such an agreement with the IAEA [16]. Three main types of safeguards agreements are used:

- CSA with NNWSs parties to the NPT;
- voluntary offer agreements (VOAs) with the NWSs to the NPT;

- item-specific safeguards agreements with States that are not parties to the NPT.

CSA constitute the majority of the safeguards agreements concluded by the IAEA. They apply primarily to NNWS parties to the NPT and to States parties to NWFZ treaties. As of 31 December 2025, CSAs had been concluded with 183 States.

Under a CSA, the IAEA has both the right and the obligation to apply safeguards to all nuclear material within the territory, jurisdiction or control of the State. The purpose of these activities is exclusively to verify that nuclear material is not diverted from peaceful uses to nuclear weapons or other nuclear explosive devices. [16, 14]

Approximately 100 States with CSAs have also concluded a small quantities protocol. Such a protocol may accompany a CSA and is available to States that possess minimal or no nuclear material and have no nuclear material located in a facility. [16, 14]

The five NWSs parties to the NPT have concluded VOAs with the IAEA. Under these agreements, each State identifies facilities that may be offered for the application of safeguards, while the IAEA selects the facilities at which safeguards will actually be implemented. Safeguards applied under a VOA are intended to verify that the nuclear material in the selected facilities remains in peaceful activities and is not withdrawn from safeguards, except under the conditions established by the agreement. [16, 14]

In India, Pakistan and Israel, which are not parties to the NPT, safeguards are implemented through item-specific agreements concluded with the IAEA. Unlike CSAs, these agreements apply only to the nuclear material, facilities and other items specifically identified within them.

The IAEA applies safeguards to these items to verify that they are not used for the manufacture of nuclear weapons or in support of any military purpose. The agreements also require the specified material, facilities and other items to be used exclusively for peaceful purposes and not for the manufacture of nuclear explosive devices.[16, 14]

Additional Protocol

The Additional Protocol (AP)is not a stand-alone safeguards agreement, but a separate legal instrument that supplements an existing agreement between a State and the IAEA. It cannot therefore exist independently of a safeguards agreement; once brought into force, the two instruments are applied and interpreted together. The AP provides the Agency with further verification tools and, particularly in States with CSAs, significantly strengthens its ability to verify that all nuclear material and activities are used exclusively for peaceful purposes. [17]

In May 1997, the IAEA Board of Governors approved the Model AP contained in INFCIRC/540 (Corrected) and requested the Director General to use it as the standard text for negotiating APs [14]. The model was designed for States with any of the three types of safeguards agreements. A State with a CSA that decides to conclude an AP must accept all the provisions of the model. States with VOAs or item-specific safeguards agreements may instead accept those measures that they are prepared to implement.

The AP expands both the information available to the IAEA and its rights of access within a State. While a CSA primarily concerns declared nuclear material and facilities, the AP provides the Agency with a broader view of nuclear fuel cycle related activities. The information supplied may cover the different stages of the nuclear fuel cycle and its research and development not involving nuclear material, the manufacture and export of specified nuclear-related equipment and materials, buildings located on nuclear sites, exempted nuclear material and the State's future plans for nuclear fuel cycle related activities. [17]

The principal strengthening measures provided by the AP include:

- broader declarations concerning nuclear material, facilities and nuclear-fuel-cycle-related activities;
- access to all buildings located on a nuclear site;
- information concerning uranium mines, uranium and thorium concentration plants, and certain forms of source material not reported under a CSA;
- information on nuclear-related manufacturing, imports and exports;
- access to relevant research and development activities that do not involve nuclear material;
- the collection of environmental samples beyond declared locations when considered necessary by the IAEA;
- simplified procedures for the designation of inspectors and the issuance of multiple-entry visas;
- the use of internationally established communication systems, including satellite communication. [18]

The Agency may conduct complementary access to locations covered by the AP. Such access may be used to confirm the absence of undeclared nuclear material or activities, to resolve a question or inconsistency concerning the correctness or completeness of the information supplied by the State, or to confirm the decommissioned status of a facility or a location outside facilities where nuclear material was previously used. When a question or inconsistency arises, the State is generally given an opportunity to provide clarification before complementary access is requested, unless a delay could undermine the purpose of the access. Because the AP requires States to provide a larger amount of detailed information, it also contains provisions requiring the Agency to strengthen the protection of confidential information. [14]

By providing a more complete picture of a State's nuclear material, activities, facilities, research, trade and future plans, the AP enhances the effectiveness and efficiency of safeguards implementation. For States with

3.2 Implementation of IAEA Safeguards

The purpose of safeguards implementation is to verify that States comply with the obligations established under their respective safeguards agreements with the IAEA. Safeguards are implemented through an annual cycle consisting of four fundamental processes: the collection and evaluation of safeguards-relevant information, the development of a safeguards approach for the State, the planning, conduct and evaluation of safeguards activities, and the drawing of safeguards conclusions. [14]

3.2.1 Collection and Evaluation of Safeguards-Relevant Information

Safeguards-relevant information is used in the implementation of IAEA safeguards and in the drawing of safeguards conclusions. The Agency collects and evaluates information from three principal sources: the State, safeguards activities conducted by the IAEA, and other open sources.

Information supplied by the State represents the majority of the information used in safeguards implementation and includes reports and declarations. Depending on the safeguards instruments in force, this may include nuclear material accountancy reports, advance notifications of nuclear material transfers, facility design information and information concerning nuclear and nuclear-related activities.

The IAEA also generates information through activities carried out in the field and at its headquarters. Such information may result from nuclear material accountancy verification, NDA measurements, sampling, C/S, design information verification, complementary access and flow-sheet verification.

Other relevant information may be obtained from open sources, including government documents, operator publications, scientific and technical literature and commercially available satellite imagery, as well as from third parties that provide information voluntarily.[14]

All available safeguards-relevant information is examined through the State evaluation process. The purpose of this process is to assess the consistency of the information available to the IAEA concerning the State. The Agency evaluates whether the information declared by the State is internally consistent and compatible with all other safeguards-relevant information independently collected or generated by the IAEA. Questions, anomalies or inconsistencies identified during this process are addressed through consultations with the State and, where necessary, through

follow-up activities. The results of the State evaluation provide the basis for determining the safeguards activities to be carried out and for drawing safeguards conclusions.[14]

3.2.2 Development of a State Level Safeguards Approach

The State-level safeguards approach is a customized approach developed for an individual State based on the evaluation of all safeguards-relevant information available about that State. It identifies the safeguards measures and activities required to achieve the State's relevant objectives [14]. The safeguards objectives applicable to a State depend on the scope of its safeguards agreement, they can be generic and technical. Generic safeguards objectives are established according to the type of agreement and are common to States with the same type of agreement.

Under a CSA, the generic objectives include detecting the diversion of declared nuclear material at declared facilities or locations outside facilities, detecting undeclared production or processing of nuclear material at such locations, and detecting undeclared nuclear material or activities in the State as a whole. Under an item-specific safeguards agreement, the objectives concern the detection of diversion of the nuclear material subject to safeguards and the misuse of facilities or other safeguarded items. Under a VOA, the objective is to detect any undeclared withdrawal of nuclear material from the selected facilities, or parts of facilities, to which safeguards are applied.[14]

In addition to the generic objectives, the IAEA establishes technical objectives for each State. These objectives are intended to address the generic safeguards objectives and to guide the planning, conduct and evaluation of safeguards activities. Unlike generic objectives, technical objectives may differ between States because they take into account the characteristics of the State's nuclear fuel cycle, including the types, forms and quantities of nuclear material, the facilities and technical capabilities present, and the processes through which nuclear material is produced or used.

For States with CSA, technical objectives may be established through an acquisition path analysis. This is a structured method used to examine the technically plausible paths through which nuclear material suitable for use in a nuclear weapon or other nuclear explosive device could be acquired. Each path consists of the steps required to obtain the material and process it into a suitable form. Technical objectives are then established to detect the individual steps along each path.[14]

3.2.3 Planning, Conduct and Evaluation of Safeguards Activities

On the basis of the State-level safeguards approach, the IAEA prepares an annual implementation plan specifying the activities to be conducted in the field and at the Agency's headquarters. The safeguards measures selected for a State are intended to address the technical objectives identified through the State-level approach.

Once a safeguards activity has been completed, the Agency evaluates the extent to which it has achieved its technical objective. Any questions, inconsistencies or anomalies requiring further examination are identified and may be addressed through additional activities or incorporated into an updated implementation plan.[14]

Although the specific activities vary according to the State and its safeguards agreement, nuclear material accountancy and verification in the field remain central to safeguards implementation. Field activities are conducted to verify State reports and declarations and may include inspections, examination of facility records and supporting documentation, nuclear material measurements, item counting, design information verification, environmental sampling, C/S, and other verification measures. [14]

3.2.4 Drawing Safeguards Conclusions

Safeguards conclusions are the final product of the annual implementation cycle. They are based on the IAEA's independent verification activities, findings and evaluation of all available safeguards-relevant information. Before a conclusion is drawn, the Agency must have carried out a sufficient level of safeguards activities, completed a comprehensive evaluation of the relevant information and addressed any anomalies, questions or inconsistencies identified during the year. The type of conclusion that may be drawn depends on the safeguards agreement and other safeguards instruments in force.

For a State with both a CSA and an AP, the IAEA may conclude that all nuclear material remained in peaceful activities when its evaluations identify neither diversion of declared material nor indications of undeclared nuclear material or activities. When the evaluation concerning undeclared material and activities remains incomplete, the conclusion is limited to the finding that declared nuclear material remained in peaceful activities.

For States with a CSA but without an AP, the conclusion concerns whether declared nuclear material remained in peaceful activities.

Under item-specific agreements, the Agency determines whether the nuclear material, facilities and other safeguarded items remained in peaceful use.

Under VOAs, it assesses whether safeguarded nuclear material in selected

facilities was withdrawn only as permitted by the agreement and otherwise remained in peaceful activities. [14]

The results of safeguards activities and the associated information are assessed by a State Evaluation Group and documented in a State Evaluation Report. Any anomaly or inconsistency between a State's declarations and the Agency's findings is first addressed through requests for clarification and consultations with the State. Unresolved issues are subsequently considered during the internal review of the report and may be reflected in the Director General's annual Safeguards Implementation Report to the Board of Governors. The Board then determines whether the findings constitute non-compliance and, where appropriate, decides whether and how the matter should be reported to the United Nations Security Council. Following its consideration of the report, the Board authorizes the publication of the Safeguards Statement, which presents the Agency's annual safeguards findings and conclusions. [14]

3.3 Verification of Nuclear Material

On the basis of the safeguards approach developed for each State, the IAEA prepares an annual implementation plan specifying the safeguards activities to be conducted during the year.

In-field safeguards activities enable inspectors to verify nuclear material, facility design and the absence of undeclared activities. During inspections, facility records are compared with information reported by the State, and inspectors verify that declared nuclear material is present in the stated quantities and forms. Design information is checked against observations at the facility to confirm that the declared design is correct and that the facility has not been misused. Environmental samples may also be collected to detect traces of undeclared nuclear activities. Nuclear material inventories are verified through item counting, weighing, NDA and destructive analysis. Seals, cameras, detectors and unattended monitoring systems maintain continuity of knowledge between inspections.

Safeguards inspections vary in scope and duration according to the facility and the activities being verified. They may be routine or unannounced and may include complementary access, design information verification or physical inventory verification. At nuclear power plants, inspectors verify spent fuel stored in cooling ponds, sometimes using improved Cerenkov viewing devices (CVDs). The observed inventory is compared with the information declared by the operator and reported by the State. Inspectors may also attend significant operations, such as reactor refueling.

Containement and Surveillance

The physical integrity of the containment structure is a fundamental prerequisite for effective safeguards. In international safeguards, containment serves as a qualitative, auxiliary measure that complements quantitative nuclear material accountancy by defining material balance areas and concentrating measurement efforts [19]. This involves securing physical boundaries such as storage containers, instrument cabinets housing sensitive equipment, and data-carrying conduits. Traditionally, verifying containment has relied on manual, visual inspections by inspectors, however the IAEA periodically remeasures a small randomized percentage of material under C/S to ensure no diversion has occurred, while actively researching more advanced, less intrusive techniques.

Surveillance and monitoring systems operate continuously to improve the effectiveness and efficiency of safeguards by maintaining continuity of knowledge over nuclear material and facilities between inspections. Cameras, radiation detectors and other sensors can record activities that extend over long periods, such as reactor refuelling. The collected data may be transmitted securely to IAEA headquarters or reviewed by inspectors during subsequent visits to verify that operations were conducted as declared. Seals are also used to preserve the integrity of nuclear material, equipment and containment systems. [14]

Single-use metal cap seals have been employed by the IAEA for more than 30 years. Each seal is individually numbered and contains unique markings on its internal surfaces, which are recorded before it is issued to inspectors. During inspections, seals may be replaced and returned to IAEA Headquarters, where their authenticity and effectiveness are verified by comparing these markings with the original records. [14]

3.4 NDA Measurements on SNF

An important part of a State's nuclear material verification is represented by SNF verification, which is commonly performed at nuclear power reactors and interim storage facilities. The declared information to verify typically includes fissile masses, IE, burnup (BU), and cooling time (CT). Because destructive analysis is generally impractical for SNF assemblies, verification relies primarily on NDA measurements of their radiation signatures. These measurements typically detect gamma rays or neutrons, as SNF contains numerous radioactive fission products (FPs) and transuranic elements (TRUs), and may emit more than 10^8 neutrons and 10^{15} gamma rays per second [20]. These emissions are related to the radionuclide inventory of the spent fuel and therefore carry information about the irradiation history and fuel composition. For this reason, the passive emission of gamma rays and neutrons can be used verify that the properties of the SNF are in line with what

can be expected from a nuclear fuel undergoing irradiation in a power producing reactor, and that the operator declared information is complete and correct.

During reactor operation, energy is produced primarily through fission of ^{235}U, ^{239}Pu, and ^{241}Pu, with a smaller contribution from fast fission of ^{238}U. Ideally, their presence and quantities would be measured directly. In practice, however, their radiation signatures are difficult to distinguish from the intense gamma-ray and neutron fields generated by the many FPs and actinides present in SNF. Consequently, measured gamma-ray and neutron signatures generally provide indirect evidence of the special nuclear material contained within the fuel [20]. Innovative NDA technologies should require neither calibration curves nor detailed knowledge of the fuel's operating history, while remaining simple and effective. Because each fission event produces two principal fission fragments, which subsequently decay into a range of FPs, the resulting FP inventory and radiation signatures can be used to infer properties of the SNF and support verification of its declared composition [20].

In particular, FPs produced by ^{239}Pu have a slightly higher average atomic mass compared to ^{235}U. An example of this phenomenon is the fission yields of ^{106}Ru, which are about 11 times higher for ^{239}Pu than for ^{235}U. On the other hand, ^{137}Cs has nearly identical fission yields from both ^{239}Pu and ^{235}U, making it useful for determining the total number of fissions and the discharge BU [20]. Additionally, most FPs initially undergo beta decay, emitting gamma radiation that can be measured. The ratio between the gamma radiation from ^{106}Ru and ^{137}Cs can therefore provide information on the relative contribution of ^{239}Pu to total fissions [20].

Besides the production mechanism of FPs, the production of TRUs is also important. During operation, uranium absorbs neutrons, creating TRUs such as ^{238}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm. These nuclides emit neutrons through spontaneous fission or (α, n) reactions made possible by the large amounts of oxygen in the reactor [20]. For this reason, the passive neutron signal is dominated by TRUs and, in particular, by ^{244}Cm, whose neutron yield exceeds that of plutonium isotopes by about two orders of magnitude [20].

Detecting both gamma and neutron radiation emitted by SNF is essential to obtain information on its composition. It is important to note that gamma rays are also produced by activation products and that the large number of FPs and TRUs contributes to a complex radiation field, making signal interpretation challenging. However, gamma spectroscopy, which is able to produce energy-resolved information about detected gamma-rays, can give isotope-specific information, and is a feasible technique. Relevant radionuclides in SNF verification are listed are listed in Table 3.1. The Table summarizes their half-lives, cumulative fission yields for ^{235}U and ^{239}Pu, principal gamma-ray energies, and corresponding branching ratios. The cumulative fission yield represents the total probability of producing an isotope

either directly by fission or indirectly through the radioactive decay of precursor nuclides. Gamma-ray energy is the characteristic photon energy emitted during nuclear de-excitation, while the branching ratio is the percentage of decays that produce a specific gamma-ray emission. These values are given for both FPs, such as ^{95}Zr, ^{95}Nb, ^{103}Ru, ^{134}Cs, ^{137}Cs, and ^{154}Eu, and activation products.

Table 3.1: Half-lives, fission yields, and main gamma-ray emissions of selected FPs. [20]

FP isotopes	Half-life	Fission yield in ^{235}U (%)	Fission yield in ^{239}Pu (%)	Gamma-ray energy (keV)	Branching ratio (%)
^{95}Zr	64.0 days	6.50	4.89	724.2 756.7	43.1 54.6
^{95}Nb	35.0 days	6.50	4.89	765.8	99.8
^{103}Ru	39.4 days	3.04	6.95	497.1 610.3	86.4 5.4
^{106}Ru–Rh	366.4 days	0.40	4.28	622.2 1050.5	9.8 1.6
^{134}Cs	2.06 years	1.27×10^{-5}[a]	9.89×10^{-4}[a]	604.7 795.8 801.8 1167.9 1365.1	97.6 85.4 8.7 1.8 3.0
^{137}Cs	30.17 years	6.22	6.69	661.6	85.1
^{140}Ba–La	12.7 days	—	—	1596.0	—
^{144}Ce–Pr	284.5 days	5.48	3.74	696.5 1489.2 2185.6	1.3 0.3 0.7
^{154}Eu	8.5 years	2.69×10^{-6}[a]	9.22×10^{-5}[a]	996.3 1004.8 1274.4	10.3 17.4 35.5

[a]The values reported for ^{154}Eu and ^{134}Cs account only for their direct production from fission. In practice, both isotopes are produced primarily through neutron absorption. For PWR fuel irradiated to 25 GWd/tU, the accumulated fission yields of ^{154}Eu and ^{134}Cs were calculated to be 0.15% and 0.46%, respectively.

Similarly, the main neutron-emitting isotopes are summarized in Tables 3.3 and 3.2. Table 3.3 presents the half-lives, neutron yields, and average neutron multiplicities for spontaneous and thermally neutron induced fission. The neutron

yield is the number of neutrons emitted per second per gram of isotope, whereas the neutron multiplicity (ν) is the average number of neutrons released in each fission event. Table 3.2 summarizes neutron production through (α,n) reactions, including the alpha-decay half-life, alpha yield, average alpha-particle energy, and material-dependent (α,n) yields. The alpha yield is the number of alpha particles emitted per second per gram, while the (α,n) yield represents the resulting neutron-emission rate when these alpha particles interact with light nuclei in matrices such as oxides and fluorides. [20]

Table 3.2: Half-lives, alpha-decay properties, and (α, n) yields of selected nuclides. [20]

Isotope	Half-life [years]	α-decay half-life [years]	α yield [α/s g]	Average α energy [MeV]	(α, n) yield in oxide [n/s g]
^{232}Th	1.40×10^{10}	1.40×10^{10}	4×10^{3}	4.00	2.2×10^{-5}
^{232}U	68.9	68.9	8.0×10^{11}	5.30	1.49×10^{4}
^{233}U	1.59×10^{5}	1.59×10^{5}	3.57×10^{8}	4.82	4.8
^{234}U	2.45×10^{5}	2.45×10^{5}	2.3×10^{8}	4.76	3.0
^{235}U	7.04×10^{8}	7.04×10^{8}	8.0×10^{4}	4.40	7.1×10^{-4}
^{236}U	2.34×10^{7}	2.34×10^{7}	2.4×10^{6}	4.48	2.4×10^{-2}
^{238}U	4.47×10^{9}	4.47×10^{9}	1.2×10^{4}	4.19	8.3×10^{-5}
^{237}Np	2.14×10^{6}	2.14×10^{6}	2.6×10^{7}	4.77	3.4×10^{-1}
^{238}Pu	87.74	87.74	6.3×10^{11}	5.49	1.34×10^{4}
^{239}Pu	2.41×10^{4}	2.41×10^{4}	2.3×10^{9}	5.15	3.81×10^{1}
^{240}Pu	6.56×10^{3}	6.56×10^{3}	8.4×10^{9}	5.15	1.41×10^{2}
^{241}Pu	14.329	5.90×10^{5}	9.3×10^{7}	4.89	1.3
^{242}Pu	3.75×10^{5}	3.76×10^{5}	1.5×10^{8}	4.90	2.0
^{241}Am	432.7	432.7	1.3×10^{11}	5.48	2.69×10^{3}
^{242}Cm	163 days	163 days	1.2×10^{14}	6.10	3.76×10^{6}
^{244}Cm	18.1	18.1	3.0×10^{12}	5.80	7.73×10^{4}
^{249}Bk	330 days	6.1×10^{4}	8.8×10^{8}	5.40	1.8×10^{1}
^{252}Cf	2.645	2.731	1.9×10^{13}	6.11	6.0×10^{5}

The most common approaches for the characterization of SNF are based on either gamma-ray or neutron measurements. Gamma-ray-based methods can indirectly provide information on key fuel parameters such as BU and CT, and are widely

Table 3.3: Half-lives, spontaneous-fission properties, and neutron multiplicities (ν) of selected isotopes. [20]

Isotope	Z[b]	N[b]	Total half-life	SF[c] half-life [years]	SF[c] yield [n/s g]	ν_{SF} [c]	Induced ν_{TF} [c]
^{232}Th	90	142	1.40×10^{10}	1.2×10^{21}	1.01×10^{-7}	2.13	1.9
^{232}U	92	140	68.9	$> 6.8 \times 10^{15}$	1.43×10^{-2}	1.71	3.13
^{233}U	92	141	1.59×10^{5}	$> 2.7 \times 10^{17}$	3.70×10^{-4}	1.76	2.4
^{234}U	92	142	2.45×10^{5}	1.5×10^{16}	6.82×10^{-3}	1.81	2.4
^{235}U	92	143	7.04×10^{8}	1×10^{19}	1.05×10^{-5}	1.86	2.41
^{236}U	92	144	2.34×10^{7}	2.5×10^{16}	4.24×10^{-3}	1.89	2.2
^{238}U	92	146	4.47×10^{9}	8.20×10^{15}	1.34×10^{-2}	1.98	2.3
^{237}Np	93	144	2.14×10^{6}	$> 1.0 \times 10^{18}$	1.14×10^{-4}	2.05	2.70
^{238}Pu	94	144	87.74	4.75×10^{10}	2.56×10^{3}	2.19	2.9
^{239}Pu	94	145	2.41×10^{4}	8×10^{15}	1.49×10^{-2}	2.16	2.88
^{240}Pu	94	146	6.56×10^{3}	1.14×10^{11}	1.04×10^{3}[a]	2.154[a]	2.8
^{241}Pu	94	147	14.4	$< 6 \times 10^{16}$	2.06×10^{-3}	2.25	2.8
^{242}Pu	94	148	3.75×10^{5}	6.77×10^{10}	1.74×10^{3}	2.149	2.81
^{241}Am	95	146	432.7	1.2×10^{14}	1.14	2.5	3.09
^{242}Cm	96	146	163 days	7×10^{6}	1.98×10^{7}	2.54	3.44
^{244}Cm	96	148	18.1	1.32×10^{7}	1.11×10^{7}	2.71	3.46
^{249}Bk	97	152	320 days	1.8×10^{9}	1.0×10^{5}	3.40	3.7
^{252}Cf	98	154	2.645	85.5	2.29×10^{12}	3.757	4.06

[a] Considerable uncertainty.
[b] Z = number of protons, N = number of neutron.
[c] SF = Spontaneous-fission, TF = Thermal fission.

used for quantitative characterization of SNF. Neutron-based methods, on the other hand, are simple to implement for gross neutron counting and provide a more uniform signal over the entire radial extent of the FA, making them particularly suitable for integral measurements and BU estimations.

3.4.1 Gamma-Ray Based Techniques

Gamma-ray-based techniques characterize SNF by exploiting different features of its photon emission, including its total intensity, energy spectrum, and spatial

distribution. Total gamma-ray activity measurements provide an integral indication of fuel irradiation, whereas gamma-ray spectroscopy identifies specific radionuclides and can therefore provide information on parameters such as BU, IE, and CT. Spatially resolved measurements additionally enable the internal structure of the FA to be assessed. These complementary measurement principles form the basis of several instruments used for safeguards verification, ranging from simple attribute testers to systems capable of detecting missing fuel pins.

The total gamma-ray activity of irradiated fuel is defined as the sum of the activities from all FPs and TRUs. For CTs greater than approximately one year, the total gamma-ray activity is roughly proportional to burnup. [20]. The measurement can be performed when the FA is raised above the storage rack, on the other hand this increases intrusiveness. This measurement can be carried out using standard total gamma detectors, such as ionization chambers, with an accuracy of the order of $\sim 10\%$ [20].

Gamma-ray spectroscopy provides more detailed information on fuel parameters such as BU, IE, CT, and the concentration of individual FPs. The measured gamma-ray activity of a FP is proportional to the number of fissions that occurred during irradiation only if the following conditions are satisfied:

- the FP has similar fission yields for uranium and plutonium fissioning nuclides,
- the neutron capture cross section of the FP is sufficiently low,
- the half-life of the FP is long compared to the irradiation time,
- the emitted gamma rays have sufficiently high energy to escape the fuel pin.

If these conditions are met, the gamma-ray activity allows the determination of BU [20]. Because individual isotopes accumulate at different rates, measuring a diverse range of isotopes provides complementary information about the fuel and enables a more comprehensive assessment of its properties.

Performing gamma-ray spectroscopy on SNF is challenging due to the intense radiation field generated by FPs. Among many technologies available at the IAEA for safeguards verification, some examples are the Spent Fuel Attribute Tester (SFAT) and the Irradiated Fuel Attribute Tester (IRAT). The SFAT employs a shielded, low-resolution detector to verify the presence of SNF through detection of ^{137}Cs gamma rays using a collimated geometry. The IRAT system uses a compact cadmium zinc telluride (CZT) detector to perform side measurements of FAs and requires movement of the assembly.[20]

Partial defect verification is required before difficult to access storage of SNF. The most common instruments available for this are Fork Detectors (FDETs) and the Passive Gamma Emission Tomography (PGET). The PGET is an advanced technique used to reconstruct the spatial distribution of gamma emission within a FA and is the only one able to do pin-level verification. It allows simultaneous measurement of neutron emission, and tomographic imaging, making it the only technique capable of detecting missing fuel pins.[20]

3.4.2 Neutron-Based Techniques

The most relevant neutron-based approaches for SNF characterization include total neutron activity measurements, and neutron coincidence counting. [20]

Total neutron activity corresponds to the total neutron emission rate from irradiated fuel and can be used as an indicator of BU. Compared to gamma-ray measurements, neutron signals originate from the fuel itself to a larger extent and are less affected by attenuation within the assembly, allowing a more uniform response across the fuel pins. However, a key limitation is that the neutron emission rate is only indirectly related to BU, as it depends on the buildup of TRUs and their ability to emit neutrons in primarily spontaneous fission or (α,n) reactions. [20]

The principal neutron emitters in typical PWR spent fuel include ^{244}Cm, ^{242}Cm, ^{240}Pu, ^{238}Pu, and ^{242}Pu, with curium isotopes usually dominating the neutron emission. Due to the relatively short half-lives of ^{242}Cm and ^{244}Cm, accurate knowledge of CT is essential for correct interpretation of the neutron signal. The neutron emission rate is also affected by non-continuous irradiation histories. During shutdown periods, ^{241}Pu decays into ^{241}Am, which can subsequently capture neutrons when irradiation resumes, leading to increased production of ^{242}Cm and therefore higher neutron emission rates.[20]

The FDET is the most commonly used system that measures both neutron and gamma radiation from spent FAs. It consists of two detector arms containing ionization chambers for gamma-ray measurements and fission chambers for neutron detection, allowing simultaneous measurements on opposite sides of the assembly. Initial data analysis is performed on-site, while detailed verification is conducted using dedicated analysis software.[20]

3.4.3 Other Techniques

Additional physical phenomena can be exploited to estimate SNF parameters, and several devices have been developed for this purpose. These include CVDs and curium-to-plutonium ratios.[20]

Cerenkov radiation is emitted when a charged particle travels through a medium with a velocity exceeding the phase velocity of light in that medium. Irradiated FAs produce beta particles, gamma rays, and neutrons, all of which can generate Cerenkov light either directly or indirectly. The dominant contribution arises from high-energy gamma rays, which produce electrons through interactions such as Compton scattering and pair production.

CVDs are used for nondestructive verification of SNF. The two most commonly used systems are the Improved Cerenkov Viewing Device (ICVD) and the Digital Cerenkov Viewing Device (DCVD). The ICVD is a handheld instrument that filters ambient light and enhances near-ultraviolet radiation, allowing verification

of the presence or absence of spent fuel. The DCVD, on the other hand, provides quantitative measurements by mapping the spatial distribution of Cerenkov light and comparing it with expected patterns, enabling the detection of missing fuel pins within an assembly.[20]

The curium-to-plutonium ratio is useful because the concentration of curium increases approximately as a power function of BU, while plutonium increases more linearly. As a result, their ratio can be used to estimate BU. In safeguards measurements of SNF, the curium contribution is commonly assessed using a FDET, whose passive neutron signal is dominated by the spontaneous fission of ^{244}Cm. The plutonium content must be obtained from complementary measurements, operator data, or depletion calculations. [20]

Chapter 4

Methodology

As previously mentioned, this work aims at studying the LF implications of the ATOM core on safeguards relevant parameters such as the composition of the SNF. The ATOM SMR was previously described in Section 2, while the methodology used to carry out this task is the follwing. Firstly, a representative ATOM FA model was implemented in Serpent. Secondly, the irradiation of the FA under two different scenarios was modelled. The first irradiation scenario assumed BL operation of the reactor, and the second scenario was seasonal LF operation assuming power variations in the range of 50-100%. Lastly, a study was made to specifically investigate the CR impact, to draw conclusions on its impact on the radionuclide inventory of the SNF.

4.1 Computational Tool: Serpent2

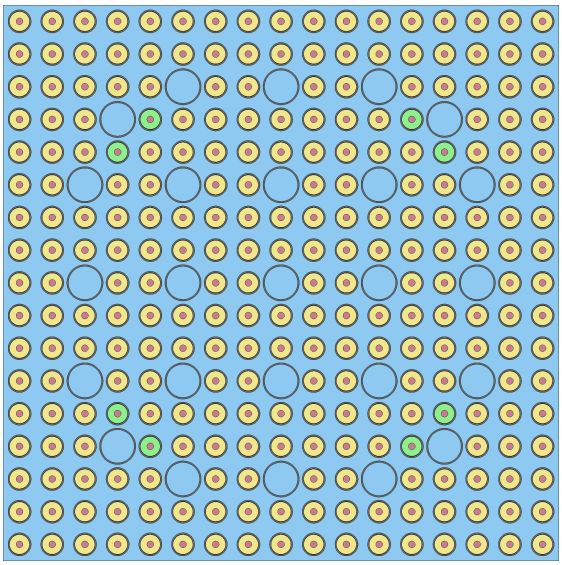
Serpent is a Monte Carlo neutron and photon transport code, used for various reactor physics, fusion neutronics, and radiation transport applications. The code is developed as part of the Kraken multi-physics computational framework at VTT Technical Research Centre of Finland [21]. In this work, Serpent2 was used to model the fuel and its irradiation during operation. Serpent2 includes extensive depletion-simulation capabilities, where the evolution of the radionuclide composition of fuel is determined by combining Monte Carlo simulations of the reactor core neutron fluence rate with the deterministic Bateman equations.

After discharge from the reactor, the neutron fluence rate is zero. As a result, the Bateman equations depend only on radioactive decay. Using this methodology, the radionuclide composition of the fuel can be estimated at different burnups and at different CTs after discharge [22].

4.2 Reference FA Model

A FA model representative of the ATOM core, zone II, was developed (see Table 2.1). The model is based on the standard ATOM TOP lattice 17×17 FA and is shown in Figure 4.1. This zone was selected because it was believed to be representative of the intermediate core region. The result of the implementation is an infinite lattice of infinitely long FAs (periodic boundary conditions in all directions) relevant for FAs in non-peripheral positions.

The fuel materials used in the model, pellet dimensions and FA parameters are reported in Table 2.1 [12].



Legend: *CSBA,* *fuel* *fuel with* Er_2O_3, *cladding,* *water*

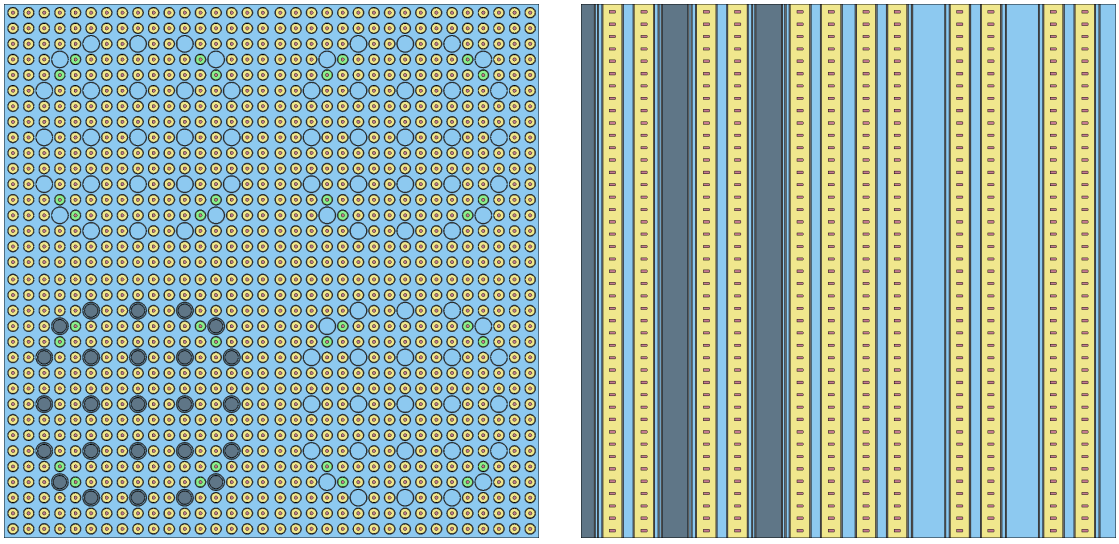
Figure 4.1: The ATOM FA cross section as implemented in Serpent shown from above.

4.3 Reference Model with CR Insertion

To investigate the effect of CR insertion, an additional four-assembly model was developed. In the reference ATOM core, 17 of the 69 assemblies are designated for control-rod insertion when the CRs are in use. To approximately preserve this ratio in an infinitely reflected simplified model, a four-assembly configuration was adopted, with one assembly containing fully inserted CRs. This can be seen in Figure 4.2.

The reference assembly model is the same assumed for the previous section. The

inserted rods correspond to type 2 regulating rods, selected because their regulating power is intermediate compared with the other CR types. The geometrical and material properties of the CRs are reported in Table 2.1. Periodic boundary conditions were applied to the model. This represents an idealized, infinitely repeating lattice and therefore does not fully reproduce the boundary effects or spatial heterogeneity of the actual core configuration.



(a) (b)

Legend: *CSBA,* *fuel* *fuel with* Er_2O_3, *cladding,* *CR,* *water*

Figure 4.2: Cross section of four ATOM FAs, one of which with inserted CR, as implemented in Serpent. Figure (a) shows the FAs from above. Figure (b) shows the FAs from the side.

4.4 Irradiation Scenarios

Two irradiation scenarios were considered in this work: a BL scenario and a seasonal LF scenario. The LF scenario was implemented keeping the same number of full-power days as the BL. Due to the reduced-power periods included in the LF profile, the final burnup obtained in the LF case was slightly lower than the initial BL reference value. For this reason, a new reference burnup of 41.4 GWd/tU was adopted for the BL calculation, allowing a direct comparison between the BL and LF results.

4.4.1 Reference Full-Power Scenario

The BL scenario represents the reference operating scenario used for comparison with the LF case. The irradiation history consists of an initial two-year period of operation at full power ($450MW_{th}$), followed by a 30-day outage designated for refueling and maintenance while the reactor power is set to zero. After the outage, the reactor operates again at full power until the reference discharge BU of 41.4 GWd/tU is reached. This happens after 495 days including outage. A CT of 10 years after discharge was considered to account for a prolonged period during which spent fuel may remain in the reactor spent fuel pool before transfer to another storage facility. This provides a conservative case for evaluating the radionuclide composition of fuel that could remain accessible in the cooling pond for an extended period after discharge.

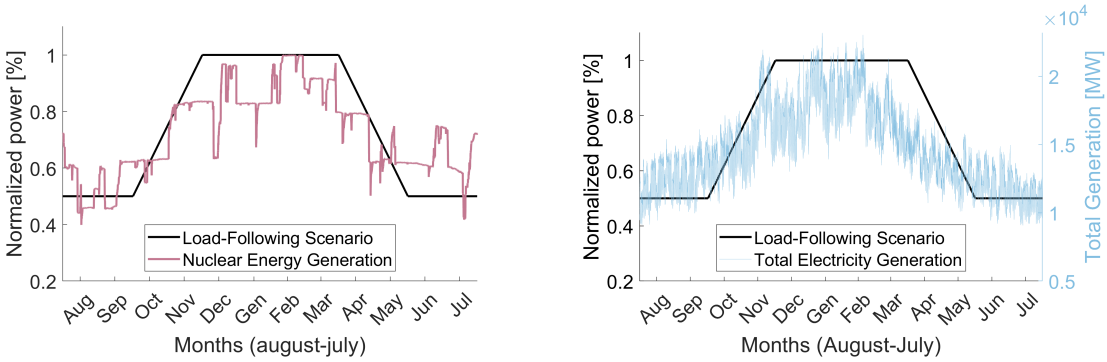
4.4.2 Seasonal LF Scenario

For this work, a seasonal LF scenario, suitable for the Swedish context was developed using data from Energy-Charts website by Fraunhofer Institute for Solar Energy Systems [23] and the International Energy Agency Energy Statistics Data Browser [24]. The scenario is shown in black in Figure 4.3 was adapted from Swedish electricity generation data, considering the total electricity produced by nuclear power in Sweden in 2025 (Figure 4.3 (a) in pink), its trend over the year, and the overall electricity generation profile (Figure 4.3 (b) in blue).

The trend was normalized so that the scenario could be adapted to different reactors. In this work, the power varies between 50% and 100% of nominal power, corresponding respectively to $225MW_{th}$ and $450MW_{th}$. The reactor is assumed to operate at 100% power for four months, at 50% power for four months, and the remaining four months are split in two equal intervals and used for the ramping up and down in power. The transition between 50% and 100% of nominal power is assumed to be linear, with an average ramp rate of approximately 3.4% of power per day. The same is assumed for the transition from 100% to 50%. The resulting trend is shown in Figure 4.3.

4.4.3 Implementation in Serpent2

The power profiles for the BL and LF operating scenarios are shown in Figure 4.4. In the BL scenario, the reactor operates at constant power for four years, except for the intermediate refueling outage. In the LF scenario, the seasonal power profile described in Section 4.4.2 is repeated over five years until the same final BU as in the BL case is reached. The power variations are implemented in Serpent by changing the prescribed power density over successive depletion intervals, following the procedure described in Reference [25]. The refueling outage is placed after two



(a) (b)

Figure 4.3: Power trends over one year of the LF scenario as implemented in this work, based on information from [23] [24]. Both figures are used to compare the power trend of the LF scenario (in black). Figure (a) compares it with the trend of normalized nuclear energy generation in Sweden in 2025. Figure (b) compares it with the trend of the total electricity generation in Sweden in 2025.

years in both scenarios. Consequently, the lower average power of the LF scenario results in a longer irradiation period to achieve the same final BU.

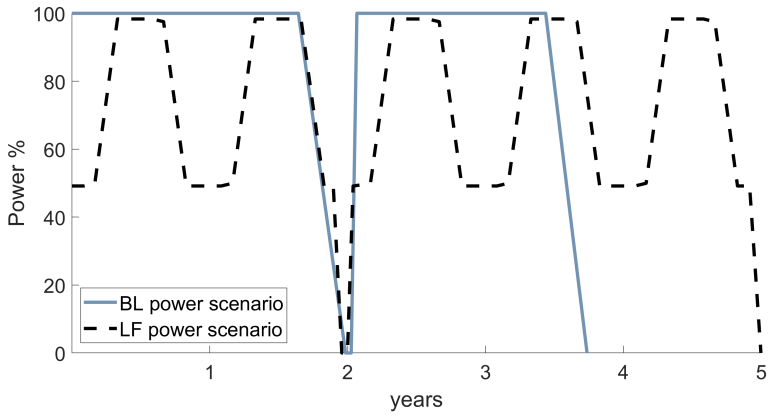


Figure 4.4: Trend of the power as a percentage of the nominal one of $450 MW_{th}$ for the two irradiation scenarios, including outage. The end of the x-axis shows the resulting time spent by the fuel inside the reactor.

Chapter 5

Results and Analysis

This chapter presents the results obtained from the Serpent2 depletion calculations performed using the ATOM FA models. The objectives are to evaluate the impact of LF operation on SNF composition and on quantities relevant for safeguards verification, and to study the effect of CR insertion. In particular, the analysis focuses on differences in fissile material inventories, plutonium vectors, minor actinides, and selected radionuclides relevant for NDA verification of SNF.

In this work, different types of safeguards-relevant signatures are considered. These include FPs and TRUs masses per axial length obtained from Serpent2, which are representative of passive gamma-ray emissions associated with selected FPs, and passive neutron emissions associated with selected TRUs.

In this analysis, radionuclide masses per axial length of the model are used as observables. The axial length is the height of a fuel pellet (1 cm). In practice, the directly detectable quantity by an NDA instrument is typically the number of counts recorded by a detector. For a well-characterized detector system with known efficiency, radionuclide activities can be inferred from instruments with (well-resolved) spectroscopic ability. For a given radionuclide, activity is proportional to the number of atoms present and therefore to its mass, through the radioactive decay constant. For this reason, radionuclide masses per axial length can be used to compare the evolution of selected fuel signatures under different operating conditions. This choice allows the analysis to focus on changes in the radionuclide inventory, without modeling a specific detector response. It also reduces the computational effort that would be required to simulate detector counts directly. The use of masses is therefore suitable for comparing the same radionuclides between BL and LF operation.

Only selected radionuclides are included in the analysis. The selection is based on previous studies showing that a limited number of radionuclides provide most of the relevant information for the estimation of IE, BU, and CT. In order to be useful as safeguards signatures, radionuclides should have sufficiently long half-lives

to remain detectable after discharge from the reactor core, they should be produced in abundant quantities, have sufficiently high gamma emission probabilities and gamma-ray energies to allow detection of radiation by an instrument in reasonable time. The contribution of the selected radionuclides to the total activity is also evaluated in order to assess their relevance as safeguards indicators. The gamma-emitting FP considered in the analysis are ^{95}Nb, ^{95}Zr, ^{106}Ru, ^{137}Cs, ^{134}Cs, ^{141}Ce, ^{144}Ce, ^{154}Eu, and ^{125}Sb [26] [22]. Their build-up during reactor operation is analyzed in order to understand their dependence on the irradiation conditions. The neutron-emitting TRUs considered in this work are ^{242}Cm, ^{238}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, and ^{244}Cm [26] [22]. These radionuclides are relevant because the total neutron emission from SNF is mainly associated with spontaneous fission in a limited number of actinides. Their inventories are therefore important for understanding the neutron source term of the spent fuel.

The results presented in this chapter include comparisons between BL operation and seasonal LF operation for the the single-assembly model, followed by results from the four-assembly model with CR insertion. Depletion results are also analyzed to identify changes in spent fuel composition and safeguards-relevant signatures. These results are used to evaluate whether LF operation in a SBF PWR SMR leads to measurable deviations from BL operation, and whether these deviations may affect the interpretation of spent fuel NDA measurements.

5.1 Build-Up Analysis of Selected Nuclides During Irradiation

In order to assess the relevance of the selected nuclides for this study, their build-ups during both operating scenarios were evaluated and plotted in Figures 5.1 and 5.2. In the case of seasonal LF operation, the discharge BU reached after 1825 days is 41.37 MWd/kgU. The same BU was reached after 1364.5 days for the case of BL operation. This difference is due to the fact that in LF operation, the reactor does not always operate at 100% nominal power. As a result, fuel burnup accumulates more slowly over time, and a longer operating period is required to reach the same final burnup value.

5.1.1 Build-Up Analysis of FPs During Irradiation

Gamma-emitting FPs are mainly produced when fissile nuclides such as ^{235}U or ^{239}Pu undergo fission splitting into neutron-rich fragments that subsequently decay towards stability, mostly through beta decay. During these decay chains, several radionuclides emit gamma rays as their daughter nuclei de-excite to lower energy states.

The behavior of gamma emitting FP is show in Figure 5.1. Some nuclides, such as ^{141}Ce, ^{95}Nb and ^{95}Zr, are strongly influenced during reactor operation by the power history and follow closely the power evolution, whereas others, such as ^{125}Sb, ^{154}Eu and ^{137}Cs, appear to be less sensitive to changes in the irradiation profile. At the end of irradiation, the two scenarios reach the same BU, but this does not necessarily result in identical FP inventories. The final masses of shorter-lived nuclides such as ^{95}Nb, ^{95}Zr and ^{141}Ce retain different final inventories, as radioactive decay during the longer LF irradiation period and the power level immediately before discharge affect their accumulation. Intermediate cases, such as ^{106}Ru and ^{134}Cs, also show smaller but visible differences. Conversely, the inventories of long-lived nuclides, particularly ^{137}Cs and ^{154}Eu, are very similar in the two scenarios.

In general, isotopes with shorter half-lives (see Table 5.1) are more sensitive to power variations, since their inventories respond more rapidly to changes in the production rate. Longer-lived FPs, on the other hand, are less affected by short-term power variations. As a result, the production of short-lived gamma-emitting FPs is closely related to the fission rate and therefore to the reactor power. However, FPs with half-lives longer than approximately one year reach similar inventories at the same BU, independently of the power history, indicating that their abundance is mostly related to the total neutron fluence, rather than the reactor power.

5.1.2 Build-Up Analysis of TRUs During Irradiation

The evolution for TRUs is shown in Figure 5.2, which shows a less direct influence of the LF scenario on the isotopes buildup compared to the FP (Figure 5.1). Additionally, the final inventory reaches comparable values for the two scenarios in most cases. An explanation for this phenomenon could be the production mechanisms of TRUs, which are mainly formed through successive neutron capture and radioactive decay processes in the actinide chain. As a result, their inventories depend more on the integrated irradiation history, such as total BU and neutron fluence, than on short-term variations in reactor power. Therefore, changes in the power profile tend to have a less direct and less immediate effect on these nuclides compared to short-lived FPs.

A notable example is ^{241}Am, which shows a very similar trend under BL and LF operation, increasing almost monotonically with irradiation time. This behavior may be related to its production pathway, which is mainly formed through the beta decay of ^{241}Pu. Since ^{241}Pu has a relatively long half-life, approximately 14 years, the evolution of ^{241}Am is less directly affected by short-term power variations than shorter-lived FPs.

Conversely, ^{242}Cm shows the clearest sensitivity to the power history among

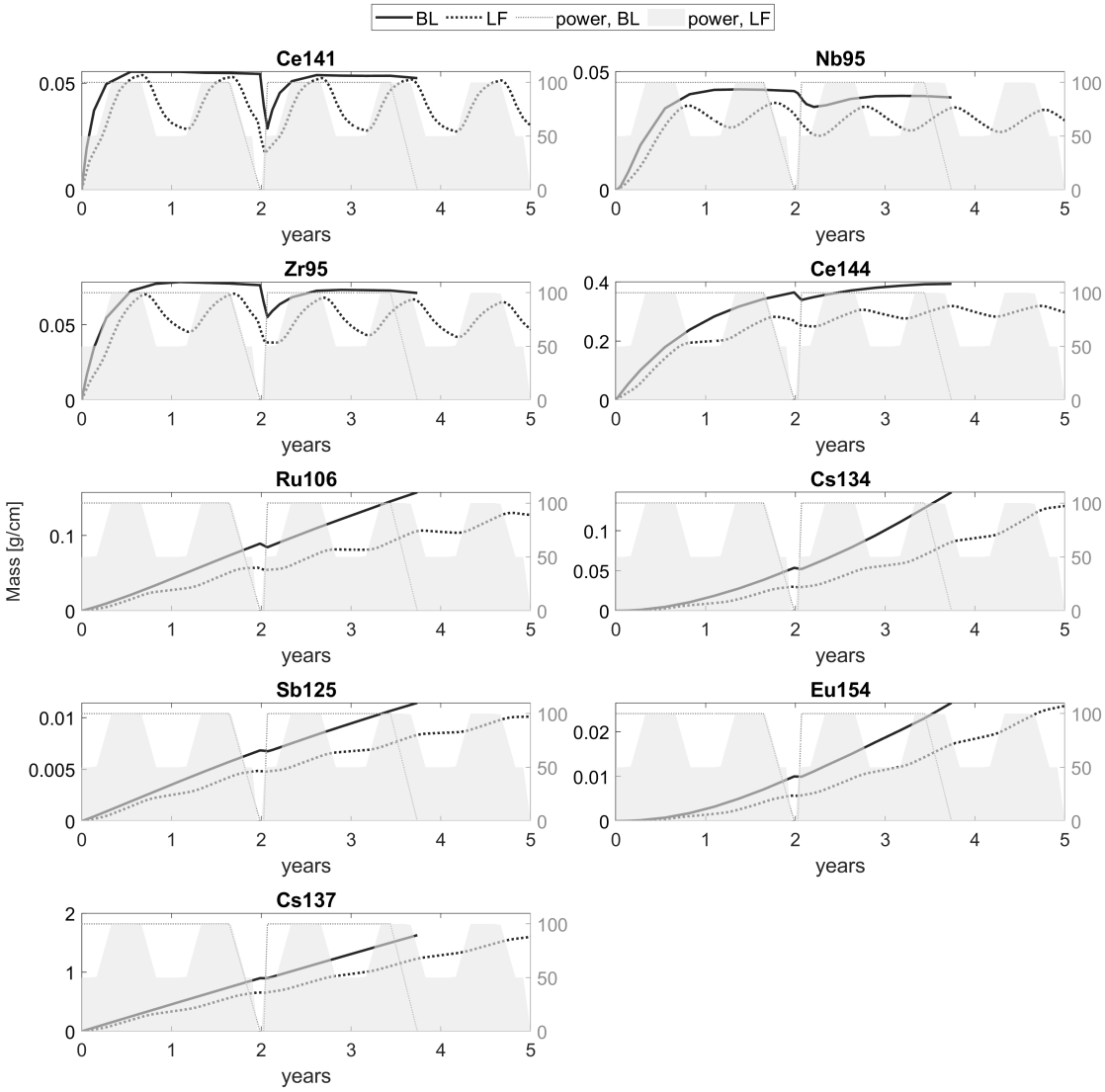


Figure 5.1: Evolution of FP mass per axial length during reactor operation in order of increasing half-life.

the selected TRUs. Its relatively short half-life of approximately 163 days causes its inventory to increase during high-power periods and decrease when production is reduced, allowing it to follow the LF profile more closely than the other TRUs. Despite their different evolutions, the BL and LF cases reach similar ^{242}Cm inventories at the end of irradiation.

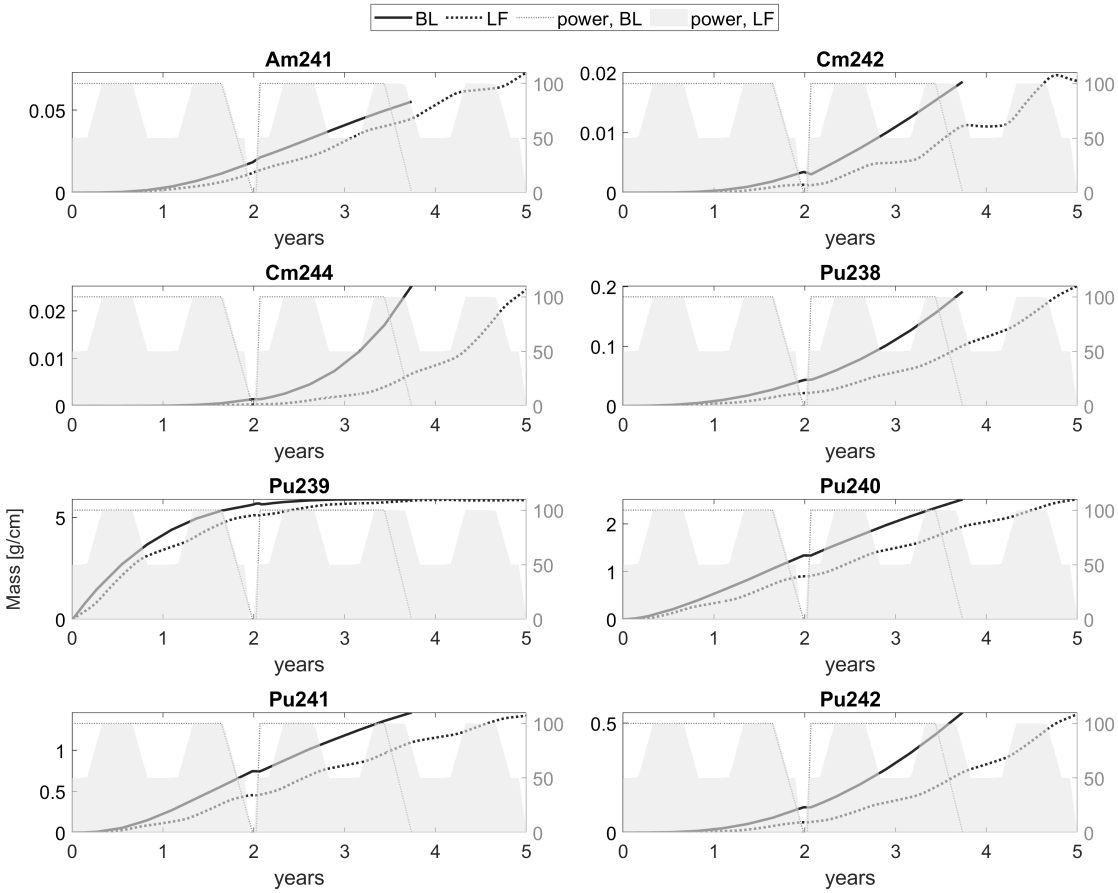


Figure 5.2: Evolution of TRU mass per axial length during reactor operation in order of increasing half-life.

5.2 Changes in Inventories with CT

In this section, the results obtained from the model of the single FA in full power and seasonal LF operation are displayed. In order to visualize the effect of LF operation, the variation of selected nuclides under LF operation relative to the BL case was calculated as described in Equation 5.1.

$$\Delta m_{rel} = \frac{m_{LF} - m_{BL}}{m_{BL}} \times 100 \tag{5.1}$$

m_{LF} and m_{BL} are the corresponding masses per unit axial length under LF and BL operation, respectively.

5.2.1 FP Inventory After Discharge.

The inventory of the selected gamma-emitting FPs at discharge is reported in Table 5.1. The nuclides are ordered by increasing half-life, from the shortest-lived to the longest-lived isotope. The table includes the FP masses per axial length (1 cm) at discharge for both BL and LF operation and the half-life of each nuclide. This provides a direct visualization of the impact of LF operation on the discharge for each nuclide.

Table 5.1: Comparison of FP masses at discharge for BL and LF operation.

Isotope	Half-life	m_{BL} [g/cm]	m_{LF} [g/cm]	$\Delta m_{\mathrm{rel,0y}}$ [%]
^{141}Ce	32.5 d	5.23×10^{-2}	3.00×10^{-2}	-42.62
^{95}Nb	35 d	3.91×10^{-2}	2.94×10^{-2}	-24.84
^{95}Zr	64 d	7.10×10^{-2}	4.68×10^{-2}	-34.08
^{144}Ce	285 d	3.94×10^{-1}	2.97×10^{-1}	-24.42
^{106}Ru	372 d	1.58×10^{-1}	1.27×10^{-1}	-19.18
^{134}Cs	2.06 y	1.48×10^{-1}	1.31×10^{-1}	-11.57
^{125}Sb	2.76 y	1.14×10^{-2}	1.01×10^{-2}	-11.65
^{154}Eu	8.6 y	2.65×10^{-2}	2.58×10^{-2}	-2.60
^{137}Cs	30 y	1.63	1.60	-1.76

The evolution of the selected FPs during CT is shown in Figure 5.3. For each nuclide, the mass is plotted as a function of CT and normalized to the maximum value obtained in the BL case. This representation allows the comparison of the decay behavior of nuclides with different absolute masses and half-lives.

Figure 5.4 shows the variations of mass per nuit axial length relative to the BL mass value at discharge and anfter 1, 5, and 10 years of cooling. After discharge, the inventories of most FPs evolve predominantly through radioactive decay, at a rate governed by their respective half-lives. For this reason, FPs inventories become extremely small after approximately five half-lives, making the relative variation a ratio between two nearly negligible quantities. Therefore, values corresponding to CTs longer than five half-lives are not reported.

For all the selected gamma-emitting FPs, it can be observed that the mass at discharge is lower in the LF case than in the BL case, despite the same BU being reached. However, the magnitude of this difference strongly depends on the half-life of the isotope. Short-lived FPs show the largest relative variations, while long-lived FPs are only weakly affected by the change in operating history. This behavior

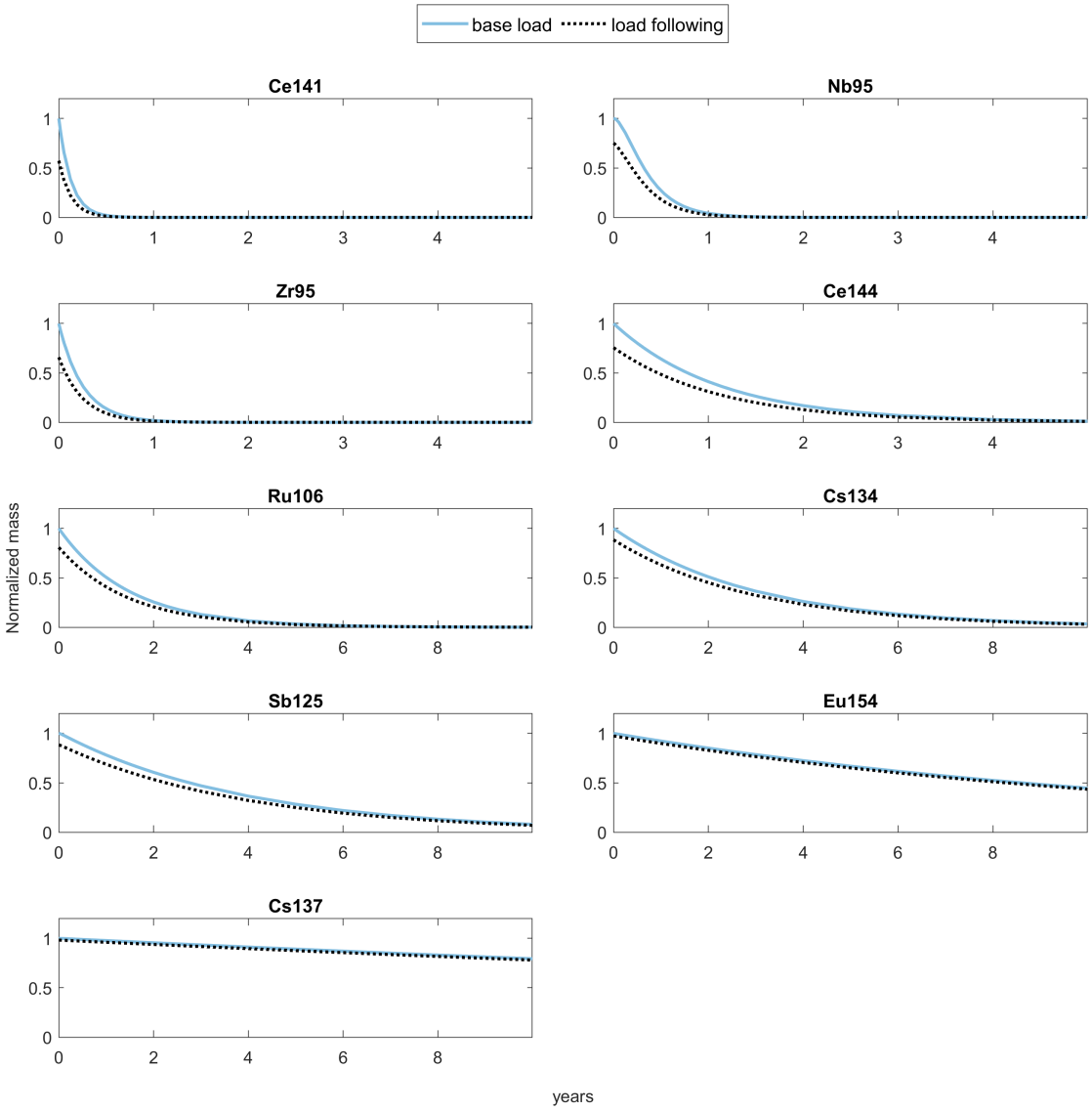


Figure 5.3: Evolution of FP masses per axial length for LF and BL operation relative to BL.

can be explained by the production mechanism of FPs. Gamma-emitting FPs are mainly produced as a direct consequence of fission reactions. Therefore, their production rate is closely related to the total number of fissions and, consequently, to the reactor power; as a result, their inventories respond rapidly to changes in the power level.

This trend is clear for ^{141}Ce, which has a relative variation at discharge of approximately 43%, similar to the order of the power reduction in the LF scenario.

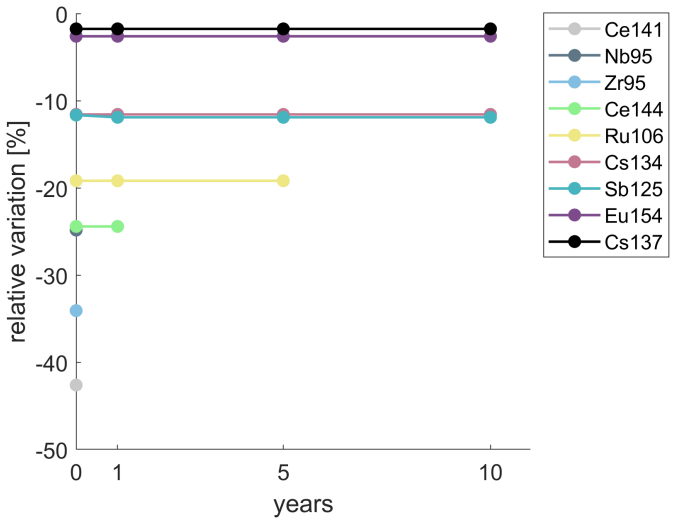


Figure 5.4: Relative mass difference between LF and BL scenarios for selected FPs, for CTs up to 10 years.

This indicates that the inventory of ^{141}Ce is strongly governed by the instantaneous power level close to discharge. A similar, although less pronounced, behavior can be observed for ^{95}Nb, ^{95}Zr, ^{144}Ce, and ^{106}Ru. These isotopes have longer half-lives than ^{141}Ce, but they are still sufficiently short-lived for their inventories to be affected by the power history. Their relative differences between BL and LF operation remain significant at discharge.

On the other hand, FPs with half-lives over 5 years such as ^{154}Eu and ^{137}Cs show relative variations below 5%. In these cases, the half-life is much longer than the timescale of the power changes and of the discharge, and their inventories depend more on the cumulative irradiation history and final BU than on short-term variations in reactor power.

The only isotope that does not have a constant relative variation in CT is ^{95}Nb. Its behavior can be attributed to the fact that after discharge, ^{95}Nb is not governed only by its own radioactive decay, but also by the ingrowth resulting from the decay of its parent nuclide ^{95}Zr.

5.2.2 TRU Inventory After Discharge

The same type of analysis carried out in the section 5.2.1 for the FPs was also performed for the selected TRUs. The inventories at discharge and their half-lives are reported in Table 5.2, while their evolution as a function of CT is shown in Figure 5.5.

The BL and LF mass inventories generally follow similar trends during cooling.

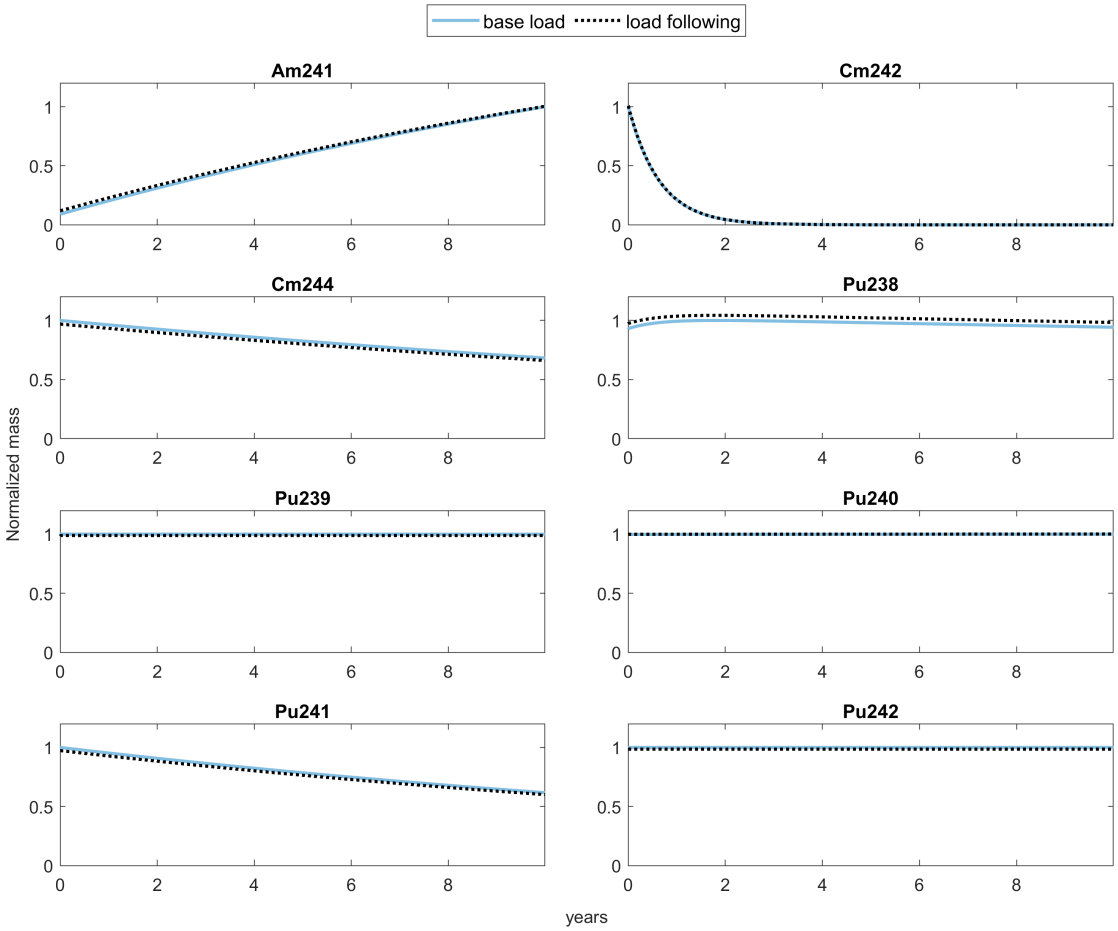


Figure 5.5: Evolution of TRU masses per axial length for LF and BL operation relative to BL.

Most TRUs change slowly because of their long half-lives, whereas ^{242}Cm rapidly decays and contributes to the initial buildup of ^{238}Pu. The increase in ^{241}Am is instead driven by the decay of its parent nuclide, ^{241}Pu.

The isotope masses normalized to the maximum value obtained in the BL case is shown in Figure 5.6, in order to compare nuclides with different absolute inventories.

As expected, the values observed for the relative variation of TRUs are smaller than those obtained for the FPs (Figure 5.4). Some variations are anyhow visible for ^{241}Am, and ^{238}Pu.

In particular, ^{241}Am and ^{238}Pu inventories initially increase during cooling because they continue to be produced through the radioactive decay of their parent nuclides, even though neutron-induced reactions have ceased after discharge. ^{241}Am is produced mainly by the beta decay of ^{241}Pu. Since ^{241}Pu has a half-life of about

Table 5.2: Comparison of TRU masses at discharge for BL and LF operation.

Isotope	Half-life	m_{BL} [g/cm]	m_{LF} [g/cm]	$\Delta m_{\mathrm{rel,0y}}$ [%]
^{242}Cm	163 d	1.85×10^{-2}	1.86×10^{-2}	0.75
^{241}Pu	14.3 y	1.46	1.42	−2.65
^{244}Cm	18 y	2.52×10^{-2}	2.45×10^{-2}	−3.08
^{238}Pu	88 y	1.915×10^{-1}	2.01×10^{-1}	5.00
^{241}Am	433 y	5.51×10^{-2}	7.27×10^{-2}	31.91
^{240}Pu	6561 y	2.51	2.51	0.12
^{239}Pu	24 110 y	5.89	5.85	−0.55
^{242}Pu	357 000 y	5.48×10^{-1}	5.40×10^{-1}	−1.43

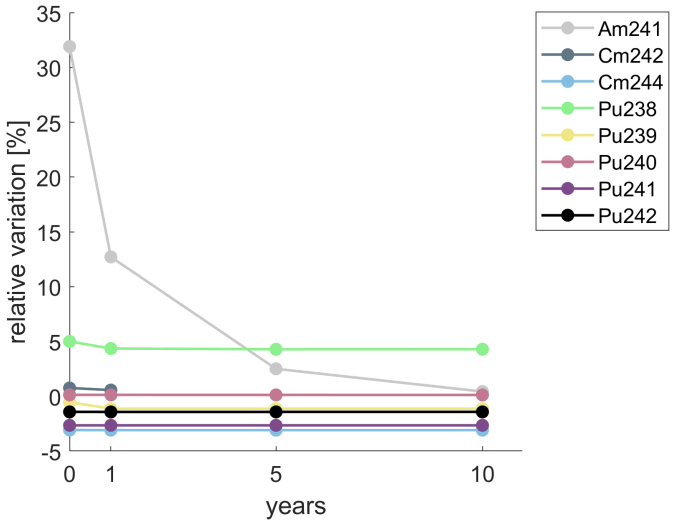


Figure 5.6: Relative mass difference between LF and BL scenarios for selected TRUs, for CTs up to 10 years.

14 years, its decay continues over CT, causing the ^{241}Am inventory to increase while the ^{241}Pu inventory progressively decreases.

A similar behaviour is observed for ^{238}Pu, which is produced by the alpha decay of ^{242}Cm. Because ^{242}Cm has a much shorter half-life of about 163 days, this contribution is particularly important during the first years after discharge. The ^{238}Pu inventory eventually reaches a maximum before gradually declining as its own radioactive decay becomes dominant. As ^{242}Cm is depleted, the production rate of ^{238}Pu decreases. Thus, unlike most of the other TRUs considered, the evolution as

a function of CT of ^{241}Am and ^{238}Pu is governed not only by their own decay but also by continued ingrowth from their respective parent nuclides.

As shown in Figure 5.6, the low-power periods in the LF case lead to an approximately 3% lower ^{241}Pu inventory compared with the BL case. Since ^{241}Pu decays to ^{241}Am, the longer irradiation time associated with LF operation allows a larger fraction of ^{241}Pu to decay during irradiation, resulting in a higher ^{241}Am inventory. This altered actinide evolution also contributes to an approximately 1% increase in ^{242}Cm at discharge. The effect is particularly relevant from a safeguards perspective because ^{242}Cm, together with ^{244}Cm, is an important contributor to the spontaneous-fission neutron source of SNF having a high fission yield (Table 3.3). Consequently, even a relatively small increase in its inventory can temporarily increase the neutron emission rate immediately after discharge and therefore affect neutron-based NDA measurements. However, because ^{242}Cm has a short half-life of about 163 days, its contribution decreases rapidly during cooling, so the associated difference in neutron emission between LF and BL operation is expected to become negligible after a few years. By contrast, ^{244}Cm has a considerably longer half-life of about 18 years and therefore remains an important contributor to the neutron source over longer CTs. In the present results, its mass is approximately 3% lower in the LF case than in the BL case, and this relative difference remains almost constant throughout the considered CT. Consequently, the variation in ^{244}Cm may represent a more persistent difference in the neutron emission signature.

For the remaining TRUs, the relative variations between the two operating scenarios remain small. Their inventories are governed mainly by the balance of production and destruction through successive neutron capture reactions and therefore depend more strongly on the cumulative neutron exposure and final BU than on short-term power variations.

5.2.3 Plutonium Vector

The plutonium vector describes relative fractions of different plutonium isotopes in the material. It changes as the fuel undergoes irradiation and cooling, since it reflects the BU and decay of the various isotopes. This section presents the isotopic composition of plutonium at discharge and after one, five, and ten years of CT.

As shown in Figure 5.7, the LF and BL plutonium vectors are almost indistinguishable at each CT considered. This indicates that different irradiation scenarios do not produce a significant variation in the plutonium composition. At discharge, the plutonium inventory is dominated by ^{239}Pu, followed by ^{240}Pu and ^{241}Pu, while ^{242}Pu and ^{238}Pu represent smaller fractions of the total plutonium mass. The contribution of the remaining plutonium isotopes is negligible.

As discussed in section 5.2.2, the relative mass variations between LF and BL are predominantly negative. In particular, the inventories of ^{239}Pu and ^{241}Pu

are slightly lower in the LF case. Conversely, ^{238}Pu shows a positive relative variation at discharge. An increased fraction of ^{238}Pu is generally favorable from a proliferation-resistance perspective because of its high specific decay heat, which increases the thermal load of the material and complicates its handling, cooling, and potential weaponization [27]. Nevertheless, the magnitude of these variations remains small and does not lead to an appreciable change in the overall plutonium vector.

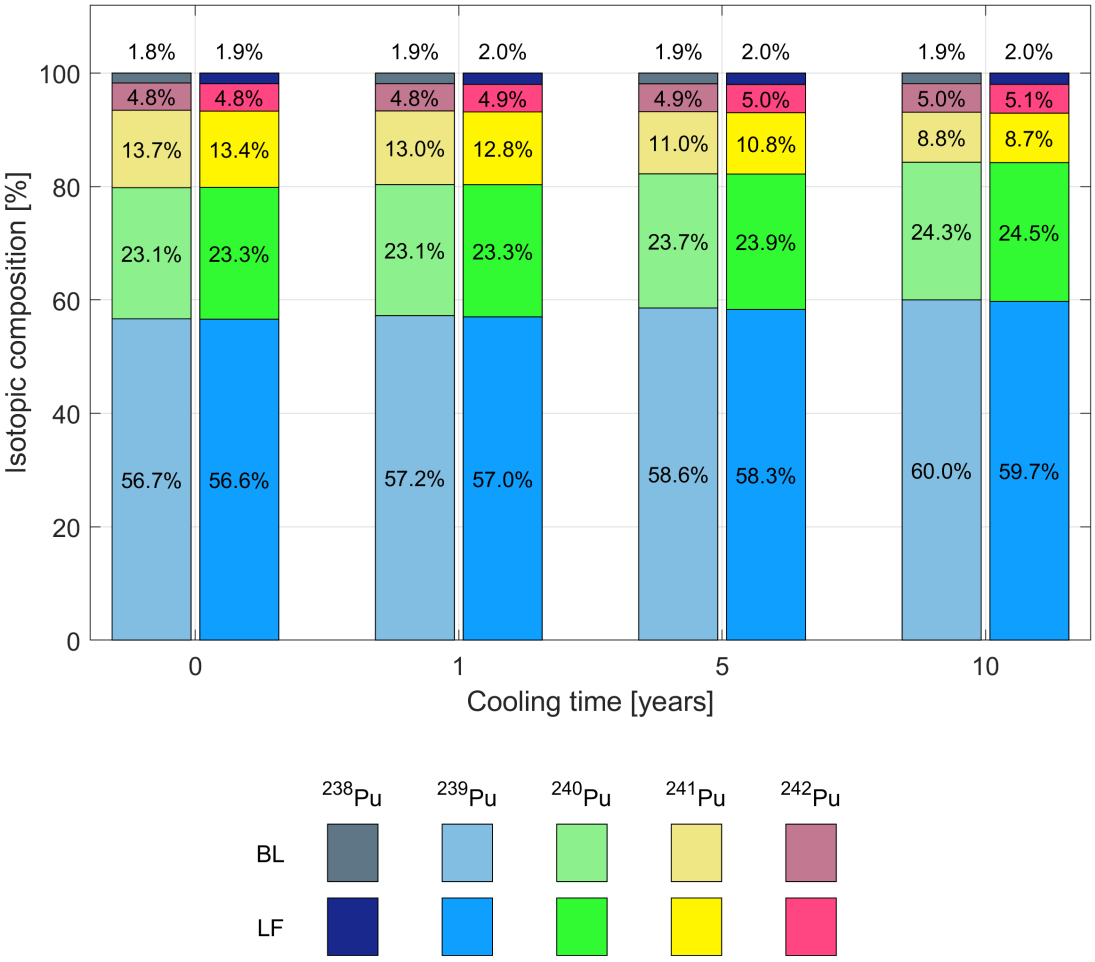


Figure 5.7: Variation in plutonium isotopes as a percentage of the total Pu for BL and LF.

During cooling, the most evident change in the plutonium isotopic composition in Figure 5.7 is the reduction of the ^{241}Pu fraction. This behaviour results from the radioactive decay of ^{241}Pu into ^{241}Am, as discussed in the previous section. As the

^{241}Pu inventory decreases, the relative contribution of the longer-lived plutonium isotopes, particularly ^{239}Pu and ^{240}Pu, increases slightly. After ten years of cooling, the reduction in the ^{241}Pu fraction is clearly visible; however, the difference between the LF and BL cases remains very small. Therefore, the evolution of the plutonium vector is governed primarily by CT rather than by the operating mode.

The almost overlapping LF and BL vectors indicate that LF operation does not significantly modify the plutonium quality in this case. Consequently, although LF operation produces measurable differences in the inventories of individual plutonium isotopes, these variations are not sufficiently large to cause a significant change in the relative plutonium isotopic composition.

In addition to the isotopic vector, Table 5.3 reports the total plutonium mass contained in a single FA for both BL and LF operation and compares it with the IAEA significant quantity (SQ) of 8 kg [28].

Table 5.3: Total plutonium mass per assembly and relative mass to the SQ.

CT	$m_{\mathrm{Pu,BL}}$ [kg]	$m_{\mathrm{Pu,LF}}$ [kg]	$\mathrm{SQ}_{\mathrm{BL}}$	$\mathrm{SQ}_{\mathrm{LF}}$
0 y	2.12	2.11	0.26	0.26
1 y	2.12	2.10	0.26	0.26
5 y	2.07	2.05	0.26	0.26
10 y	2.02	2.01	0.25	0.25

At discharge, the plutonium masses are approximately 2.1 kg for both BL and LF operation, corresponding to approximately 0.26 SQ. The total plutonium inventory progressively decreases during cooling, reaching approximately 2 kg after ten years. The difference between the two operating scenarios therefore remains small throughout the considered CTs.

5.3 Four FAs with CRs

As previously described, a four FA configuration was modelled to investigate how the presence of the CRs impacts the neutron environment and the SNF isotopic inventories.

As in the previous analysis, the main quantity used for the comparison is the radionuclide mass per axial length. However, the values obtained from the single and four FA models cannot be compared directly, since the latter contains four times the number of FAs. To remove this geometrical scaling effect, the total mass per axial length obtained from the four FA model was divided by four. The resulting quantity represents the average mass per axial length per FA and can be compared directly with the corresponding value obtained from the single FA model. Since only one of the four FAs contains the CRs, any local effect associated with their presence is diluted by the contributions from the other three FAs. Consequently, the normalized results describe an average behaviour rather than the specific response of the FA to the CR.

This normalization makes it possible to focus on differences arising from the change in the model configuration.

To compare the two models, the buildup of FPs and TRUs for the BL scenario of the two models is presented, followed by the evolution of isotopes for both BL and LF operation as a function of the same CTs considered in the single FA analysis. Particular attention is given to the selected FPs and TRUs, since changes in their inventories may indicate modifications of the neutron spectrum and reaction rates caused by the presence of CRs.

5.3.1 Comparison of Build-Up Inventories in BL Operation

This section compares the build-up of the selected FPs and TRUs during irradiation under BL operation for the two model configurations. To facilitate the comparison, a scaling factor, f_S, described in Equation 5.2 was introduced to match that of single FA model.

$$f_S = \frac{m}{m_{1FA}} \tag{5.2}$$

In this equation m denotes the mass per unit axial length obtained from the model being evaluated. Deviations of f_S from unity indicate differences between the average inventory per assembly in the four FA configuration and the corresponding inventory in the single FA model. Values of $f_S > 1$ indicate a larger average inventory in the four FA configuration, whereas values of $f_S < 1$ indicate a lower average inventory compared with the single FA model. The evolution of this scaling factor during irradiation provides a direct indication of the differences introduced by the four-assembly configuration and the presence of the CRs.

The results for the selected FPs and TRUs are shown in Figures 5.8 and 5.9, respectively. In general, the effect of CRs is expected to be more evident for the neutron emitting TRUs. This is because CRs absorb thermal neutrons modifying the neutron spectrum inside the fuel. As a consequence, the production and depletion of TRUs can be affected. A smaller effect is also expected in the case of FPs, specifically for those whose evolution depends significantly on neutron capture reactions.

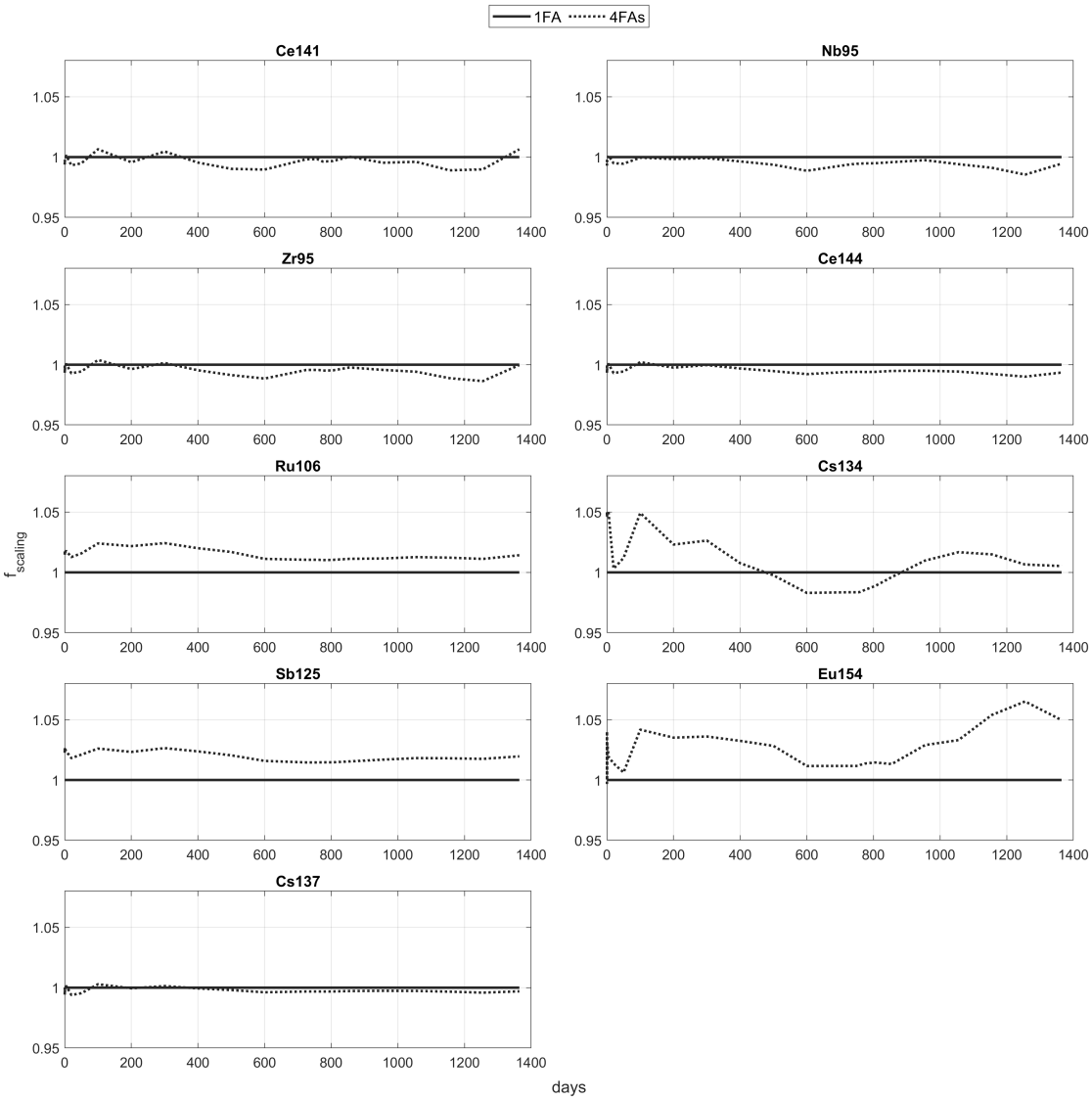


Figure 5.8: FP f_S evolution as a function of days of irradiation in the reactor for a single FA and for an average FA of the four FA model.

Figure 5.8 shows that the variation in f_S for the selected FPs is generally around 1-2% and remains within the range of -1% to 6%. The largest variations are observed for ^{154}Eu and ^{134}Cs. ^{154}Eu is produced mainly through neutron capture on stable ^{153}Eu, and its variation reaches a maximum of approximately 6%. Similarly, ^{134}Cs is produced primarily through neutron capture on stable ^{133}Cs, and its variation ranges from approximately -1% to 5%. All other isotopes are mainly produced by $\beta-$ decay. As expected, the two isotopes produced by neutron capture are the most influenced by the presence of CRs.

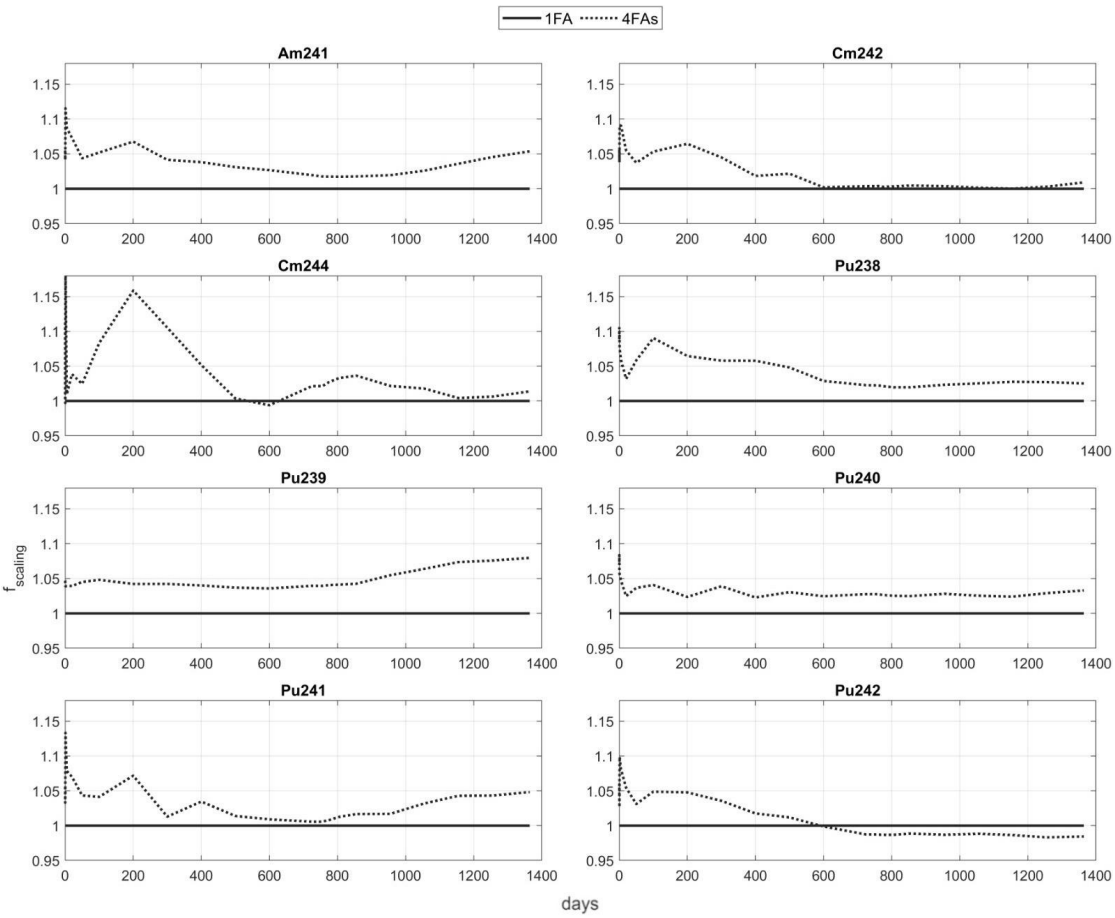


Figure 5.9: TRU f_S evolution as a function of days of irradiation in the reactor for a single FA and for an average FA of the four FA model.

Figure 5.9 shows the evolution of f_S as a function of days of irradiation for the selected TRUs. In general, the deviations from unity remain small, reaching maximum values of approximately 10%. Most of the considered nuclides exhibit an initial increase in f_S, followed by a peak and a subsequent transition towards a more stable, approximately linear trend as irradiation proceeds. This behavior

may be related to the production mechanisms of TRUs, which are mainly governed by successive neutron capture and radioactive decay chains. Their inventories therefore depend on both the neutron spectrum and the availability of their precursor isotopes. The presence of the CRs modifies the neutron environment by preferentially absorbing thermal neutrons, thereby altering the neutron spectrum and the spatial distribution of the flux. Since the simulations are normalized to the same prescribed power, the neutron flux adjusts to compensate for the additional neutron absorption introduced by the CRs and to sustain the required fission rate. These changes may also affect the neutron capture rate and depletion behavior of the CSBA, further modifying the neutron environment during irradiation.

During the initial part of irradiation, the actinide composition evolves relatively rapidly as plutonium and heavier TRUs progressively build up. Differences in neutron capture rates between the two configurations can therefore lead to a more pronounced variation in the inventories of their precursor and daughter nuclides, producing the initial evolution and peak observed in f_S. As irradiation proceeds, the changes in the precursor inventories and in the neutron environment become more gradual, and the relative difference between the two configurations tends to stabilize.

5.3.2 Comparison of the Two Models During Cooling

To compare the impact of LF operation on the two configurations, the relative variation of each selected nuclide was evaluated for the four FA model. This variation was calculated for LF operation relative to the corresponding BL case. The comparison was performed at discharge and after 1, 5, and 10 years of CT, consistently with the analysis previously carried out for the single FA model.

As shown in Figures 5.10 and 5.11, the relative variations obtained for the four FA model were compared directly with those previously calculated for the single FA model at the same CTs. This makes it possible to assess whether the presence of CRs modifies the sensitivity of the selected FP and TRU inventories to the irradiation history.

Figure 5.10 compares the relative mass variation introduced in Section 5.2.1 between LF and BL scenarios at discharge and during CT for the same isotopes. In general, the dependence on irradiation history previously observed for the single FA model is preserved in the four FA configuration. After discharge, the corresponding inventories in the LF and BL cases are generally governed by radioactive decay processes. Consequently, the relative difference between the two inventories remains approximately unchanged. For ^{141}Ce and ^{95}Zr, the reduction is slightly larger in the four FA model, with the relative variation becoming approximately 0.6 and 0.35 percentage points more negative, respectively. A similar, although smaller, effect is observed for ^{106}Ru. Conversely, ^{144}Ce shows a slightly smaller variation in the

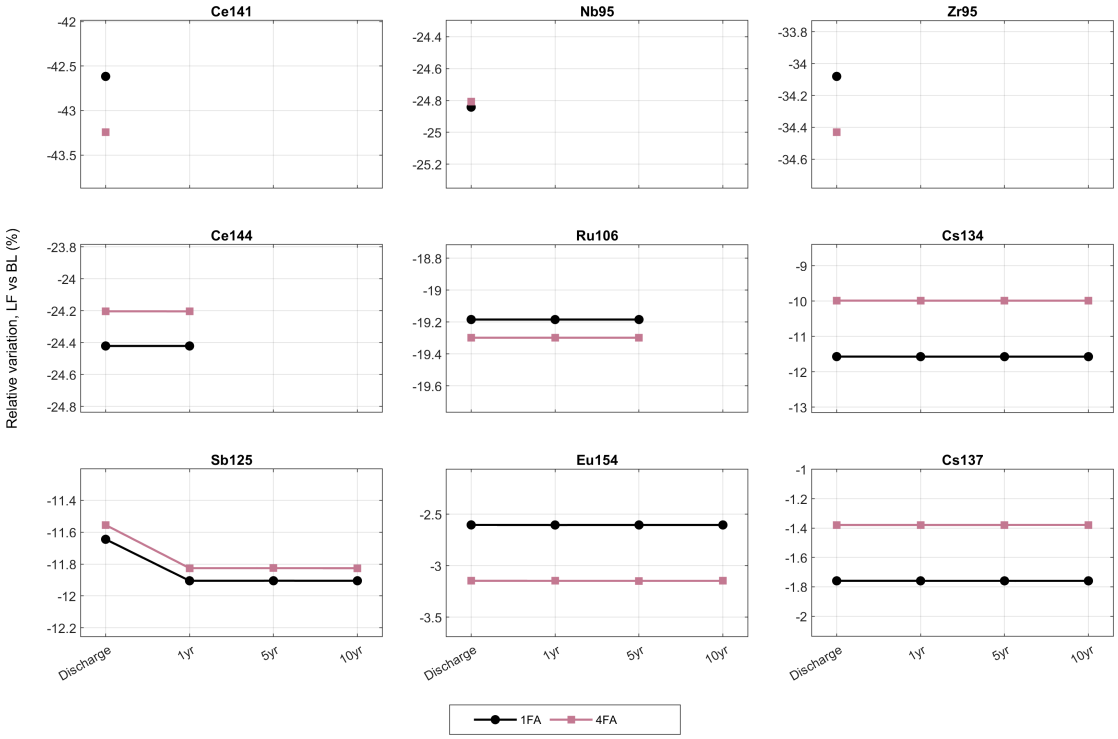


Figure 5.10: Comparison of the mass relative variations of FPs inventories in the single FA model and the four FA model.

four FA model. The largest difference between the two configurations is observed for ^{134}Cs. Its relative variation changes approximately 1.5 percentage points. A noticeable difference is also observed for ^{154}Eu, for which the relative variation changes by 0.7 percentage points. These two isotopes are particularly interesting because, as previously explained, their inventories are significantly influenced by neutron capture reactions. A difference between the two configurations is also observed for ^{137}Cs, although it remains small. For ^{125}Sb, the two configurations remain very similar, with only a small change between discharge and the first year of cooling and the relative mass variation of ^{95}Nb is almost unchanged.

Figure 5.11 shows that the differences between the two configurations in the relative variation for TRUs are generally modest but are more pronounced for some plutonium isotopes. In particular, the four FA model produces limited changes for ^{241}Am, ^{242}Cm, ^{244}Cm and ^{238}Pu, whereas more differences are observed for ^{239}Pu and ^{240}Pu. For ^{241}Am, the four FA model follows the same trend as the single FA model, with a systematically slightly lower relative variation. The difference between the two configurations remains small throughout all CTs. For ^{242}Cm, the relative variation at discharge is below 1% in both models. For ^{244}Cm, the relative

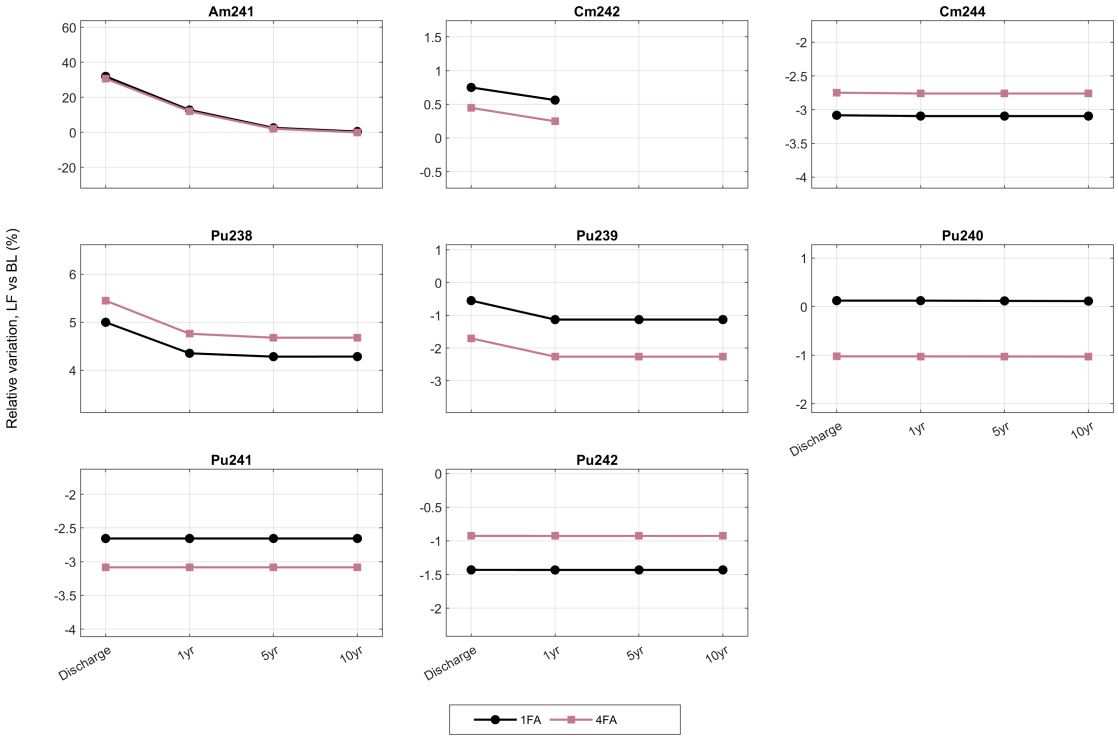


Figure 5.11: Comparison of the relative variations of TRUs inventories in the single FA model and the four FAs model.

variation in the four FA model is approximately 0.3 percentage points lower than in the single FA model. This difference remains essentially constant throughout and represents a small but persistent effect, since ^{244}Cm is an important spontaneous fission neutron source over the considered CT. For ^{238}Pu, the relative variation in the four FA model is approximately 0.4 percentage points higher than in the single FA model and remains visibly different throughout CT. Small differences are also observed for ^{241}Pu and ^{242}Pu. The relative variation for ^{241}Pu is approximately 0.35 percentage points higher in the four FA model, whereas that for ^{242}Pu is approximately 0.5 percentage points lower.

Some small differences are also observed for ^{239}Pu and ^{240}Pu. For ^{239}Pu, the relative variation changes from approximately -0.6% in the single FA model to about -1.7% in the four FA model. After one year of cooling, the relative variation becomes slightly more negative in both models and then remains nearly constant thereafter. For ^{240}Pu, the variation changes from slightly positive in the single FA model, around $+0.1\%$, to approximately -1.0% in the four FA configuration. Therefore, the LF operation produces a slight increase in the ^{240}Pu inventory relative to BL in the single FA model, but a decrease in the four FA model. Taken

together, these differences suggest that the configuration with CRs modifies slightly the progression along the plutonium capture chain.

5.3.3 Comparison of the Plutonium Vectors

The Pu isotopic vector was evaluated for the four FA model at discharge and after 1, 5, and 10 years of CT and compared to the results obtained for the single FA model (Section 5.2.3). For each CT, the results are displayed in the following order: single FA BL, four FA BL, single FA LF, and four FA LF. This arrangement allows both the effect of the CR configuration and the effect of the irradiation history to be assessed directly. In particular, the comparison between the single FA and four FA results under the same operating conditions highlights possible changes in the plutonium vector associated with the presence of CRs. At the same time, the comparison between BL and LF results within the four FA model makes it possible to evaluate whether the presence of the CRs modifies the influence of LF operation on the plutonium isotopic composition. As in the previous analysis, the plutonium vector is expressed as the relative contribution of each plutonium isotope to the total plutonium inventory.

As in the single FA case, the plutonium vector shows only minor differences between BL and LF operation, indicating that the different irradiation histories have a limited effect on the relative plutonium isotopic composition. The evolution during cooling is also consistent with that observed for the single FA model: the fraction of ^{241}Pu progressively decreases due to its radioactive decay into ^{241}Am, while the relative contributions of the longer-lived plutonium isotopes, particularly ^{239}Pu and ^{240}Pu, consequently increase.

A more noticeable, although still relatively small, difference is observed when comparing the single FA and four FA configurations. In the four FA model, the fraction of ^{239}Pu is systematically higher than in the corresponding single FA case. At discharge, for example, the ^{239}Pu fraction increases from approximately 55.5% to 56.7% in BL operation and from 55.6% to 56.6% in LF operation. This difference remains visible throughout cooling and, after 10 years, ^{239}Pu represents approximately 60.0% of the plutonium vector in the four FA BL case compared with 58.9% in the single-FA BL case. A similar difference is observed under LF operation, with values of approximately 59.7% and 58.7%, respectively.

Conversely, the four FA model presents slightly lower fractions of ^{240}Pu and ^{242}Pu. The ^{240}Pu fraction is approximately 1.5% lower in the four FA configuration over the considered CTs, while the ^{242}Pu fraction is generally reduced by approximately 5-7%. The differences in ^{241}Pu and ^{238}Pu are considerably smaller.

The systematic increase in ^{239}Pu accompanied by a reduction in the heavier plutonium isotopes can be related to the different neutron environment introduced by the new configuration with CRs. Since ^{240}Pu is produced primarily through

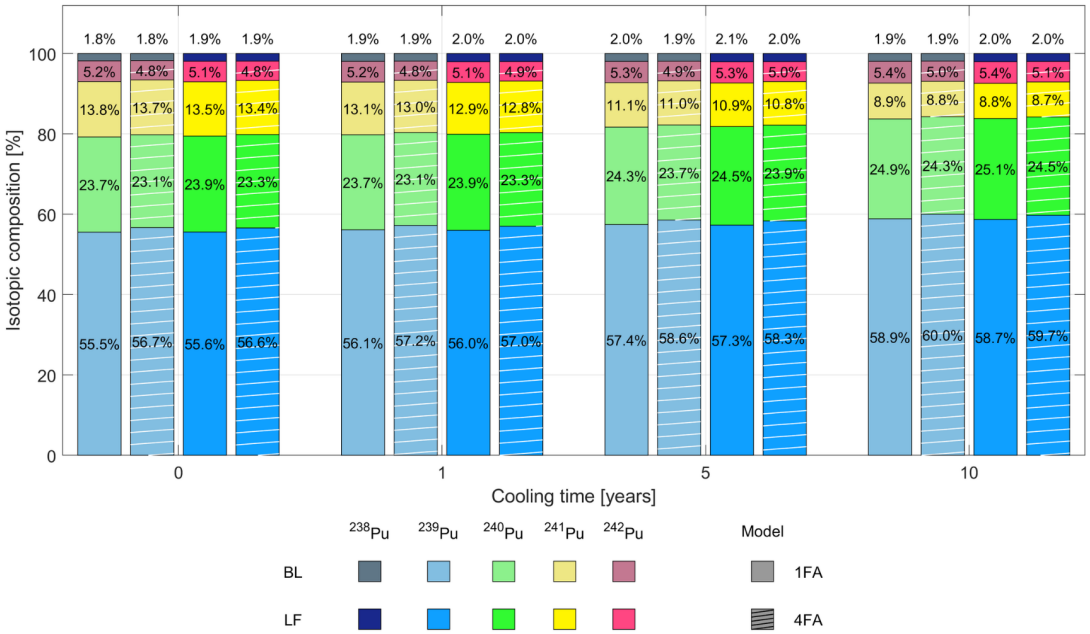


Figure 5.12: Comparison of Pu vectors across different scenarios. For each CT, the results represent values single FA BL, four FA BL, single FA LF, and four FA LF respectively. The different colors represent different isotopes and the striped background is used for the four FAs model.

neutron capture in ^{239}Pu, and the heavier plutonium isotopes are subsequently formed through further neutron capture reactions, a modification of the neutron spectrum and capture rates can alter the progression along the plutonium isotopic chain. The observed abundance of ^{239}Pu and deficiency of ^{240}Pu and ^{242}Pu in the four FA model are consistent with such an effect.

Overall, the differences associated with the model configuration are larger than those produced by the change from BL to LF operation, indicating that the presence of CRs appears to have a greater influence on the relative plutonium isotopic composition than the considered LF irradiation history.

Table 5.4 shows the the model with CRs has a slightly higher total plutonium inventory than the single FA case, but the difference between BL and LF operation remains small at all CTs. In both operating scenarios, the total plutonium mass decreases gradually during cooling, while the corresponding fraction of the IAEA significant quantity remains close to 0.26 - 0.28 per assembly. This indicates that the new configuration has a more visible effect on the absolute plutonium inventory than the LF history itself, whereas LF does not substantially alter the overall plutonium quantity from a safeguards perspective.

Table 5.4: Total plutonium mass per assembly and relative mass to the SQ for the four FA model.

CT	$m_{\mathrm{Pu,BL}}$ [kg]	$m_{\mathrm{Pu,LF}}$ [kg]	$\mathrm{SQ}_{\mathrm{BL}}$	$\mathrm{SQ}_{\mathrm{LF}}$
0 y	2.24	2.21	0.28	0.28
1 y	2.25	2.20	0.28	0.28
5 y	2.19	2.15	0.27	0.27
10 y	2.14	2.10	0.27	0.26

Conclusion

The prospect of increasingly flexible SMR operation, driven by the growing integration of variable energy sources, motivated this work, which aimed at assessing whether non-constant irradiation histories affect the spent fuel signatures used for safeguards verification. In particular, this work investigated the impact of seasonal LF operation on the SNF composition of the SBF ATOM SMR, with particular attention to quantities relevant for safeguards verification. Serpent2 depletion calculations were performed for BL and LF irradiation histories using two different models: a single FA model and a four FA model including CR insertion. The irradiation periods differed between the two scenarios: because the power in the LF case varied between 50% and 100% of the nominal value, an additional year of irradiation was required to reach the same final BU as in the BL case. The evolution of selected FPs, TRUs and the Pu vector was evaluated at discharge and during cooling, allowing the effects of the LF irradiation history to be assessed in both configurations and compared while accounting for the different neutron environments.

The results show that the effect of LF operation is most evident for short lived gamma emitting FPs. At discharge, all selected FPs have lower inventories in the LF case. The magnitude of the variation generally decreases with increasing half-life, indicating that the inventories depend mainly on cumulative irradiation and discharge BU rather than on short term changes in reactor power. Since radionuclide activity is directly related to the number of atoms present, these differences can affect the gamma ray signatures on which many NDA measurements rely. LF operation is therefore expected to have its greatest influence on safeguards measurements at short CTs, while longer-lived fission-product signatures remain comparatively robust.

Smaller differences were generally observed for the TRU inventory. Some isotopes nevertheless show a clear dependence on the longer irradiation time associated with LF operation. In particular, the mass of ^{241}Am is approximately 32% higher at discharge, although this difference decreases strongly during cooling. These variations may influence passive neutron measurements, especially because curium isotopes are important contributors to the neutron source of spent fuel.

LF operation has only a minor effect on the Pu inventory and isotopic composition. At discharge, the total Pu mass per assembly corresponds to approximately 26% of the IAEA significant quantity of 8 kg. The Pu vectors are almost indistinguishable for the two operating scenarios throughout the cooling period and remain reactor-grade. CT produces a more visible modification of the vector. Therefore, under the operating conditions investigated, LF operation does not significantly modify the Pu quality or increase the direct proliferation relevance of the spent fuel.

The analysis including CRs showed that the modified neutron environment introduced by their presence has a limited but measurable effect on the selected radionuclide inventories. During irradiation, the differences between the single FA model and the average FA of the four FA configuration generally remain small. For the selected FPs, the largest deviations are observed for ^{134}Cs and ^{154}Eu, whose production is strongly influenced by neutron capture reactions. For the TRUs, deviations during build-up are generally small, with peaks at the beginning of irradiation around or below 10%, which decrease to approximately 1% later in the irradiation.

When the inventories during CT are compared between the two model configurations, the general trends identified with the single FA model are preserved. However, the presence of CRs modifies the magnitude of the LF induced variation for some nuclides. Among the FPs, the most visible differences are again observed for ^{134}Cs and ^{154}Eu, while among the TRUs the largest changes occur for ^{239}Pu and ^{240}Pu. These results indicate that the modification of the neutron environment introduced by the CRs affects neutron capture reactions and, consequently, the buildup of heavier Pu isotopes. While LF and BL operation produce very similar Pu compositions within each model, some differences can be noted. The four FA model presents a systematically higher fraction of ^{239}Pu and lower fractions of ^{240}Pu and ^{242}Pu. The average Pu inventory is also slightly higher in the configuration including CR, corresponding to 0.28 SQ. Overall, the differences associated with the model configuration are larger than those produced by the change from BL to LF operation for the Pu vector.

Overall, the results of this work show that seasonal LF operation may modify safeguards-relevant SNF inventories even when the same final BU is reached. The magnitude of this effect is strongly nuclide-dependent, with the most pronounced variations generally observed for short-lived gamma-emitting FPs, whereas the influence on most TRUs and on the total Pu inventory is considerably smaller. The Pu isotopic composition is only weakly affected by LF operation and remains reactor-grade throughout the investigated CT. From a safeguards perspective, the conclusions obtained for the single FA model remain generally valid when CRs are included throughout the irradiation of the reactor. The comparison with the four FA model demonstrates that changes in the neutron environment associated

with CRs primarily affect the evolution of TRU inventories and can modify the magnitude of the differences for some nuclides, while the overall trends observed for the single FA model remain unchanged.

The present analysis is based on simplified periodic fuel-assembly models and on radionuclide inventories rather than on the response of a specific safeguards instrument. In addition, the four FA configuration represents an average effect of CR insertion and does not reproduce the spatial and time-dependent behavior of the CRs in the complete ATOM core. Furthermore, because an actual safeguards instrument may not be able to discriminate between the contributions of individual FP and TRU isotopes, differences in calculated inventories do not necessarily translate into distinguishable measured signals. No uncertainty or sensitivity analysis was performed; therefore, it cannot be established whether the small variations observed between the different configurations and irradiation histories represent physically meaningful effects or are comparable to the numerical and modeling uncertainties of the calculations. The results therefore identify which safeguards relevant radionuclides are sensitive to LF operation and to changes in the neutron environment, but they do not directly establish whether these differences would be distinguishable by an actual NDA system. Future work could extend the analysis to more representative core configurations and realistic CR histories, include uncertainty and sensitivity analyses to quantify the significance of the observed variations, and evaluate the corresponding gamma and neutron source terms. The implementation of specific detector response calculations would then make it possible to determine whether the inventory differences identified in this work have a measurable impact on SNF safeguards verification.

Appendix A

Generation IV SMR Technologies

This appendix provides a brief overview of Generation IV reactor technologies selected for SMRs. SMRs can be classified according to their reactor technology and coolant, which determine their basic operating principles and the way heat is removed from the core. While this thesis focuses on LWR technology, several advanced SMR concepts employ gas, liquid metals, or molten salts as coolants. These technologies may enable higher operating temperatures, alternative neutron spectra, and different fuel-cycle strategies. The following sections provide a brief overview of their main characteristics and operating principles.

According reference [2], following LWRs, the second most popular SMR design is the gas-cooled kind, with 32 designs currently explored worldwide. Gas-cooled reactors require really high operating temperature and usually have two types of coolants: carbon dioxide (CO_2) and helium (He). Even though CO_2 has a lower cost, He is the dominant coolant [3]. The main advantages of He-cooled reactors are the chemical inertness of helium and the possibility of high-temperature operation. High temperature gas reactors (HTGR) are a kind of gas-cooled reactors with some unique properties. Firstly they can use gas turbines, making heat exchangers and steam loops unnecessary but turbine control more complex. Secondly, their high outlet temperature can increase the thermal efficiency of energy storage and flexible loads. Finally, due to their harsh operating environment, TRISO (TRi-structural ISOtropic) fuel is being developed in the form of pebbles or in that of prismatic fuel blocks.

The next types of reactor are the one cooled by liquid metals. They can be of different kind, depending on the coolant that they use, which is usually either sodium, lead or lead-bismuth eutectic. Both materials have excellent heat transfer ability, operate at atmospheric pressure, and are weak neutron moderators. If

neutrons are not significantly moderated, the reactor operates with a fast neutron spectrum. This means that the neutrons maintain a high energy, which increases the probability of neutron capture in fertile isotopes such as ^{238}U and supports their conversion into fissile isotopes, such as ^{239}Pu. As a result, liquid-metal-cooled reactors can breed fissile material from fertile isotopes, making them more efficient for fuel recycling. Currently 26 SMR designs are being developed as metal-cooled reactors [2] and some of them are Integral Fast Reactors (IFRs): a type of fast-neutron reactor that features a compact core design and a high fuel burnup rate [29].

Finally, MSRs are presented because they represent a wide family of reactor concepts, with 26 SMR designs currently using MSR technology [2]. In general, MSRs use molten salts as coolant and can allow coolant temperatures as high as 600°C at nearly atmospheric pressure. Just as in HTGRs, the high temperature can benefit thermal energy storage. However, MSRs often have an intermediate loop, which can provide additional thermal inertia while also introducing heat losses. This loop is needed to isolate radioactive tritium, which is produced by neutron interactions with molten salts. MSRs include many different possible configurations: the fuel can be either liquid or solid, and the reactors can operate with either a thermal or fast neutron spectrum. In the case of liquid fuel, with the fissile material dissolves directly in the molten salt, while in solid fuel configurations the salt used only as coolant. One of the main advantages of liquid-fuel MSRs is the possibility of online refueling, without shutting down the reactor. On the other hand, the power control of liquid-fuel reactors is more complex, partly because delayed neutron precursors can move with the circulating fuel. The fast or thermal spectrum depends on the presence or absence of moderating materials, often graphite.

Some Generation IV SMR technologies have already progressed beyond the conceptual design stage and reached different phases of deployment. In particular, Russia's BREST-OD-300 is under construction at Seversk. It is a 300 MWe lead-cooled fast-spectrum reactor and represents a different technological pathway from conventional water-cooled SMRs. The project is intended as a pilot demonstration unit and is linked to Russia's broader strategy for advanced reactors and closed fuel-cycle technologies. As of the mid-2020s, construction and assembly work have continued, with physical launch targeted for 2026 and grid connection expected afterwards. Russia is also developing the RITM series of water-cooled thermal SMRs. The RITM-200 design, originally developed for nuclear icebreakers, is being adapted for floating NPPs and land-based electricity production. In the United States, the Hermes demonstration reactor by Kairos Power represents the first advanced non-water-cooled reactor to receive a construction permit from the U.S. NRC in more than fifty years. Hermes is a 35 MWt reduced-scale fluoride-salt-cooled high-temperature reactor prototype. The project is now expected to support future deployment of more advanced commercial reactor systems. [2]

Although most Generation IV SMR concepts remain under design, some are already being developed and a few are operational. In China, the HTR-PM at Shidao Bay in Shandong Province is the world's first modular high-temperature gas-cooled pebble-bed reactor power plant to enter commercial operation. It consists of two reactor modules connected to a single steam turbine and demonstrates the use of helium coolant, graphite moderation, and spherical fuel elements [2]. In Russia, the Akademik Lomonosov is the world's first floating NPP. Powered by two KLT-40S reactors, it was connected to the grid in Pevek in 2019 and entered commercial operation in 2020.

Appendix B

The IAEA

The IAEA is an independent intergovernmental organization within the United Nations system and the principal international forum for cooperation in the nuclear field. It is headed by a Director General, who oversees its activities and represents the organization internationally. The IAEA's central objective is to promote the safe, secure and peaceful use of nuclear science and technology while contributing to international efforts to prevent the proliferation of nuclear weapons. In this role, the Agency supports its Member States in applying nuclear technologies to meet social, economic and development needs, including electricity generation, medicine, agriculture, environmental protection, water-resource management and industrial production.

The work of the IAEA is based on several closely connected areas. First, it facilitates the transfer of nuclear knowledge and technology, particularly to developing countries, through technical cooperation, training and capacity-building programmes. Second, it develops internationally recognized safety standards and assists countries in maintaining high levels of nuclear and radiation safety. These activities are intended to protect people and the environment from the harmful effects of ionizing radiation. Additionally, the Agency also contributes to nuclear security by helping States prevent the theft, misuse or unauthorized access to nuclear and other radioactive materials.

A further fundamental responsibility of the IAEA is nuclear verification. Through its safeguards system, the Agency conducts inspections and applies technical measures to verify that States are complying with their international obligations and that nuclear materials and facilities are used exclusively for peaceful purposes. This verification function, carried out under the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) and other international agreements, gives the IAEA a central role in the global nuclear non-proliferation regime.

History of the IAEA

The establishment of the IAEA originated in United States President Dwight D. Eisenhower's "Atoms for Peace" speech, delivered to the United Nations General Assembly on 8 December 1953. Eisenhower proposed the creation of an international organization that would encourage the peaceful use of nuclear energy while reducing the risks associated with the spread of nuclear weapons. Following international negotiations, the IAEA Statute was unanimously approved by 81 States on 23 October 1956 and entered into force on 29 July 1957, formally establishing the Agency as an autonomous organization within the United Nations system. The first IAEA General Conference was held in Vienna in October 1957 and brought together diplomats and scientists from 57 countries. A relationship agreement concluded with the United Nations in 1959 allowed it to benefit from its association with the UN system and requires it to follow certain UN policies. Under Articles III.B.1 and III.B.4 of its Statute, the Agency also submits an annual report to the United Nations and reports matters that may fall under the responsibility of the Security Council for international peace and security.

Over time, the IAEA developed into a major international institution linking nuclear development, safety, non-proliferation and safeguards. Recognition of its contribution to international peace culminated in the award of the 2005 Nobel Peace Prize jointly to the IAEA and its Director General Mohamed ElBaradei. The award reflected the Agency's continuing efforts to ensure that nuclear technology is used safely and peacefully and does not contribute to the development of nuclear weapons.

Building on the work carried out during its first six decades, the IAEA will continue its activities in nuclear energy, nuclear safety and security, nuclear applications, technical cooperation and non-proliferation. The Agency will maintain a strong focus on making nuclear science and technology available to its Member States, particularly developing countries, in a safe, secure and peaceful manner, while supporting the achievement of the United Nations Sustainable Development Goals.

The IAEA Statute

The "Atoms for Peace" proposal led to the drafting of the Statute of the IAEA, which entered into force in 1957 and formally established the IAEA. The Statute is an international treaty, and a State becomes a Member State of the Agency by becoming a party to it. As an autonomous international organization, the IAEA has its own institutional structure, policymaking organs and legal authority.[14]

The principal policymaking organs of the Agency are the General Conference and

the Board of Governors. The General Conference is composed of all Member States, and its powers are defined in Article V of the Statute. Among its responsibilities, it approves the Agency's budget and annual report and may propose matters for consideration by the Board of Governors. The Board consists of 35 Member States, and its composition and powers are established in Article VI. In particular, Article VI.F provides that the Board has the executive authority of the Agency and is responsible for carrying out its functions in accordance with the Statute, subject to its responsibilities to the General Conference.

The administrative structure of the Agency is addressed in Article VII, which provides that the staff is headed by a Director General. The Director General serves under the authority of the Board of Governors and is subject to its control. This institutional relationship is particularly relevant to safeguards. When the IAEA concludes a safeguards agreement with a State, the Director General requests authorization from the Board of Governors to conclude and implement the agreement. This authorization provides the legal basis for the Agency's implementation of safeguards under that agreement.

Safeguards constitute one of the core functions assigned to the IAEA by its Statute. The Statute authorizes the Agency to establish and administer safeguards under specified circumstances, while Article XII contains the basic provisions governing their implementation. These provisions include the possibility of reporting non-compliance with safeguards obligations to the United Nations Security Council and the General Assembly. However, the safeguards provisions contained in the Statute not self-executing and must be accepted by the State, normally through their incorporation into a safeguards agreement. At the same time, the IAEA may also conclude safeguards agreements with States that are not members of the Agency. Consequently, membership in the IAEA and participation in its safeguards system are legally distinct matters. In both cases, the essential legal instrument is the safeguards agreement: where such an agreement is in force, the Agency is authorized to implement safeguards in accordance with its provisions.[14]

Appendix C

Nuclear Weapon Free Zones and the Non Proliferation Treaty

As discussed in section 3.1.1, the NPT distinguishes between NWSs and NNWSs, establishing different obligations for the two groups within the international non-proliferation framework. NWFZs complement this global framework through regional treaties by which States undertake to exclude nuclear weapons from defined territories and establish corresponding verification and control mechanisms. This appendix first describes the development and main features of these zones, then it reproduces the text of articles III and IV of the NPT.

Nuclear-Weapon-Free Zones

Nuclear weapon free zones (NWFZs) are an important and concrete contribution to the nuclear disarmament regime; their establishment represents a regional approach to delegitimize nuclear weapons, strengthen global nuclear non-proliferation and consolidate international efforts towards peace and security. [30]

Article VII of the NPT states that "Nothing in this Treaty affects the right of any group of States to conclude regional treaties in order to assure the total absence of nuclear weapons in their respective territories.". In fact, NWFZs contribute to the preparation of a world free of nuclear weapons by establishing regional commitments concerning the total absence of nuclear weapons and by creating systems of verification and control. Efforts to strengthen these zones include promoting cooperation and consultation among existing zones, encouraging NWS to adhere to the relevant protocols, and supporting the establishment of additional

zones. [30]

General Assembly resolution 3472 (XXX) B of 1975 defines a NWFZ as:

> "... any zone recognized as such by the General Assembly of the United Nations, which any group of States, in the free exercises of their sovereignty, has established by virtue of a treaty or convention whereby: (a) The statute of total absence of nuclear weapons to which the zone shall be subject, including the procedure for the delimitation of the zone, is defined;
> (b) An international system of verification and control is established to guarantee compliance with the obligations deriving from that statute".

The development of Nuclear-Weapon-Free Zones has taken place gradually across different regions. The first such zone was established by the 1959 Antarctic Treaty, which prohibited military activities, nuclear explosions and the disposal of radioactive waste in Antarctica. The concept was subsequently extended to populated regions, beginning with Latin America and the Caribbean. [31]

The 1967 Treaty of Tlatelolco established a Nuclear-Weapon-Free Zone in Latin America and the Caribbean. Its negotiation followed increasing regional concern about nuclear weapons, particularly after the Cuban Missile Crisis. The Treaty prohibited the development, acquisition and stationing of nuclear weapons within the zone and established a system of inspection and verification.[31]

In the South Pacific, concerns related to nuclear testing, radioactive-waste disposal and visits by nuclear-armed ships contributed to the adoption of the 1985 Treaty of Rarotonga. The Treaty prohibited nuclear explosive devices, nuclear testing within the zone and the dumping of nuclear waste at sea.[31]

The 1995 Treaty of Bangkok established a NWFZ in Southeast Asia. It prohibited nuclear weapons in the region and extended its provisions to the exclusive economic zones of its States parties. The Treaty also sought to reinforce the non-proliferation commitments already undertaken by the members of the Association of Southeast Asian Nations.[31]

In Africa, the Treaty of Pelindaba was signed in 1996 following several decades of regional initiatives. Its adoption became possible after major political changes in the region, including the end of apartheid in South Africa and the dismantling of the country's nuclear weapons programme. The Treaty prohibits nuclear weapons and includes provisions concerning the dismantling of existing nuclear-weapon-related facilities.[31]

The 2006 Treaty of Semipalatinsk established a NWFZ in Central Asia after nine years of negotiations. The Treaty was strongly influenced by the region's history of nuclear testing, uranium mining and other nuclear weapon related activities during the Soviet period. In addition to prohibiting nuclear weapons, it prohibits research connected with their development.[31]

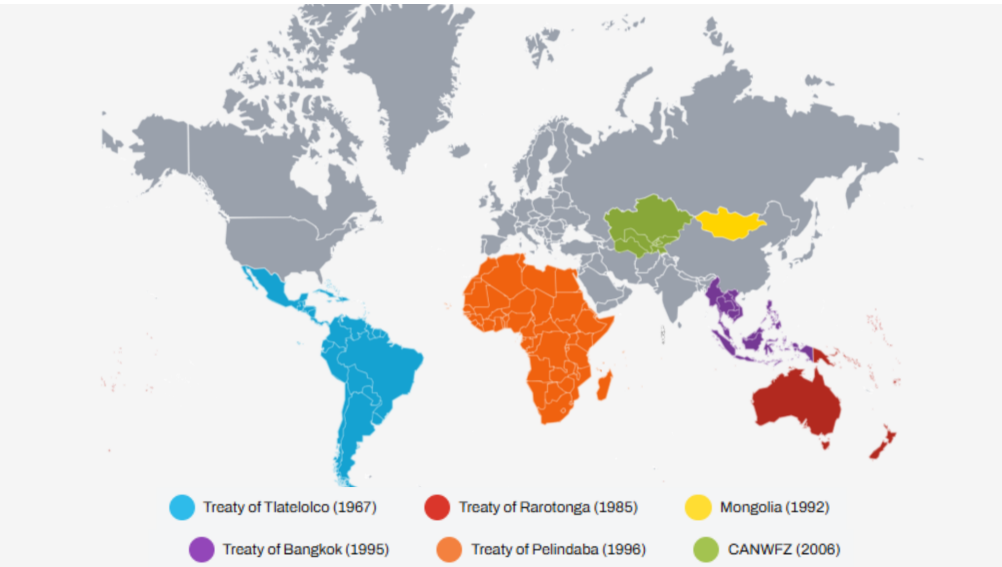


Figure C.1: World map of NWFZs [32].

Figure C.1 shows the geographical extent of the established NWFZ in Latin America and the Caribbean, the South Pacific, Southeast Asia, Africa and Central Asia. Together, these regional treaties have contributed to reinforcing regional non-proliferation commitments and reducing the possibility of nuclear weapons being developed, stationed or tested in these areas. The five regional treaties also require their parties to conclude CSA with the IAEA, linking the regional prohibitions to international verification [30].

NPT

The full text of the NPT is reproduced below [33].

TREATY ON THE NON-PROLIFERATION OF NUCLEAR WEAPONS

The States concluding this Treaty, hereinafter referred to as the "Parties to the Treaty",

Considering the devastation that would be visited upon all mankind by a nuclear war and the consequent need to make every effort to avert the danger of such a war and to take measures to safeguard the security of peoples,

Believing that the proliferation of nuclear weapons would seriously enhance the danger of nuclear war,

In conformity with resolutions of the United Nations General Assembly calling for the conclusion of an agreement on the prevention of wider dissemination of nuclear weapons,

Undertaking to co-operate in facilitating the application of IAEA safeguards on peaceful nuclear activities,

Expressing their support for research, development and other efforts to further the application, within the framework of the IAEA safeguards system, of the principle of safeguarding effectively the flow of source and special fissionable materials by use of instruments and other techniques at certain strategic points,

Affirming the principle that the benefits of peaceful applications of nuclear technology, including any technological by-products which may be derived by nuclear-weapon States from the development of nuclear explosive devices, should be available for peaceful purposes to all Parties to the Treaty, whether nuclear-weapon or non-nuclear-weapon States,

Convinced that, in furtherance of this principle, all Parties to the Treaty are entitled to participate in the fullest possible exchange of scientific information for, and to contribute alone or in co-operation with other States to, the further development of the applications of atomic energy for peaceful purposes,

Declaring their intention to achieve at the earliest possible date the cessation of the nuclear arms race and to undertake effective measures in the direction of nuclear disarmament,

Urging the co-operation of all States in the attainment of this objective,

Recalling the determination expressed by the Parties to the 1963 Treaty banning nuclear weapon tests in the atmosphere, in outer space and under water in its Preamble to seek to achieve the discontinuance of all test explosions of nuclear weapons for all time and to continue negotiations to this end,

Desiring to further the easing of international tension and the strengthening of trust between States in order to facilitate the cessation of the manufacture of nuclear weapons, the liquidation of all their existing stockpiles, and the elimination from national arsenals of nuclear weapons and the means of their delivery pursuant to a Treaty on general and complete disarmament under strict and effective international control,

Recalling that, in accordance with the Charter of the United Nations, States must refrain in their international relations from the threat or use of force against the territorial integrity or political independence of any State, or in any other manner inconsistent with the Purposes of the United Nations, and that the establishment and maintenance of international peace and security are to be promoted with the least diversion for armaments of the world's human and economic resources,

Have agreed as follows:

Article III

1. Each NNWS Party to the Treaty undertakes to accept safeguards, as set forth in an agreement to be negotiated and concluded with the IAEA in accordance with the Statute of the IAEA and the Agency's safeguards system, for the exclusive purpose of verification of the fulfilment of its obligations assumed under this Treaty with a view to preventing diversion of nuclear energy from peaceful uses to nuclear weapons or other nuclear explosive devices. Procedures for the safeguards required by this Article shall be followed with respect to source or special fissionable material whether it is being produced, processed or used in any principal nuclear facility or is outside any such facility. The safeguards required by this Article shall be applied on all source or special fissionable material in all peaceful nuclear activities within the territory of such State, under its jurisdiction, or carried out under its control anywhere.

2. Each State Party to the Treaty undertakes not to provide: (a) source or special fissionable material, or (b) equipment or material especially designed or prepared for the processing, use or production of special fissionable material, to any NNWS for peaceful purposes, unless the source or special fissionable material shall be subject to the safeguards required by this Article.

3. The safeguards required by this Article shall be implemented in a manner designed to comply with Article IV of this Treaty, and to avoid hampering the economic or technological development of the Parties or international co-operation in the field of peaceful nuclear activities, including the international exchange of nuclear material and equipment for the processing, use or production of nuclear material for peaceful purposes in accordance with the provisions of this Article and the principle of safeguarding set forth in the Preamble of the Treaty.

4. Non-nuclear-weapon States Party to the Treaty shall conclude agreements with the IAEA to meet the requirements of this Article either individually or together with other States in accordance with the Statute of the IAEA. Negotiation of such agreements shall commence within 180 days from the original entry into force of this Treaty. For States depositing their instruments of ratification or accession after the 180-day period, negotiation of such agreements shall commence not later than the date of such deposit. Such agreements shall enter into force not later than eighteen months after the date of initiation of negotiations.

Article IV

1. Nothing in this Treaty shall be interpreted as affecting the inalienable right of all the Parties to the Treaty to develop research, production and use of nuclear energy for peaceful purposes without discrimination and in conformity with Articles I and II of this Treaty.

2. All the Parties to the Treaty undertake to facilitate, and have the right to participate in, the fullest possible exchange of equipment, materials and scientific and technological information for the peaceful uses of nuclear energy. Parties to the Treaty in a position to do so shall also co-operate in contributing alone or together with other States or international organizations to the further development of the applications of nuclear energy for peaceful purposes, especially in the territories of non-nuclear-weapon States Party to the Treaty, with due consideration for the needs of the developing areas of the world.

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