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System Analysis Approaches in the Nuclear Safeguards of Liquid-Fuelled Molten Salt Reactors

A Safeguardability Assessment Tool and NDA Applicability Study

TESI DI LAUREA MAGISTRALE IN
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Abstract

To address the growing energy demands and combat climate change, alternative technologies are being explored globally. Liquid-fuelled molten salt reactors (MSRs) are promising from a safety and performance aspect, given their intrinsic technical characteristics. However, given the inevitable increase in the number of reactors, the question of how to effectively and efficiently put them under International Safeguards is of the utmost importance. These reactors represent a paradigm shift from traditional fuel item-based systems, and the current safeguards approaches prove inadequate for their bulk-fuel operational modalities. The evaluation of the new Safeguards approaches and their suitability to reactor designs is needed to direct research efforts in bridging the existing gap. The first outcome is a state-of-the-art of Nuclear Safeguards, and the relative challenges posed by MSRs. This work develops a comprehensive Safeguardability Assessment Tool (SAT) by synthesising existing safeguardability evaluation attributes from the literature and tailoring them to the specific characteristics of liquid-fuelled MSRs. The research employs two methodologies adapted from the System Analysis framework to quantify inherently qualitative safeguardability assessments: Multi-Attribute Utility Theory (MAUT) to capture expert preferences through the SAT's attribute weighting, and Dempster-Shafer Theory (DST) through the Italian Flag methodology to manage uncertainty and incompleteness in the pre-conceptual design phase. The tool's layered architecture provides flexible assessment at multiple levels of detail, enabling both policy-level evaluations and granular inspector-focused assessments, thereby serving the diverse stakeholder needs within the safeguards ecosystem. The study of the applicability of Non-Destructive Assay to the MSFR through the Italian Flag methodology provides actionable insights for identifying knowledge gaps, successfully propagating expert judgment through hierarchical dependencies, and thus proving the adequacy of the SAT for the quantitative estimation of the design's safeguardability. This framework bridges designers and inspectors by establishing an early-stage safeguardability assessment, ensuring safeguards integration at minimal retrofit costs and informing the inspectorate's long-term instrumentation development priorities.

Keywords: Nuclear Safeguards, molten salt reactors, safeguardability assessment, safeguards-by-design, system analysis in safeguards, uncertainty propagation, non-destructive assay

Abstract in italiano

Per far fronte all'aumento del fabbisogno energetico e combattere i cambiamenti climatici, la comunità scientifica sta studiando tecnologie alternative. I reattori a sali fusi (MSR) alimentati a combustibile liquido sono promettenti dal punto di vista della sicurezza e delle prestazioni, date le loro caratteristiche tecniche intrinseche. Tuttavia, dato l'inevitabile aumento del numero di reattori, la questione di come sottoporli in modo efficace ed efficiente alle salvaguardie internazionali è della massima importanza. Questi reattori rappresentano un cambiamento di paradigma rispetto ai tradizionali sistemi, e gli attuali approcci di salvaguardia basati sugli elementi di combustibile si dimostrano inadeguati per le loro modalità operative con combustibile sfuso. È necessario valutare i nuovi approcci di salvaguardia e la loro applicabilità ai design dei reattori, per orientare gli sforzi di ricerca nel colmare il divario esistente. Il primo risultato è uno stato dell'arte delle salvaguardie nucleari e delle relative sfide poste dagli MSR. Questa tesi sviluppa uno strumento completo di valutazione della salvaguardabilità (SAT) sintetizzando gli attributi di valutazione della salvaguardabilità esistenti nella letteratura e adattandoli alle sfide specifiche poste dagli MSR alimentati a combustibile liquido. Questo studio utilizza due metodologie adattate dal campo di Analisi dei Sistemi per quantificare le valutazioni di salvaguardabilità intrinsecamente qualitative: la Multi-Attribute Utility Theory (MAUT) per cogliere le preferenze degli esperti tramite la ponderazione degli attribute del SAT, e la teoria di Dempster-Shafer (DST) attraverso la metodologia Italian Flag per gestire l'incertezza e l'incompletezza nella fase iniziale della progettazione concettuale. La struttura a livelli della SAT offre flessibilità rispetto alle esigenze delle parti interessate, dalle necessità delle valutazioni politiche ai bisogni di dettaglio nelle valutazioni degli ispettori. L'applicazione della metodologia Italian Flag all'applicabilità dell'NDA per il MSFR fornisce informazioni utili per identificare le lacune nella strumentazione, propagando con successo il giudizio degli esperti attraverso dipendenze gerarchiche. Il framework sviluppato collega progettisti e ispettori stabilendo una valutazione della salvaguardabilità nella fase iniziale, aiutando nell'integrazione delle salvaguardie a costi di adeguamento minimi e informando le priorità di sviluppo della strumentazione a lungo termine dell'ispettorato.

Parole chiave: Salvaguardie nucleari, reattori a sali fusi, valutazione della salvaguardabilità, salvaguardie progettuali, analisi di sistema nelle salvaguardie, propagazione dell'incertezza, analisi non distruttiva

The existence of nuclear weapons presents a clear and present danger to life on Earth. Nuclear arms cannot bolster the security of any nation because they represent a threat to the security of the Human race. These incredibly destructive weapons are an affront to our common humanity, and the tens of billions of dollars that are dedicated to their development and maintenance should be used instead to alleviate Human need and suffering - Óscar Arias Sánchez, Nobel Peace Prize Laureate

Oggi, in uno scenario segnato da guerre, crescenti tensioni e contrapposizioni, occorre ribadire con forza che l'uso o anche la sola concreta minaccia di introdurre nei conflitti armamenti nucleari appare crimine contro l'umanità. L'architettura globale del disarmo e della non proliferazione delle armi nucleari, tra i cardini del sistema multilaterale faticosamente costruito nel secondo dopoguerra, non può essere abbandonata, a rischio di accelerare un clima di scontro -
Sergio Mattarella, Presidente della Repubblica Italiana

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List of Acronyms of Interest

AHP: Analytic Hierarchy Process

Am: Americium

AP: Additional Protocol

ARE: Aircraft Reactor Experiment

BTC: Basic Technical Characteristics (document equivalent to the DIQ)

Cf: Californium

Cm: Curium

Cl: Chlorine, but could refer to Chloride, which is the negative ion Cl^-

CoK: Continuity of Knowledge

CT: Cooling Time

C/S: Containement & Surveillance

CSA: Comprehensive Safeguards Agreement

DA: Destructive Analysis

DIQ: Design Information Questionnaire

DIV: Design Information Verification

DP: Detection Probability

DPA: Diversion Path Analysis

DST: Demster-Schafer Theory

EC: European Commission

F: Fluorine

FP: Fission Product

FSA: Facility Safeguardability Analysis

GIF: Generation IV International Forum

HALEU: High-Assay Low Enriched Uranium (close to but lower than 20% ^{235}U)

HEU: Highly Enriched Uranium (more than 20% ^{235}U, generally WG)

HKED: Hybrid K-edge Densitometry

HPGe: High-Purity Germanium Detector

IAEA: International Atomic Energy Agency

ICBM: Intercontinental Ballistic Missile

ID: Inventory Difference, Material Unaccounted For

IMCI: Inventory-Monitor Containment-Inventory safeguards approach for MSRs

INPRO: International Project on Innovative Nuclear Reactors and Fuel Cycles

KED: K-edge densitometry

KMP: Key Measurement Point

LEU: Low Enriched Uranium (up to 20% ^{235}U, generally 3-5%)

LLD: Lower Limit of Detection

LWR: Light Water Reactors

MA: Minor Actinide

MAUF: Multi-Attribute Utility Function

MAUT: Multi-Attribute Utility Theory

MBA: Material Balance Area

MC&A: Material Control and Accounting (or Material Counting and Accounting)

MSFR: Molten Salt Fast Reactor

MSR: Molten Salt Reactor

MSRE: Molten Salt Reactor Experiment

MUF: Material Unaccounted For

NDA: Non-Destructive Analysis

NES: Nuclear Energy Systems

NM: Nuclear Material

NMA: Nuclear Material Accountancy

NMAC: Nuclear Material Accountancy and Control

NNWS: Non-Nuclear Weapon State (under the NPT definition)

NPT: Non-Proliferation Treaty

NWS: Nuclear Weapon State (under the NPT definition)

ORNL: Oak Ridge National Laboratory

PIV: Physical Inventory Verification

PIT: Physical Inventory Taking

PNNL: Pacific Northwest National Laboratory

PR: Proliferation Resistance

PP: Physical Protection

Pu: Plutonium

PUREX: Plutonium Uranium Extraction

PWR: Pressurised Water Reactors

RG: Reactor Grade

RM: Remote Monitoring

SBD: Safeguards-by-Design

SF: Spontaneous Fission

SNF: Spent Nuclear Fuel

SNRI: Short Notice Random Inspection

σMUF: Uncertainty on the MUF

SEID: Standard Error of Inventory Difference, uncertainty on the MUF

SMR: Small Modular Reactor

SQ: Significant Quantity

SSAC: State System of Accounting and Control

U: Uranium

TRANSFORM, Triton, CHEMTriton, SCALE: ORNL-developed depletion codes

TRU: Transuranic

WG: Weapon Grade

XRF: X-ray fluorescence

Zr: Zirconium

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Introduction

Nuclear weapons proliferation started with Project Manhattan during the Second World War. After the USA bombings on Hiroshima and Nagasaki, and the evidence of the bombs' destructive potential, it became clear that nuclear weapons should not be used. However, since the USSR achieved nuclear weapon capability, an arms race followed, in which both states ramped up production, scale, and effectiveness of the bombs. The world nuclear arsenal reached tens of thousands of units, enough to effectively destroy Humanity. Nuclear weapons are meant to serve as a deterrent, but just one aggression could be enough to start a chain reaction of mutual destruction. It became evident that it is essential to at least limit the number of players in this dangerous game. The Treaty on the Non-Proliferation of Nuclear Weapons (NPT), established in 1968, sought to curb the spread of nuclear weapons by distinguishing between nuclear-weapon states (NWS) and non-nuclear-weapon states (NNWS). Under the NPT, NNWS committed to forgo nuclear weapons development in exchange for access to peaceful nuclear technology, hard to develop independently by most states at the time, while NWS agreed to pursue disarmament. The NPT mandates that NNWS place all their nuclear activities under international safeguards, enforced by the International Atomic Energy Agency (IAEA) (Article III), through inspections, material accounting, and monitoring, to detect any diversion of nuclear materials for weapons purposes. Thus, the NPT and the IAEA formed a symbiotic relationship: the NPT provides the legal basis for non-proliferation, while the IAEA supplies the technical means for its enforcement.

Despite these safeguards, proliferation attempts revealed gaps in the system. The development of a new approach became necessary: developing nuclear civil systems with proliferation resistance (PR) in mind. Nuclear energy systems (NES) have to become inherently less exploitable for weapons' pursuit. The PR concept should be applied "by-design" in the IVth generation of reactors, which should be intrinsically safe, secure, and proliferation-resistant. The goal of the IAEA and other inspectorates is to promote the inclusion of these requirements in the early stages of design to avoid losses in time and money by ensuring the facilities' readiness to be put under international safeguards. While designers usually have in mind safety (and in a relative way security), it is the role of the inspectorate to accompany them in the development of the safeguardability of their NES. **This thesis aims to be a solid state of the art for what Nuclear Safeguards are, and how they evolve in their social context**, by grouping the often-fragmented information. Historically, the Safeguards approach was studied once the design was complete, and this approach only stood because the different designs had little variation from one another, allowing replicability of safeguard measures. Gen IV is incredibly vast, varying in coolants, fuels, sizes, and applications. A different approach in the deployment of safeguards has to be structured to ensure the efficiency and effectiveness of their application.

Molten salt reactors (MSRs) are a family of reactors of the IVth generation based on the utilisation of molten salt instead of water as coolant. Dozens of designs, at different stages of conception, are being developed worldwide. A group of reactors will use liquid salts both as coolant and fuel, marking a significant difference from the current fuel item system, and making the reactors much closer to a reprocessing facility. It is expected that the current safeguards approach will not apply to liquid-fuelled MSRs and will vary drastically between each design. A paradigm shift combined with disruptive innovation might be needed to effectively meet the safeguards requirements. An early analysis of the applicability of the safeguards approach to the MSRs can be done through a safeguardability assessment that will help to include the safeguards requirements since the early phases of the design. The European Commission, in its role of inspectorate delegated by the IAEA and accepted by the Member States through the Euratom Treaty, is reaching out to several private and public institutions to promote safeguards-by-design practices. ENDURANCE is a project led by Politecnico di Milano to advance the deployability of MSRs and particularly promote a safeguards-by-design approach to designers. This thesis evolves within ENDURANCE. **The first chapter offers a broad view of the PR concept and how safeguards evolve with it. The second chapter focuses on safeguards** and their objectives, functioning, and instrumentation. **The third chapter is a deep dive into the MSRs' technical characteristics**, a presentation of the taxonomies, and a description of the aspects that make the current safeguards approaches not suited for liquid-fuelled MSRs. **The Safeguards' challenges and opportunities are both presented**, along with proposed Safeguards approaches found in the literature.

The scope of this thesis is ultimately to bridge the gap between designers and inspectors, and to establish a basis for assessing safeguardability and implementing safeguards in MSRs. This is an important step to take at this phase of the reactors' development, to be able to include any modification without increased retrofit costs. Parallelly, given the timelines of the inspectorate's new instrumentation development (up to 10 years), any need must be identified now. A safeguardability assessment allows for this double-sided use. A quantitative assessment is sought for its effective application to decision-making and for its ability to provide actionable knowledge. The first step is the development of a safeguardability assessment tool (the SAT), based on a literature review of the available safeguards assessment models. However, these models are often suited only to traditional reactors and a tailoring to the peculiarities of liquid-fuelled MSRs is thus done. The SAT is a list of attributes that should be respected for having a high safeguardability of a plant. The evaluation of these attributes is often qualitative, as safeguardability itself and the capture of experts' knowledge are. The challenge of converting a qualitative assessment into a quantitative one is done through two risk-informed methodologies. The Multi-Attribute Utility Theory (MAUT), borrowed from Decision Theory, is used to capture the safeguards experts' preferences, through the establishment of evaluation parameters and a weight for each attribute representing its importance towards the whole safeguardability. The score for each attribute is aggregated through a utility function to obtain an overall safeguardability score for a facility, which can inform both the designers and the inspectors on where a design is lacking in terms of effectiveness or efficiency of the safeguards approach, thus directing research.

A key aspect of MSRs is the lack of information tied to the early conceptual design phase, in which we still are. A methodology from the Evidence Theory, the Italian Flag Methodology, is applied to tackle the issue of uncertainty and incompleteness of information. This methodology allows for capturing experts' opinions on the success of the process, which in this case is the effective applicability of NDA instrumentation to liquid-fuelled MSRs. The methodology allows for a layered use, which captures relevant information both at a high and a low level of detail. The experts' judgment on the success of the low-level processes of NDA applicability is propagated to the higher level of detail thanks to the dependencies between the processes. The obtained knowledge is particularly actionable in the way the results are presented, which is one of the strengths of the Italian flag. The application of these methodologies is presented in chapters 4 and 5, along with a comparative discussion.

1. Proliferation Resistance

1.1 Historical context of proliferation and non-proliferation efforts

1.1.1 Proliferation

Nuclear fission was discovered in Germany in 1938 by the scientists Otto Hahn and Fritz Strassmann when they bombarded uranium with neutrons, realising it had split into lighter elements. Otto Frisch and Lise Meitner set the theoretical foundations for nuclear fission after the practical experiments and rationalised the energy output that fission produces.

Leo Szilard theorised the nuclear chain reaction as early as 1933. He patented a visionary idea of an ante-litteram nuclear reactor in 1934 (without any clue about how to make it work), even before Hahn and Strassman's experiments. 1939 has been a pivotal year, following the Einstein-Szilard letter, which discussed the possibility of creating a nuclear-chain-reaction-based device that would be extremely powerful and equally dangerous.

The development of the atomic bomb during WW2, within the Manhattan Project, constituted an arms race with Nazi Germany in itself. Once the idea of a bomb had been thought as possible, the major powers raced to be the first to be able to detonate it, perhaps thinking it would win them the war.

In December 1942, the team led by Enrico Fermi achieved the first self-sustaining chain fission reaction in the Chicago Pile 1 experiment, marking the beginning of the first nuclear reactor, which was located under a baseball stadium in Chicago. The "pile" finally allowed Plutonium (Pu) production.

The Manhattan Project pursued both Uranium-based "gun-type" devices as well as Plutonium-based implosion devices. It was thought that both fissile materials could be used in a gun-type bomb, but it became evident that Pu was not suited, and the design changed radically during the project. However, the implosion device is a much more complicated design, and the implosion mechanism required testing. The first nuclear explosion, which was of the implosion type, occurred on the 16^{th} of July 1945 in what was called the Trinity Test, yielding 25 kilotons (kt) of equivalent power of TNT from the fission of only 1.4kg of Pu [1] (which already led to the irradiation of multiple people within and out of the project because no evacuation was done and poor precautions were taken [2]).

By then, Germany had already surrendered, and Japan was probably defeated, but the US, under President Truman's orders, decided to drop the bombs on Japan to supposedly avoid an invasion of the mainland and avoid the deaths of many more (than with the nuclear bombings alone). On the 6th and 9th of August 1945, two atomic bombs were dropped on Hiroshima and Nagasaki, killing cumulatively between 150.000 and 250.000 people, of whom many were immediate. Many died from radiation exposure by the end of the year (to put these numbers into perspective, the 20 years of the Vietnam War resulted in around 58.000 military US casualties).

During the war, the USSR sought to achieve nuclear capability and relied heavily on espionage efforts from within the Manhattan Project. After the end of the war, it eventually obtained the blueprints for the implosion-type device. It first operated a nuclear reactor for plutonium production in 1946, and finally detonated a device on the 29th of August 1949, much earlier than expected by the US intelligence, which had claimed that the USSR would be able to do so only in the mid-1950s.

While it is arguable that such a power should not be held by a single entity, to avoid unilaterality, the achievement of nuclear capability by the USSR led to an arms race between them and the United States that led to the production of thousands of warheads with increasing yields, particularly due to the development of fusion bombs. In the meantime, delivery of warheads radically changed from strategic bombers to intercontinental ballistic missiles.

Both superpowers adopted the doctrine of Mutually Assured Destruction (MAD) as a deterrent. Each one had the capability of obliterating the other, even in case of a highly destructive first strike, in a so-called “second attack”, mainly relying on strategic nuclear submarines carrying intercontinental ballistic missiles deployed around the world. It was a race both in terms of quantity of warheads, but also in terms of technology. Each one was consistently trying to develop a new technology that would give it the edge over the other one, and the world lived in constant fear that this toppling of balance would be an incentive for striking first.

The crisis between the two superpowers is considered to have reached a peak during the Cuban Missile Crisis. A B-59 Russian nuclear submarine that was driven to exhaustion by the US Navy and had been cut from Moscow for several days, believing the war had started, came close to firing a nuclear warhead at US ships (10kt, similar to Hiroshima’s). One of the three officers who had to unanimously give consent, Vasily Arkhipov, refused and possibly prevented a global nuclear war.

After the Cuba crisis, however, production still ramped up until a peak was reached in 1985.

In the meantime, other countries pursued nuclear capabilities. The UK in 1952, France in 1960, and China in 1964 all successfully tested their own atomic bombs. Since then, all five permanent members of the UN Security Council have had nuclear capabilities. Israel allegedly secretly produced nuclear weapons at the end of the 60s. India conducted its first nuclear test in 1974, and Pakistan in the 80s. South Africa developed nuclear weapons at the end of the 70s.

1.1.2 Non-proliferation

It was evident to the nuclear-capable countries that the spread of nuclear weapons had to be stopped. The principle of non-proliferation treaties is that the smaller the number of players and the smaller the number of bombs, the smaller the risk of nuclear annihilation.

The direction taken in this sense by the Baruch Proposal seemed right. In 1946, the USA proposed to undergo full disarmament and the creation of an agency that would, under the UN authority, have the monopoly on nuclear material and control that countries would not pursue nuclear weaponry. It also proposed automatic and exempt-from-the-veto-right sanctions on countries that would do so. This proposal was unfortunately rejected by the USSR because they still feared a US monopoly, especially of know-how, among other reasons. The Baruch plan remained US policy (even though the nuclear program was in great expansion), and its efforts revolved much around developing a civilian nuclear program, as well as the idea that nuclear power could be harnessed for peaceful purposes globally. As part of this effort and its administration's agenda to ease the public's perception of nuclear power, Eisenhower delivered a speech at the United Nations calling for Atoms for Peace. This set the basis for international talks, which fostered discussions for an independent international agency aiming at spreading the use of nuclear power for non-military purposes. The negotiations led to the creation of the International Atomic Energy Agency (IAEA) in 1956 (approval of the statute), which came into force in 1957.

In the meantime, the nuclear powers were still ramping up the production of their nuclear arsenal and particularly testing thermonuclear weapons. Between 1951 and 1958, the US conducted 166 atmospheric tests, the Soviet Union conducted 82, and Britain conducted 21; only 22 underground tests were conducted in this period (all by the US) [3]. Notably, Castle Bravo in 1954 was the most powerful test ever conducted by the US. The yield had been greatly underestimated, and therefore, the evacuation area was insufficient. People suffered radiation exposure, and one Japanese fisherman died from it. The public's awareness began to rise on the risks of nuclear fallout in the atmosphere, and support in the US public for a test ban grew from 20% in 1954 to 63% by 1957. With the launch of Sputnik by the USSR in 1957 (which proved USSR superiority to the US in terms of launching capabilities), Khrushchev's own will, and support in the UK, talks for a ban on nuclear tests began, which led to the Partial Test Ban Treaty (PTBT) in 1963. The treaty banned atmospheric and outer space tests, but not underground tests. This represents the first result towards non-proliferation and disarmament. Not being able to test was also hoped to limit other countries from reaching the bomb. To establish a non-proliferation regime, nuclear-capable states took a chance with the Treaty on the Non-Proliferation of Nuclear Weapons (NPT), which came into force in 1970.

In this asymmetrical treaty, a distinction was made between nuclear-weapon states (NWS), countries that already possessed nuclear weapons before 1968, and non-nuclear-weapon states (NNWS), all other countries that were prohibited from obtaining them. To get the NWS on board, the bargain was that NWS would help NNWS in accessing nuclear energy for civilian purposes, if they agreed to forgo any development of nuclear science for military use and submit to international verifications on the matter. On the other hand, NWS agreed to gradually, but effectively, forgo their nuclear arsenals.

The treaty is organised in three pillars. Non-proliferation is the first pillar, composed of three articles. In the first two articles, NNWS and NWS agree respectively not to pursue nuclear weapons, and not to help others in obtaining them. Of particular interest to this work is Article III, which states that NNWS parties accept the application of safeguards by the International Atomic Energy Agency (IAEA) to verify that they are not diverting nuclear energy from peaceful uses to military ones.

The second pillar is disarmament. As written in the NPT: "Under Article VI of the NPT, all Parties undertake to pursue good-faith negotiations on effective measures relating to cessation of the nuclear arms race, to nuclear disarmament, and to general and complete disarmament". [4]

This article is the only binding commitment in an international treaty to the goal of disarmament.

The third pillar is the use of nuclear science for peaceful purposes, which gives each country the right to do so and allows cooperation between states.

[5] describes well the interaction between the NPT and the IAEA: "The foundations of the safeguards system lie in the IAEA's Statute (Section 12.1.1 of [6]). Article III.A.5 of the Statute authorises the IAEA to establish and administer safeguards. In broadest outline, safeguards comprise four functions: accountancy, containment and surveillance, inspection/in-field verification, and evaluation of information […]. Safeguards are an extrinsic measure comprising legal agreements between the party having authority over the nuclear energy system and a verification or control authority, binding obligations on both parties and verification using, inter alia, onsite inspections."

Other international agreements on non-proliferation have been established. Namely:

- The Treaty for the Prohibition of Nuclear Weapons in Latin America (the Tlatelolco Treaty) opened for signature in 1967.
- The South Pacific Nuclear Weapon Free Zone Treaty (the Rarotonga Treaty) entered into force in 1986.
- The African Nuclear Weapon Free Zone Treaty (the Pelindaba Treaty), opened for signature in 1996;

- The Southeast Asia Nuclear Weapon-Free Zone Treaty (the Bangkok Treaty), entered into force in 1997, and
- The Central Asian Nuclear Weapon-Free Zone Treaty was signed in 2006.

In addition to the regional non-proliferation cooperation efforts listed above, several other agreements are in existence, such as the safeguards system implemented by the European Commission (under the EURATOM Treaty provisions) and the bilateral agreement between Argentina and Brazil establishing a joint inspectorate for verification (ABACC). [5]

In 1996, the UN adopted the comprehensive ban on nuclear tests, including underground ones, but it has not entered into force yet.

In 1997, the IAEA implemented the Additional Protocol (AP), aiming at strengthening the safeguards regime. [5] states: "For States with safeguards agreements and additional protocols in force, the IAEA aims to provide assurance not only regarding the non-diversion of nuclear material for weapons purposes but also of the absence of undeclared nuclear material and activities. However, even for such States, the safeguards system can only provide strong (credible) assurances with the full cooperation and transparency of a State".

After agreeing on limitations to the delivery of nuclear weapons (ballistic and intermediate-range missiles), since 1985, the USSR and the US have been reducing their atomic arsenal thanks to bipartisan treaties like START.

Another example of a successful non-proliferation effort is the dismantlement of South Africa's nuclear arsenal in the 90s.

Figure 1: Estimated global nuclear warhead inventories, 1945-2018 [7]

[8] is provided as a reference article on the matter of the arms race.

1.1.3 Limitations and failures of the NPT

First, the NPT discriminates between two categories of countries. Those "allowed" to have nuclear weapons and those who are not. Second, the disarmament of NWS has not been effective and sufficient. It is argued that the wording of the article is too vague and does not impose action, but the International Court of Justice has recognised the obligation to do so. The disarmament has not been satisfactory, and NWS are ramping up production (China) [9] and modernising their arsenal. Critics argue that the non-nuclear-weapon signatories would be justified in quitting the NPT and developing their nuclear arsenals. On the other hand, four out of the five countries that are not signatories to the NPT possess nuclear weapons (India, Israel, North Korea, Pakistan), and thus no NWS would give up its nuclear arsenal without being sure that no other country possesses nuclear capability. Finally, reducing the number of nuclear warheads could arguably make proliferation more attractive: "Logic suggests that as the number of nuclear weapons decreases, the marginal utility of a nuclear weapon as an instrument of military power increases. At the extreme, which it is precisely disarmament's hope to create, the strategic utility of even one or two nuclear weapons would be huge" [10]. However, when the bargain of the NPT fails to materialise, it erodes mutual trust and builds resentment from NNWS.

The attempts at covert proliferation in Iraq are illustrative of the limitations of the NPT and the safeguards protocol in addressing clandestine proliferation. Even if the AP contrasts this, the adherence to it is not mandatory and depends on the goodwill on cooperation of the state.

North Korea has withdrawn from the treaty and "openly" pursued a nuclear weapons program and tested what is compatible with a thermonuclear explosive device in an underground test.

In Iran, which is a NPT signatory, the IAEA found indications of the existence of a structured nuclear military programme prior to 2003. The uncovering of the construction of two different gas centrifuges enrichment facilities in Natanz and Fordow led to increased international concerns about the nature of the Iranian nuclear programme. The subsequent international discussions led to a number of UN Security Council Resolution against Iran and finally to the negotiation of a Joint Comprehensive Plan of Action (JCPOA) between Iran and the EU/E3+3 (EU, France, Germany, UK, USA, the Russian Federation and the People's Republic of China) to resolve the outstanding issues related to the Iranian nuclear programme (adopted by the UN Security Council as UNSC Resolution 2231). The JCPOA imposed strict limitations to the Iranian nuclear fuel cycle and additional inspections, in exchange of the graded lifting of sanctions. After the unilateral withdrawal of the USA from the JCPA, Iran gradually stopped implementing the related nuclear provisions. Finally, the unilateral military strikes of nuclear sites in Iran is a recent event that contributes weaken the strength of International Law, with potential negative repercussions on the credibility of the NPT and other international treaties and organisations. Countries might be led to consider the option of nuclear latency, meaning they do not possess nuclear weapons but maintain the technological and industrial capacity to develop them quickly if they choose to do so. Such a country remains just below the threshold of weaponisation, allowing it to preserve plausible deniability, remain within international agreements and cooperation, while retaining a breakout option.

1.2. Nuclear Material Acquisition for Proliferation

1.2.1 Achieving proliferation nowadays

"Proliferation – Acquisition of one or more nuclear weapons by a nation or subnational group that currently does not have them" [11].

What does it mean nowadays to pursue proliferation? The essential step is to obtain fissile material. There are different paths that an entity could undergo. The existing literature documents the situation extremely well.

First, the "entity" must be defined. The potential threats of nuclear proliferation constitute a complex spectrum from small terrorist cells to countries with an advanced nuclear fuel cycle. Generally, proliferants are grouped into three categories: subnational/terrorist-like group, non-industrialised state (or state that does not possess sophisticated technology), and developed/sophisticated state that already possesses to some extent a nuclear fuel cycle or nuclear technology. Depending on the depth of the analysis, these three categories can further be subdivided depending on their aspirations regarding the number of weapons sought, weapon yield, weapon reliability, and delivery vehicle [11].

A terrorist organisation could aspire to have only one weapon, as retaliation would not be significant for a group scattered in many small cells. On the contrary, a country at war with an already nuclear-capable state would have much more to lose than to gain in obtaining just one device. As [12] states: *"with enough material verified for only a single weapon, a state proliferator might find itself more of a target for international intervention than possessing a valuable deterrent".*

It is important to note that obtaining fissile material is not equal to obtaining a bomb, which itself does not necessarily imply achieving nuclear capability.

1.2.1.1 Fissile material and design choice

For creating a bomb from scratch, obtaining fissile material (especially if undetected) is practically the most complicated action. Three types of material diversion scenarios are usually considered, and one could choose to follow them all, at risk of leaving more observable evidence of proliferation. A gross defect is when a whole item or batch (usually at least close to an SQ) is removed at once. Since this abrupt diversion scenario is the most detectable one, it is usually coupled with concealment actions trying to hide the anomalies that indicate the diversion and is usually linked to an effort aiming at minimizing the proliferation time. A biased defect occurs when only a small fraction of the declared material is missing, often included in what's acceptable for a given facility, to evade routine measurements unless sensitive tests are applied. This is performed through a diversion scenario protracted in time (typically many years) and easier to conceal. A partial defect is a medium-scale shortfall where some fraction (more than small bias, less than gross) of the declared material is missing from an item or batch, detectable with quantitative verification methods.

An influence diagram, along with other instruments like fault trees and event trees, is a tool for the visualisation of the relationships between systems and events and is applied here to visualise the intricacy of the proliferation through the enrichment of U (or its obtention).

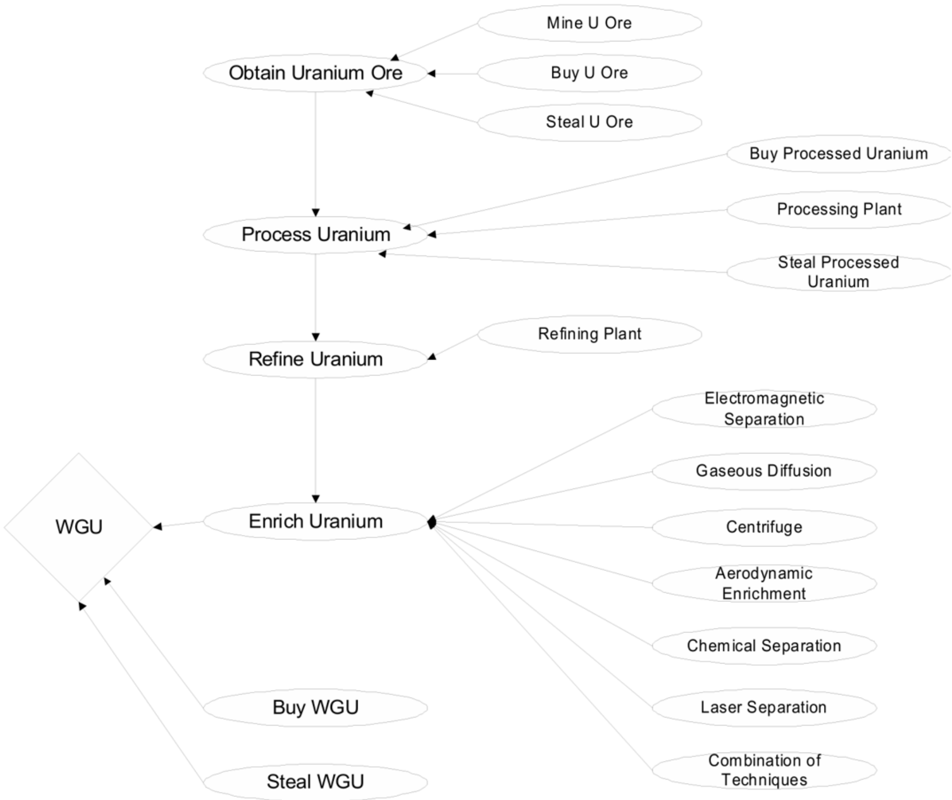

Figure 2: Influence diagram for obtaining WG U [11]

Even when WG material is obtained, making a bomb still involves a combination of several processes that work together to obtain all the other components that constitute it. For instance, detonators are a rather delicate part of atomic bombs. In assessing a state's potential pathways to develop nuclear weapons, it is assumed, based on available literature, that there are two main design options, depending on the nuclear material used [13]:

1. Gun-Type Device: This design achieves criticality by rapidly driving together subcritical masses of fissile material through conventional explosives. While mechanically simple and highly reliable. Even the first highly enriched uranium (HEU) bomb was not tested and deemed directly reliable. However, it is severely constrained by material limitations. Specifically, plutonium with even small concentrations of Pu-240 and Pu-238 is unsuitable due to spontaneous neutron emissions, making HEU the only practical fissile material. It is a design that is usually very difficult to miniaturize, and therefore unsuited for delivery means like ballistic missiles.

2. Implosion-Type Device: This more complex design compresses a subcritical fissile mass (often a spherical "pit" resembling an empty shell) into a supercritical state via precisely engineered explosive lenses, generating an inwardly focused implosive shock wave. Although it requires advanced scientific expertise, it accommodates HEU, weapons-grade plutonium, as well as lower qualities like RG or MOX fuel, and is more compact. This was the design used on Nagasaki's Fat Man. Contrary to the gun-type design, the implosion-type design required testing. [14]

1.2.1.2 Impact of Material Quality on the Bomb's Design

For a gun-type device, even weapon-grade (WG) Pu (ideally with a low Pu-240/Pu-238 content, typically fixed at less than 7% in even-number Pu isotopes and ≤ 0.01% Pu-238) cannot be reliably used, as even isotopes of Pu emit a lot of "stray" neutrons due to spontaneous neutron generation. As in a gun-type device, the sub-critical masses move together relatively slowly (1000 m/s); these stray neutrons may trigger a premature fission reaction and cause only a partial yield in the explosion, known as a fizzle. Reactor-grade (RG) Pu is unusable as the proportions of these isotopes are too high.

Non-fissile isotopes of Pu can also impact implosion-type devices. Stray neutrons can impact the neutron triggers, which are devices used to ensure that the nuclear reaction starts at the exact time the core is most supercritical. Higher concentrations of ^{240}Pu and ^{242}Pu elevate the spontaneous fission rate and can degrade both the surrounding explosive, compromising reliability, and the metallic fissile material, in which even microscopic warps can affect the yield by reducing the symmetry of the implosion.

The presence of the ^{238}Pu isotope, which is a strong alpha particle emitter, increases thermal output through radioactive decay heat (~0.56 W/g vs. ~0.001 W/g for ^{239}Pu). Given that an implosion-type nuclear explosive device requires the use of a high-potential organic explosive and that this explosive surrounds the plutonium core, the heat generated by the plutonium needs to be compatible with what can be tolerated by the explosive before degradation of triggers and electronics or even melting of the explosives (at around 150°C).

[15] states that plutonium becomes significantly less attractive for weapons use by less technologically advanced proliferant states when it contains approximately 8% ^{238}Pu, particularly for nations seeking reliable, high-yield nuclear devices. However, such material may still be viable for either extremely sophisticated state actors with advanced technical capabilities or subnational groups that may accept lower weapon reliability standards.

Complementing these findings, [13] demonstrated that Pu containing up to 9% ^{238}Pu remains potentially weapons-usable, but only with access to advanced technological solutions. Their analysis identifies this 9% threshold as a critical limit, beyond which the thermal output from the plutonium would become unmanageable even for technologically advanced states, effectively rendering the material unsuitable for weaponisation. See Table 4 for neutron emission rates of different isotopes.

These characteristics collectively complicate weapon design, reduce shelf life, and may compromise reliability compared to weapons employing more favourable material compositions. A more detailed description of material attractiveness is done later.

1.2.1.3 Deliverability

Once the material has been collected, and the bomb has been fabricated, a capable and adapted delivery system has to be developed to achieve nuclear weapon capability. A nation that seeks to strike a sophisticated country cannot realistically expect to do so with a conventional bomber, as the country would possess air defences. Intercontinental Ballistic missiles (ICBMs) would have to be bought (but it is unlikely that a sophisticated nation would be selling its technology to an aspiring proliferator), or rather, conceived. Then, a device that must fit in a ballistic missile has to be miniaturised, which further complicates the engineering, which practically rules out the enriched uranium bomb option in the near term or for any country without a history of weapon research. North Korea is currently working towards the fitting of highly enriched U-based bombs in an ICBM. The nuclear device additionally must be able to sustain strong accelerations, pressures, and extreme temperatures during its journey embarked on an ICBM. On the contrary, a terrorist group could use a hidden truck and would not have issues with sizing and additional constraints.

A conservative approach to non-proliferation considers the acquisition of nuclear material as the ultimate goal, but a broader picture can be drawn, an example of which is presented in the influence diagram below.

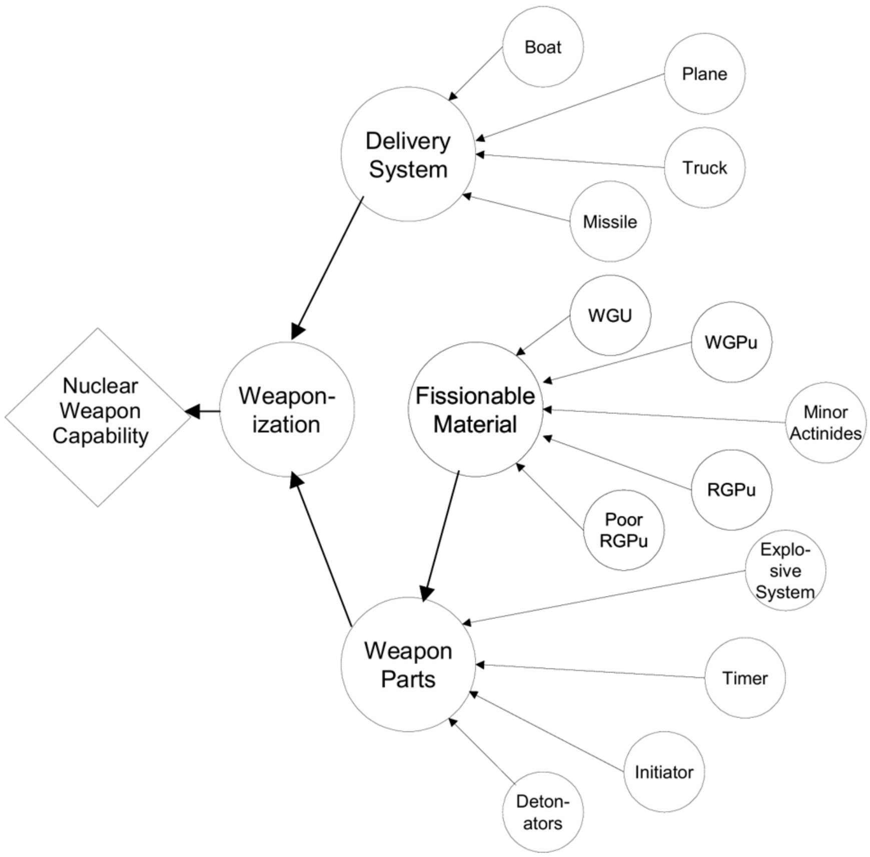

Figure 3: Influence Diagram for achievement of Nuclear Weapon Capability [11]

1.2.1.4 Other extrinsic factors to consider

The proliferation choice begins with the choice of the material, U or Pu. The preferred path choice surely depends on the technical capability of a nation, but especially when covert, external considerations come into play. Time and investment constraints are relevant as far as a proliferator's main objective is not to go undetected. However, if a proliferation path requires ten times the effort and time, but will likely remain covert, it is likely that it would be chosen. Such considerations depend on the geopolitical situation of a State, or the availability of help and resources.

The matter can be illustrated with real-world historical examples, see Table 1 below.

Table 1: Extrinsic considerations by a proliferant state to acquire a specific nuclear material [16]

Extrinsic Consideration	Example
Availability of natural resource	A uranium route may be preferred if the proliferant state has access to natural uranium within its border
Technical capability	A uranium route may be preferred if the proliferant state weighs the importance of achieving success in its first nuclear test, versus the consideration of weapons deliverability
Availability of external assistance	DPRK (Democratic People's Republic of Korea, i.e., North Korea) pursued a plutonium route with the help of a graphite-moderated research reactor and reprocessing technology from the former USSR; and South Africa succeeded in a uranium route with the help of acquiring an enrichment technology from Europe
Domestic policy and politic	Iraq pursued a uranium route using Calutron for enrichment and stayed un-detected by international safeguards for tens of years
International safeguards agreements and obligation	The ease of concealment of clandestine (covert) operation to acquire nuclear weapons capability may depend on the different safeguards agreements obligated by the proliferant state, e.g., IAEA (1968), or IAEA (1972), IAEA (1998a,b), etc.

1.2.1.5 Summary

Overall, nuclear weapon capability is a combination of technical (intrinsic) and situational (extrinsic) constraints and having a bomb does not necessarily mean being able to use it effectively.

As stated earlier, the three main categories for considered proliferators can be further divided according to their aspirations in terms of number, yield, and delivery system. A summarising table can be found below.

Table 2: Nominal Weapons Aspirations of Various Types of Proliferation Threats [11]

Threat Categories		Nominal Weapon Aspirations: number	yield	reliability	delivery	to be stockpiled
1	sub national	1 or 2	any	any	truck/boat	no
2		5 to 10	any	any	truck/boat	no
3	non-industrialized State	1 or 2	any to 20kt	50-95	plane	maybe
4		5 to 10	any to 20kt	50-95	plane	maybe
5		10 to 50	any to 20kt	50-95	plane	maybe
6	developed State	1 or 2	any to 20kt	50-95	plane	no
7		5 to 10	any to 20kt	95	plane	yes
8		10 to 50	20-200kt	95	missile	yes

1.2.2 Material attractiveness

As previously mentioned, not all nuclear material (NM) is equivalent to the production of bombs. Some materials are more suited, and thus more attractive to a given proliferator. This feature is addressed in what is characterised as material attractiveness.

The feasibility of manufacturing a capable nuclear weapon with optimal performance characteristics is fundamentally constrained by the inherent physical properties of the fissile material. Key weapon parameters such as design simplicity, operational reliability, and delivery constraints are directly influenced by these material properties. The characteristics impacting the weapon design are **isotopic content (enrichment, spontaneous neutron generation, heat), critical mass, and chemical form** [5]**.** Metrics exist to assess these characteristics, typically categorising NM in terms of its intended use.

For instance, reactor-grade (RG) plutonium presents challenges due to its undesirable isotopic mix. Nonetheless, it has to be noted that from a non-proliferation legal perspective, there is no discrimination of plutonium according to its isotopic composition, apart from Pu vectors containing Pu-238 at 80% or more.

1.2.2.1 Isotopic composition of the fuel

The Pu isotopic composition's impact has been discussed above and can be categorised as [17]:

- Pu = weapons-grade plutonium, nominally 94% fissile Pu isotopes.

- RG-Pu = reactor-grade plutonium, nominally 70% fissile Pu isotopes in the spent fuel.
- DB-Pu = deep burn plutonium, nominally 43% fissile Pu isotopes in the spent fuel.

For Uranium (U) :

- HEU = high-enriched uranium, nominally 95% ^{235}U, which is WG
- LEU = low-enriched uranium, nominally 5% ^{235}U, up to 20%

A more detailed description, including what RG is for different reactors (these compositions are functions of reactor operating parameters, including the burn-up of the discharged fuels), is summarised in Table 3.

Table 3: Isotopic composition of fissile material produced through various reactors (typical operations), after processing [16]

Pu	***^{238}Pu (wt%)***	***$^{240}Pu/^{239}Pu$***	***Pu Enrichment (wt% odd isotopes)***
WG	*0.01*	*0.062*	*93.9*
RG:			
GMR	*0.2*	0.231	*81.4*
HWR	*0.11*	*0.383*	72.7
LWR	*2.4*	*0.41*	69.8
LWR/ MOX	3.5	*0.585*	61.1
ESFR	*4.2*	0.69	49.0
HTGR	*1.0*	1.058	42.8
MSR	*3.3*	*3.92*	*14.7*
U			***U Enrichment (wt% U-235)***
HEU			*>90*
LEU			*<20*

In nuclear reactors, the output material is never produced as a unique and clean product. It is always a mixture of fertile material, fissile material that has not undergone fission, fissile material produced from breeding of fissionable material, fertile material, fission products and other materials included in fuel. [15] provides figures of merit and evaluation of different fuel cycles at different burnup levels to assess their attractiveness in terms of resistance to proliferation. [18] performs a comparative review of material attractiveness evaluation methodologies for the application to the materials in the sodium fast reactor. These references are provided in case the reader wants to go deeper into detail about material attractiveness for weapon use, but it is out of the scope of this work.

Similarly, the critical mass of the material primarily determines the weapon's minimum achievable size and weight, which are crucial factors for deliverability. Additional considerations include the material's thermal characteristics and isotopic composition, which affect both weapon longevity and engineering requirements.

1.2.2.2 Minor Actinides

It is worth noting that other minor actinides (MAs) and Transuranic elements (TRU) can be weaponised. Even if their direct use might be more complicated from a weapon design point of view, difficulties could always be overcome for a determined actor. Some other materials decay directly into usable ones, such as 233Pa, which decays into ^{233}U. Additionally, legal provisions to safeguard these minor actinides might be lacking, making them more attractive. See Table 4 for theoretically usable isotopes, with the relative bare-sphere critical masses (i.e. the minimal mass needed for achieving an uncontrolled nuclear reaction without the use of reflectors). A smaller critical mass means miniaturisation is easier, and reflector technology needs to be less advanced, decreasing weight. Both features affect the deliverability of a weapon. Note that a bare sphere is not actually the real minimal mass; it is the minimal mass when neutron reflectors or other reaction-enhancing techniques are not used. Even then, not all the NM undergoes fission. In the Trinity test, the fission consumed only 1.4 kilograms out of the 5.9 kilograms of Pu [2].

Table 4: Nuclear Properties of different fissile and fertile isotopes [5]

Isotope	*Half-life (year)*	*Neutrons/ (sec·kg)*	*Watts/ (kg)*	*Critical mass of a bare sphere (kg)*
231Pa	*32.8 x E3*	*nil*	*1.3*	*162*
232Th	*14.1 x E9*	*nil*	*Nil*	*infinite*
233U	*159 x E3*	*1.23*	*0.281*	*16.4*
235U	*700 x E6*	*0.364*	*6 x E-5*	*47.9*
238U	*4.5 x E9*	*0.11*	*8 x E-6*	*infinite*
237Np	*2.1 x E6*	*0.139*	*0.021*	*59*
238Pu	*88*	*2.67 x E6*	*560*	*10*
239Pu	*24 x E3*	*21.8*	*2.0*	*10.2*
240Pu	*6.54 x E3*	*1.03 x E6*	*7.0*	*36.8*
241Pu	*14.7*	*49.3*	*6.4*	*12.9*
242Pu	*376 x E3*	*1.73 x E6*	*0.12*	*89*

241Am	*433*	*1540*	*115*	*57*
243Am	*7.38 x E3*	*900*	*6.4*	*155*
244Cm	*18.1*	*11 x E9*	*2.8 x E3*	*28*
245Cm	*8.5 x E3*	*147 x E3*	*5.7*	*13*
246Cm	*4.7 x E3*	*9 x E9*	*10*	*84*
247Bk	*1.4 x E3*	*nil*	*36*	*10*
251Cf	*898*	*nil*	*56*	*9*

1.2.2.3 Conversion time

A critical parameter for proliferation is the conversion time, especially for breakout (theft) scenarios that do not go unnoticed. This parameter is closely related to the chemical form in which the material is obtained. The closer it is to the metallic form, the faster the weaponisation. A broader distinction is made between direct-use materials and indirect-use materials. A table summarising the conversion times for different chemical forms is found in Table 7.

1.2.2.4 Other material considerations

While the (electromagnetic) radiation field of a material does not directly impact weapon design, it increases the difficulty in accessing the NM, which is why it is evaluated as a material attractiveness metric. A highly radioactive material would pose a deadly threat to the health of the proliferator, thereby increasing the need for either determination or technical sophistication (such as remote handling, shielding, etc.). This is the case for ^{233}U, which is contaminated in parts with ^{232}U, from which the decay chain emits strong gamma rays and is thus highly hazardous, making ^{233}U less attractive for certain aspects than ^{235}U. The same goes for irradiated spent fuel rather than fresh fuel assemblies.

A table summarising the dose rate of some of the different actinides and the spent fuel isotopic mix can be found below.

Table 5: Intrinsic dose Rate of a 4 kg metal sphere @ 1 m from centre, normalised to WG-Pu [16]

Material	Dose Rate
Weapons-Grade Pu	1
GMR (Magnox) Pu	--
HWR (Candu-NU) Pu	--
LWR (LEU) Pu	3.5
233U	3.5
LWR (MOX) Pu	--
ESFR Pu	--

Moreover, the radiation signature increases the detectability of NM. It makes concealed diversion harder to go unnoticed, thereby increasing the need for swift weaponisation, which in turn grows the constraints.

Finally, the material throughput of a facility affects the availability of NM and detection, as it is more likely that a diversion of 1 kg of material will be noticed from a process treating 10 times than in a process treating 1000 times that quantity.

Some intrinsic materials' characteristics do not directly impact the weapon design but affect the desirability of NM. It becomes clear that proliferation cannot be evaluated from the technical difficulty of weapon crafting only. There are proliferation possibilities and barriers to these. A broader mind has to be adopted for analysing proliferation as a process in which every aspect has a part to play for it to be successful.

1.3. Proliferation Resistance: barriers to proliferation and their evaluation strategies

As mentioned in the previous chapter, the degree of difficulty in fabricating a weapon with the desired characteristics depends on the intrinsic material properties. However, these inherent properties not only impact the weapon's design, but also other aspects of the proliferation process. In a broader sense, proliferation is considered to be the process of circumventing barriers in place, which occurs within a specific context and time period. Thus, having already established the existence of intrinsic barriers to this proliferation process, the existence of extrinsic barriers is implied. The next logical step is to study the barriers to proliferation, intrinsic and extrinsic, and how these work symbiotically. This field of study is addressed as Proliferation Resistance (PR). Access to fissile materials remains the most significant barrier to nuclear proliferation, rather than an engineering and theoretical Physics problem, so everything should be done to put into place barriers to material access and its usefulness [19].

The Generation IV International Forum (GIF) Proliferation Resistance and Physical Protection Working Group (PRPPWG), which is the reference for PR&PP for fourth generation systems, provides clear definitions of PR&PP. Proliferation resistance (PR) is the set of characteristics of a NES that makes it the least attractive option when opting between proliferation options for a state actor. The final goal is to impede the diversion or undeclared production of nuclear material or misuse of technology by the host state seeking to acquire nuclear weapons or other nuclear explosive devices. (defined by the IAEA and accepted by GIF).

Physical protection (PP) is that characteristic of an NES that impedes the theft of materials suitable for nuclear explosives or radiation dispersal devices and the sabotage of facilities and transportation by subnational entities and other non-host state adversaries [17]. Given that the considered threats are posed by non-state actors, PP is strictly related to security: if considering a terroristic attack, nuclear facilities would be a critical target. Not only would it have a setback on the country's electricity generation capability, impacting its overall stability, but an eventual radiological fallout would severely affect lives and the ecosystem for potentially decades. PP encompasses the design features that inherently make a NES more secure, considering the previously mentioned threats, and complements extrinsic security features. For instance, one can consider a containment that is designed for resisting penetration or external impacts (intrinsic protection), and a security system made of guards and surveillance (extrinsic). When considering the acquisition of NM by a non-state actor, theft of NM is much more rapid, it requires less sophistication than its production, making it an attractive path countered through PP measures.

According to the PR&PP evaluation methodology [17], the proliferation strategies for a host state actor (through a NES on its own territory) are concealed diversion, concealed production of nuclear material, breakout, and production in clandestine facilities. The PR methodology foresees an acquisition pathway analysis to evaluate the system's response to potential proliferation challenges according to a set of measures and metrics. The considered measures are grouped into two main categories (intrinsic features and institutional measures). The PR methodology foresees an acquisition pathway analysis to evaluate the system's response to potential proliferation challenges according to a set of measures and metrics. The considered measures are grouped into two main categories (intrinsic features and institutional measures).

An ongoing challenge of PR&PP, and especially PR, lies in evaluating the effectiveness of the challenges in a quantitative way (differing from the previous example of low to high). Two main methodologies that try to cross the gap from qualitative to quantitative are considered at the moment.

The GIF implements six PR measures that are quantified for the evaluation in an acquisition pathways analysis:

1. Proliferation Technical Difficulty: The technical sophistication and materials handling capabilities required to overcome the multiple barriers to proliferation
2. Proliferation Cost: The investment required to overcome the technical barriers to proliferation, including needed facilities
3. Proliferation Time: The minimum time required to overcome the barriers to proliferation
4. Fissile Material Type: previously discussed.

5. Detection Probability (DP): The cumulative probability of detecting proliferation attempt in a segment or pathway
6. Detection Resource Efficiency (DE): The efficiency in the inspectorate's use of resources to apply international safeguards to the NES. [17]

An evaluation metric is suggested for each of the 6 PR measures.

Table 6: Nonproliferation barriers and attributes description and their effectiveness with respect to sophistication of the rogue actor [20]

Type of Barrier	Attributes	Attributes description	Sophisticated State, Covert	Unsophisticated State, Covert
Material Barriers	**Isotopic**	The extent to which isotopic composition is suitable for obtaining weapon-grade material	Low	High
	Chemical	Difficulty of chemical processes needed to separate wanted material	Very Low	High
	Radiological	Difficulty of handling and processing of material due to its radioactivity	Low	Moderate
	Mass and Bulk	Quantity of material that has to be processed/stolen for reaching the goal	Low	Low
	Detectability	The degree to which materials may be detected thanks to their characteristics (eg. Radiological signature)	Low	Moderate
Technical barriers	**Facility Unattractiveness**	The extent to which facilities, equipment and processes are resistant to production of weapon-grade material	High	Moderate
	Facility Accessibility	The extent to which facilities and equipment intrinsically restrict access to fissile materials (e.g. PWR core)	Low	Low
	Available Mass	The unattractiveness of a facility or an equipment is inversely proportional to the amount of nuclear material that is processed in/with it	Moderate	High
	Diversion Detectability	The extent to which diversion or theft of materials from processes and facilities can be detected	Moderate	Moderate
	Skills, Expertise and knowledge	The extent to which fuel cycle-related know-how could be transferred to a military programme	Low	Moderate
	Time	The time that materials (and to some extent facilities and technologies) are available to potential proliferators	Very Low	Moderate
Institutional Barriers	**Safeguards**	Measures implemented to assist in the monitoring, detection, and deterrence of facility misuse and/or of material diversion or theft	High	High
	Access Control and Security	The extent to which access control and physical security is a deterrent for diversion of material or misuse of facility	Low	Moderate
	Location	Site remoteness may make a facility harder to attack, but it can also make it difficult to defend and increase the defenders' response time	Very Low	Low

The IAEA has developed another framework that considers the soft layer in which a NES evolves more significantly. The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) offers a comprehensive framework to guide the sustainable development of NES. It adopts a holistic approach, assessing NES across multiple dimensions: economics, safety, environment, waste management, infrastructure, and proliferation resistance. The long-term objective of INPRO is to ensure NES meet energy needs while adhering to global sustainability goals. A key pillar of INPRO's methodology is PR, addressed in Volume 5 of its assessment guidelines.

INPRO is organised as a set of User Requirements (UR) that answer the need posed by a basic principle (BP). The basic principle for PR is "*intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that innovative nuclear systems will continue to be an unattractive means to acquire fissile material for a nuclear weapons program. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself*." [5]

Each UR represents the pillars of the barriers to implement for the NES to be proliferation resistant. Each UR is then divided into several criteria for evaluation (CR) that help the PR assessor in evaluating the NES. Further details are provided, including indicators (IN), acceptance limits (AL), and Evaluation Parameters (EP), to state how a CR can be met explicitly.

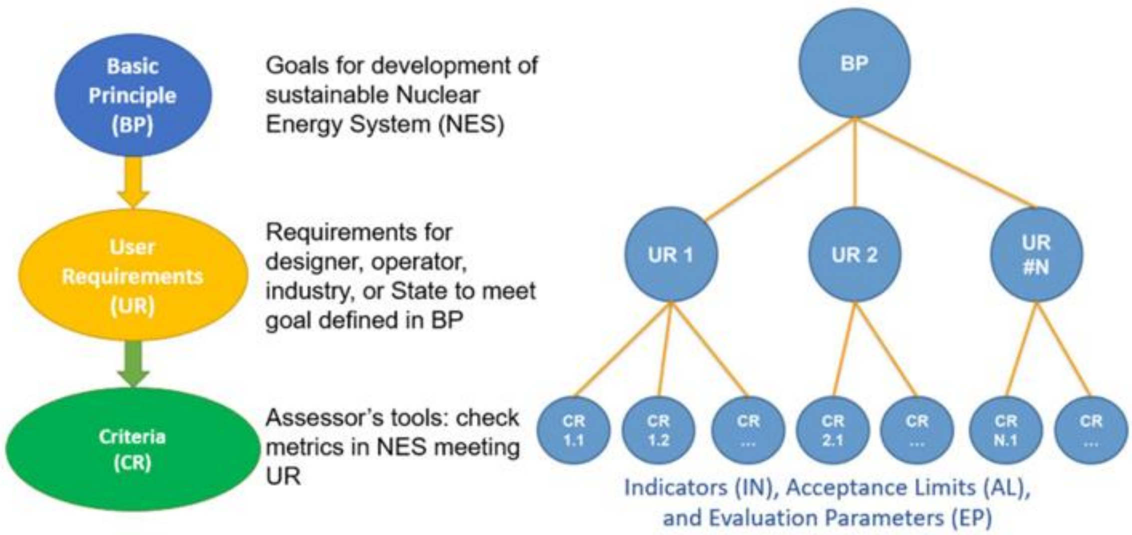

Figure 4: diagram showing INPRO framework and relationship between BP, UR, and CR [21]

An illustrative review of PR in the INPRO methodology is provided below, excluding sections on IN, AL, and EP.

Figure 5: The INPRO assessment in PR from the 2008 manual [21]

In summary, PR&PP refer to the characteristics of a NES that act against the possible proliferation paths for both state and non-state actors. They blend intrinsic characteristics inherent in the design, such as a reactor producing low-quality Pu, with extrinsic traits that include the legislative framework within which the NES is evolving. These characteristics have to be evaluated throughout the fuel cycle, during the whole life cycle of the facilities and NM.

2 Nuclear Safeguards

2.1. Nuclear Safeguards in the PR context

It has not yet been presented clearly to the reader how Safeguards evolve in this PR context. As stated earlier, safeguards are introduced in Article III of the NPT as an "extrinsic measure" applied to NES and are essential to non-proliferation.

Using the NPT's own wording, it is natural to classify them as part of the extrinsic measures that constitute PR. This view is confirmed by an early definition of PR by the IAEA in STR-332 [22] states that "*The degree of proliferation resistance results from a combination of, inter alia, technical design features, operational modalities, institutional arrangements and safeguards measures*." From this definition, we can observe that the contributing factors come into play from the conception of the NES to the everyday application of best practices. The technical limitations are intrinsic to the design, while Safeguards are measures that are extrinsic and must be developed and evolve through a legislative framework. Safeguards evolve at the intersection between a hard technical layer comprising instrumentation, measurements, and verifications of NM, and a soft layer made by the agreements between the operators, the host state, and the IAEA. In an upcoming update to the INPRO methodology, changes have been made to adapt the evaluation of the degree of PR to the ease of enforcement and the effectiveness of the safeguard approach.

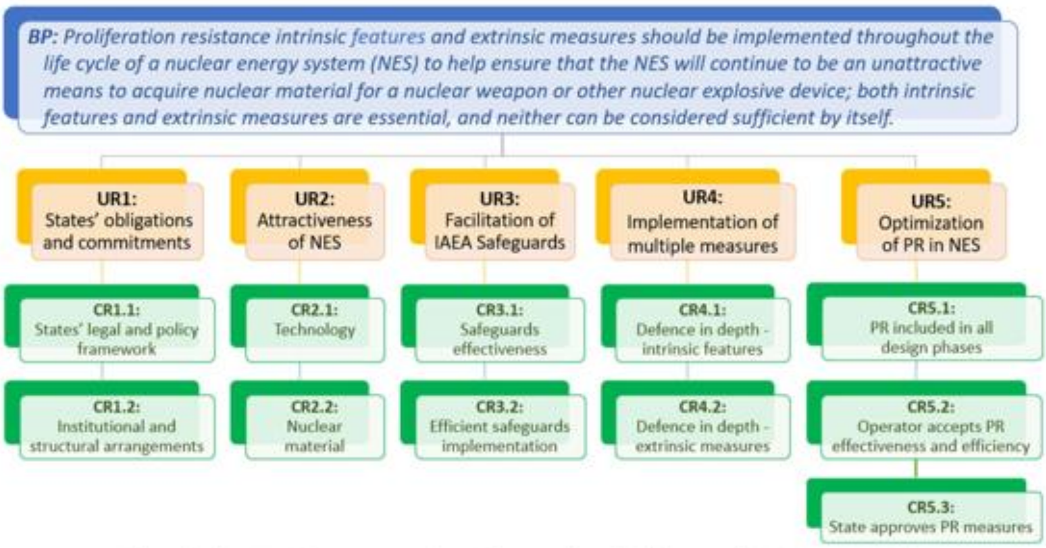

Figure 6: The updated INPRO assessment in PR [21]

Particularly, UR3 has been changed from assessing the difficulty and detectability of diversion to "facilitation of IAEA Safeguards", which inherently works towards increasing the ease in meeting the detection targets (detection efficiency). Further details are given in the Safeguards-by-Design sub-chapter.

2.2. The International Safeguards system

The goal of the IAEA safeguards system is to provide credible and verifiable assurance to the international community that a State is fully complying with its legal commitments to use nuclear material and technology exclusively for peaceful purposes. The legal commitments of a State, and thus the IAEA's specific objective, depend on the agreements between the IAEA and the host state, stipulated with each country. . The Comprehensive Safeguards Agreement (CSA) defines the legal context and the boundaries for the development of a safeguards approach [5]. Under a CSA (enabled by the entry into force of an agreement based on INFCIRC/153 and the IAEA Safeguards standard until 1990s), the IAEA verifies that, given the country's declarations, there is no undeclared production of fissile material, or diversion towards non-peaceful purposes, in declared facilities. The development of the safeguards system's responsibility relies solely on the IAEA.

The discovery of clandestine programs in NNWS in the past (the most notable of which is the Iraqi one) shed light on the fact that despite the legal mandate was covering both declared and undeclared activities, the standard safeguards verification toolbox had substantial weaknesses in providing credible assurance of the absence of clandestine activities in declared and undeclared nuclear sites. To overcome these weaknesses, the 1990s saw the definition of an additional agreement expanding on one hand the amount of information the State has to provide to the IAEA, and on the other hand the verification tools, techniques and equipment that can be devised in a safeguards approach. This new type of agreement, captured in INFCIRC/540 (known as the Additional Protocol - AP) paved the way to the transition from traditional safeguards (ruled by a CSA) to Integrated Safeguards and the current State-Level Approach (SLA). When a host state agrees to the AP, the IAEA is able to provide increased assurance that there have been no attempts at proliferation through declared or undeclared facilities throughout the country.

Within the European Union, the Euratom Treaty establishes a nuclear material supervision system, known as Euratom safeguards, under the responsibility of the European Commission.

The Euratom Treaty explicitly requires the Commission to ensure non-diversion of civil nuclear materials from their intended uses and compliance with the safeguards obligations assumed by the Euratom Community under the relevant international agreements. The IAEA acknowledges the existence of Euratom Safeguards and coordinates and cooperate with the Commission in the discharge of its responsibilities, but maintains its independence from the Euratom Safeguards and draws independent conclusions [23].

2.3. The Technical Objective of Safeguards

The objective of safeguards is the "timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection” [24]. The wording is important. The first term that needs further defining is significant quantity (SQ). An SQ of nuclear material is the approximate quantity of NM with respect to which, considering any conversion process involved, the possibility of manufacturing a nuclear explosive device cannot be excluded [25]. It is important to note that this quantity is related but different from both the bare-sphere critical mass (see above) and the minimum quantity of fissile material required for a bomb (using reflectors). Secondly, the meaning of “timely” has considerable implications. It refers to the timeliness of detection of a diversion. Ultimately, the IAEA must detect a diversion before this proliferation act can lead to weaponisation. The meaning of “timely” is then strictly related to the time needed to collect an SQ through the chosen proliferation path, as well as the conversion time from the collected material to the direct use metallic state and weaponisation. The IAEA has defined a clear frequency and timing with which it must draw its conclusions, as well as the quantities that these conclusions refer to. *Detection time* is defined as the maximum time which may elapse between a diversion and its detection by Agency safeguards” [25].

The inclusion of the expression "or for purposes unknown" is of great importance for the practical application of safeguards. This expression means that the safeguards authority no longer has to prove that weaponisation is occurring, which could be a practically impossible task in some scenarios, and any discrepancy in terms of NM inventory is to be potentially considered a diversion/undeclared production.

The international safeguards objectives are generally defined for the SQ of NM:

- Detect the diversion of 8 kg of plutonium in the form of unirradiated fresh PuO_2 or MOX, within one month of possible diversion, or in the form of irradiated core or spent fuel, within three months of possible diversion.
- Detect the diversion of 25 kg of U-235 in the form of unirradiated highly enriched uranium (HEU, U-235 ≥ 20%), within one month of possible diversion, or in the form of irradiated core or spent fuel, within three months of possible diversion.
- Detect the diversion of 8 kg of U-233, bred from thorium, in the form of irradiated fuel within three months of possible diversion, or separated U-233 within one month of possible diversion.
- Detect the diversion of 75 kg of U-235 in the form of depleted, natural, or low-enriched uranium (LEU; U-235 < 20%), within one year of possible diversion.

- Detect the diversion of 20 tonnes of thorium within one year of possible diversion.
- Detect possible facility misuse for the undeclared irradiation and production of plutonium and/or U-233, or other undeclared nuclear activities.

It is directly understandable that the timeliness objectives are related to the conversion times. A summary table is found below.

Table 7: Estimated material conversion time for finished Pu or U metal components and SQs of NM [5]

Direct use nuclear materials		
Beginning material form	**Conversion time**	**Significant quantity (SQ)**
Pu, HEU (^{235}U ≥ 20 %), or ^{233}U metal	Order of days (7-10)	Pu[a] 8 kg Pu ^{233}U 8 kg ^{233}U HEU (^{235}U ≥ 20 %) 25 kg ^{235}U
PuO_2, $Pu(NO_3)_4$ or other pure Pu compounds; HEU or ^{233}U oxide or other pure U compounds	Order of weeks (1-3)	
MOX or other non-irradiated pure mixtures containing Pu, and U (^{233}U + ^{235}U ≥ 20 %)	Order of weeks (1-3)	
Pu, HEU and/or ^{233}U in scrap or other miscellaneous impure compounds	Order of weeks (1-3)	
Pu, HEU or ^{233}U in irradiated fuel	Order of months (1-3)	
Indirect use nuclear materials		
Beginning material form	**Conversion time**	**Significant quantity (SQ)**
U[b] containing < 20 % ^{235}U and ^{233}U;	Order of months (3-12)	75 kg ^{235}U (or 10 t natural U or 20 t depleted U)
Th	Order of months (3 - 12)	20 t Th

[a] For Pu containing less than 80 % ^{238}Pu.
[b] Including low enriched, natural and depleted uranium.

The diversion activity is assumed to be part of a planned sequence of actions chosen to give a high probability of success in manufacturing one or more nuclear explosive devices with minimal risk of discovery until at least one such device is manufactured. Given the complexities and cost to develop a weapons program, timely detection by the IAEA is generally measured in months [26]. Timeliness objectives can be grouped by category of material. In States with no AP in force or where the IAEA has not drawn and maintained a conclusion of the absence of undeclared nuclear material and activities, the timeliness detection goals are at most:

- One month for unirradiated direct use material (UDU),
- Three months for irradiated direct use material (IDU),
- One year for indirect use material, such as fertile material.

- In particularly sensitive operations within the fuel cycle, the detection of an anomaly should be instantaneous and inspector's presence on site is continuous

Longer timeliness detection goals may be applied in a State where the IAEA has drawn and maintained a conclusion of the absence of undeclared activities.

2.4. The Safeguards approach and implementation

The international safeguards protocol is composed of three main areas of measures to take for the IAEA, namely design information verification (DIV), nuclear material accountancy (NMA), and containment and surveillance (C/S).

2.4.1. Design Information Verification

DIV refers to the independent verification that the nuclear facility is consistent with the blueprints and that its technical description accurately corresponds to its physical design and operational purpose. The core purpose of DIV is to establish a reliable baseline through verification, identifying any potential misuse of the facility for clandestine purposes by detecting unreported modifications or inconsistencies. DIV include the initial examination of design information questionnaires submitted by the State, followed by systematic on-site inspections.

2.4.2. Nuclear Material Accounting and Control and Nuclear Material Accountancy

The first layer of the international safeguards system is the Nuclear Material Accounting and Control (NMAC) of the nuclear facility operator. It is a set of procedures aimed at establishing and verifying the NM quantities in the system.

The IAEA (2002) [22] defines Nuclear Material Accounting as: "*Activities carried out to establish the quantities of nuclear material present within defined areas and the changes in those quantities within defined periods. Elements of nuclear material accounting include establishment of accounting areas, record keeping, nuclear material measurement, preparation and submission of accounting reports*".

Control refers to "*the verification of the correctness of the nuclear material accounting information*", meaning the operator's own procedures to control the accuracy of its accounting system, which provides the data for the safeguards declarations.

The facility operator tracks the NM that, given deliveries, usage, and storage, *should* be in the facility, and records it in the so-called *book inventory*. Accounting functions through the establishment of Material Balance Areas (MBAs), which are areas that may or may not physically exist and represent the perimeter within which NM is accounted, and the book is established. The whole nuclear facility could be considered a single MBA, but it is not necessarily the case. A formal definition is provided by [5]: "*MBA is an area in or outside a nuclear facility for which the quantity of nuclear material in each transfer into or out of it can be determined with specified procedures in accordance with the requirements for safeguards*". **The operator must perform periodic physical inventory takings (PIT-typically one per year)**, in which at a given point in time the NM that is actually in the MBA is accounted for. The PIT marks the closure of a Material Balance Period and the beginning of a new one. Any further modification of the inventory until the next PIT will be recorded and tracked incrementally. **The operator implements its own accounting based on the domestic safeguards regulations**.

The second layer of the International Safeguards is the State System of Accounting and Control of nuclear material (SSAC). States are required under the NPT to establish and maintain an SSAC [26] the role of which is to collect the declarations from the operators and transmit them in the proper format to the IAEA In the EU, this role is taken by the European Commission through the Euratom inspectorate. Even though some peculiarities might appear in the domestic safeguards taxonomy worldwide, NMAC is interchangeable with SSAC as it is directly based on it.[1]The second layer of the International Safeguards is the State System of Accounting and Control of nuclear material (SSAC). States are required under the NPT to establish and maintain an SSAC [26] the role of which is to collect the declarations from the operators and transmit them in the proper format to the IAEA. In the EU, this role is taken by the European Commission under the Euratom safeguards provisions. [26]

The third layer is the international verification of these domestic procedures by the Inspectorate (IAEA or Euratom). **The inspectorates rely on Nuclear Material Accountancy (NMA)** as a set of procedures aimed at verifying the operator's NMAC (and the State's declaration). The NMA is the backbone of International Safeguards, and the related inspector activities revolve around the verification of the operator's NMAC sheets, material flows, and **verification of the consistency of the NM declarations with what is found in the facility** (e.g. not replaced by dummies). After the operator's PIT taking, inspectors perform a physical inventory verification (PIV) [24], aimed at confirming the correctness of the PIT reports declared to the inspectorate. In some MBAs there could be a non-zero difference between the inventory declared in the PIT and the inventory verified during the PIV, which

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[1] Confusion appears at a semantic level given the different adaptations of the NMAC concept, particularly when referring to the USA safeguards. In USA domestic safeguards, NMAC is referred to as Material Control and Accountability (MC&A). Another appellative is Nuclear Material Control and Accounting (NMC&A) In the USA, formally, NMAC (MC&A) and SSAC differ in their overall purpose. While SSAC addresses, through safeguards, exclusively a State-threat, MC&A addresses more generally a non-State threat. Thus, Control refers to the implementation of any measures preventing diversion by a non-state actor and is more relevant to security rather than safeguards. Practically, the two converge and are generally merged. For any further information, refer to [26].

represents the **Material Unaccounted For (MUF)** (or Inventory Difference (ID) in domestic safeguards; the two terms refer to the same thing [27]). MUF is inevitable and expected in MBAs where the NM is not restrained in items, namely in bulk **handling facilities**, for which measurements uncertainties makes the expected value of the **MUF different from zero**. The MUF itself is subject to uncertainty due to the accuracy of the measurements, the need to estimate the amount of hidden inventory and process hold-ups and other similar sources of uncertainties. Where the NM inventory is made of items whose integrity is never lost (item-based facilities), MUF needs to be zero. The independence of the verification is stipulated in the NPT and builds the trust it seeks to establish.

The interaction between the three layers is displayed in the figure below.

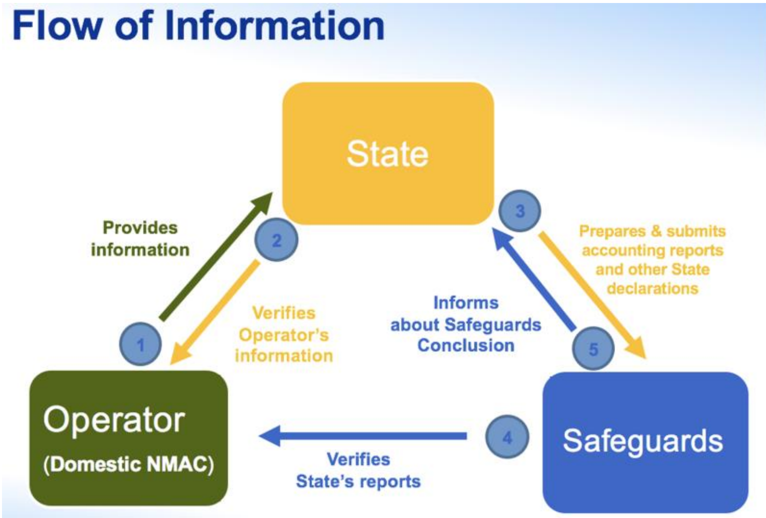


Figure 7: Flow of Information between the Safeguards Actors [26]

The inspector must rely on instruments he can trust, which are usually not the ones the operator necessarily uses for its verification purposes and must be able to conclude on diversion attempts. The detectability of NM is *inter alia* related to the ease of identifying the type, composition, and radiation signatures of the NM, which could be used for the identification of NM and for the application of monitoring systems [5].

The available nuclear material characterisation techniques can be grouped into Destructive Analysis (DA) and Non-Destructive Assays (NDA). For DA, the inspector takes samples of material that are analysed, generally through mass spectroscopy, and not reinserted into the system, thus "destroyed". The isotopic composition of the sample is precisely determined; however, the cost, accommodations, and required time are high. NDA relies on external measurement and is acknowledged as less expensive and less time-intensive [5]. NDA will be further described in 2.6.

These techniques are applied to samples taken from, or to flow monitoring at, certain strategic points called **Key Measurement Points (KMP)**, which are locations where essential information on flow and inventory can be gathered and verified and at which nuclear material appears in such a form as to lend itself to such measurements [25]. The entrance and exit points of MBAs are often KMPs, for instance. Not all the material is sampled and verified. Here, too, there is a statistical approach that aims at characterising all the material by verifying a portion of it. The conclusions of non-diversion are then also subject to the statistical approach used for the sampling plan. [25] states that "*The technical conclusion of the IAEA's verification activities must include an estimate of the operator's combined measurement uncertainties and the operator's MUF adjusted for any differences between the IAEA's and the operator's measurements. This technical conclusion gives an indication of the accuracy of the IAEA's measurements and of the degree of agreement between the operator's measurements and those of the IAEA*". MUF is discussed in 2.5.2.2

2.4.3. Containment and Surveillance

The purpose of C/S is to reduce the workload of safeguards inspections while maintaining the Continuity of Knowledge (CoK) between inspections by monitoring activities during the absence of the inspectors onsite. The main goals of C/S measures are to ensure the integrity of samples, to confirm that previously verified material has not been changed and thus reduce the need for reverification, to ensure that the instrumentation and material of the inspectorate have not been tampered with, and finally to isolate items that need to be verified until it can be done [5] [28]. Surveillance is mainly constituted of surveillance cameras, useful for tracing and reconstructing the movements of nuclear material to verify the declarations of the operator. A containment measure leverages existing structural characteristics to maintain the physical integrity of an area or item by preventing the undetected access or movement of nuclear material or equipment, thereby ensuring the continuity of knowledge [5]. The working horse of containment measures is tampering-indicating seals, the goal of which is not to effectively seal off a contained item by rendering it inaccessible (e.g. locks), but to indicate whether the item has been accessed between inspections.

2.4.4. The Diversion Attempts' Detection through Inspections

2.4.4.1. The types of inspections

The three safeguard instruments described above (DIV, NMAC, C/S) are implemented through inspections. There are several categories of inspections based on purpose: routine, ad hoc, and special.

- Ad Hoc Inspections: Focus on verifying baseline information, including design data, initial reports, later modifications, and international transfers.

- Routine Inspections: Conducted to ensure a state's operational records align with its declared nuclear material, confirming its location, identity, and quantity. They also investigate measurement inconsistencies and the MUF.
- Special Inspections: Used when the provided information is inadequate. Their purpose is to verify the contents of special reports or to collect the additional information needed for the IAEA to complete its verification duties.
- Additionally, in the AP framework, Complementary Access Unannounced Inspections and Short Notice Random Inspection (SNRI) are an additional tool to improve the inspectorate's effectiveness, especially regarding timeliness.

2.4.4.2. The Inspection effectiveness metric

Finally, the measurable objective of an inspection is the detection probability (DP), which refers to the minimum probability of reaching the technical objectives of the IAEA's safeguards (and previously defined in The Technical Objective of Safeguards). Emphasis should be placed on the fact that the safeguards goal is only to detect and deter diversion through the risk of detection, and not to prevent it, which is a more general aspect of proliferation resistance. As mentioned earlier, this probability depends on factors used in establishing the inspection plan, such as the sampling distribution. There are set intervals in which the probability and the confidence level must be.

The sought DP can go up to 95% [25].

2.4.4.3. The DP in Practice

Some more specific examples can be found below; however, the objectives are defined on a case-by-case basis by the inspectorates.

For HEU and separated Pu not stored under IAEA safeguards seals or surveillance, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, to have a 90% probability of detecting the diversion of 1 SQ of the nuclear material.*
- *To have a 50% probability of detecting the diversion of 1 SQ of the nuclear material within 1 month of its diversion.*

For spent fuel (SF) and other irradiated materials containing Pu, HEU, or ^{233}U, not stored under IAEA safeguards seals or surveillance, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, to have a 50% probability of detecting the diversion of 1 SQ of the nuclear material; and*

- *To have a 20% probability of detecting the diversion of 1 SQ of the nuclear material within 3 months of diversion.*

For unirradiated depleted, natural, or low-enriched uranium and unirradiated thorium, not stored under IAEA safeguards seals or surveillance, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, to have a 50% probability of detecting the diversion of 1 SQ of the nuclear material.*

For HEU and separated Pu stored under IAEA safeguards seals, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, to have a 50% probability of detecting tampering of the seals*
- *During the transfers between MBAs, to have a 50% probability of detecting tampering of the seals*
- *To have a 20% probability of detecting tampering of the seals within 1 month of diversion.*

For other nuclear material stored under IAEA safeguards seals, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, to have a 20% probability of detecting tampering of the seals*
- *During the transfers between MBAs, to have a 20% probability of detecting tampering of the seals*
- *For irradiated materials to have a 20% probability of detecting tampering of the seals within 3 months of diversion.*

For nuclear material stored under IAEA seals or surveillance, the IAEA detection goals are:

- *During the annual Physical Inventory Verification, if review of the surveillance records does not provide any indication of diversion, then re-measurement or item counting is performed on the material to give a 10% probability of detecting the diversion of 1 SQ of material.*
- *During transfers between MBAs, to have a 90% probability of detecting the diversion of 1 SQ of HEU or separated Pu, and to have a 50% probability of detecting the diversion of 1 SQ of other nuclear material.*

These objectives are directly taken and given *as is* from [16] and serve for illustrative purposes (a paraphrase would not be adequate given these objectives' formulations and their legal connotation). Case-by-case objectives are always to be decided by the inspectorate.

2.5. The Typical Safeguards Practices in Different Facilities along the Fuel Cycle

2.5.1. The Nuclear Fuel Cycle

Nuclear power facilities (producing power through nuclear fission in reactors) are the most common and most well-known nuclear facilities. However, NM is sensitive and must be safeguarded throughout all its existence, from the uranium ore to the disposal of spent fuel. The set of all processes that the NM undergoes constitutes the nuclear fuel cycle.

Uranium is abundant in the earth's crust, as it is included in small percentages (around 4ppm) in the granite. Deposits in which the uranium concentration is high enough for making its exploitation profitable are called uranium ore.

The fuel cycle is split into two. The front end is regarded as the processes that the NM undergoes before its irradiation (e.g. mining, conversion, fuel fabrication). The back-end encompasses the processes after its irradiation (e.g. irradiated fuel reprocessing, storage, disposal). The fuel cycle can be "once-through," meaning that no fuel reprocessing is done for further fuel fabrication, or "closed."

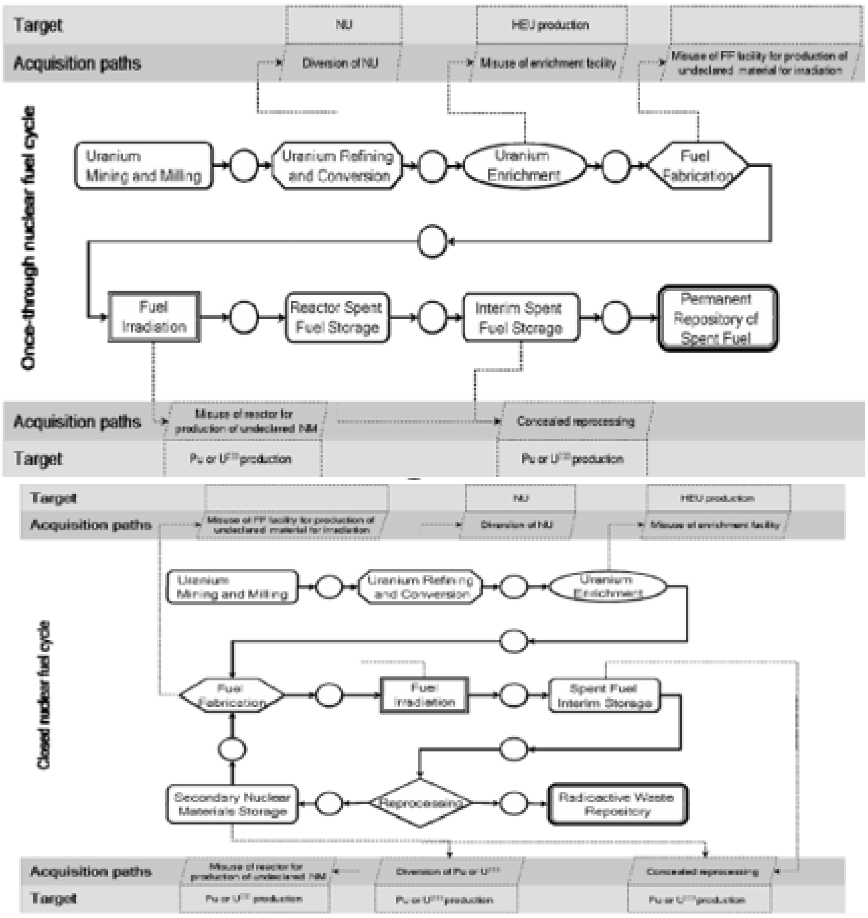


Figure 8: Once-through and closed fuel cycles with examples of diversion pathways [5]

For understanding the peculiarities of the safeguards approaches along the fuel cycle, it is necessary to go through a description of what makes each step of the fuel cycle unique in terms of processes. This is described in Appendix.

2.5.2. Nuclear Safeguards Approaches along the Fuel Cycle

In line with the goals of the IAEA, the entire nuclear fuel cycle must be safeguarded to provide a State-wide non-proliferation assurance. The INPRO initiative highlights the fact that innovative nuclear systems must consider that they are both a facility and part of a larger cycle. An integration must be done to allow for smooth safeguards implementation. It is then natural to ponder whether the safeguards approach can be unique across the fuel cycle. However, given the multitude of forms and proliferation risks, it becomes obvious that the safeguards approaches must evolve, directly driven by the changing proliferation sensitivity and the practicality of verification.

2.5.2.1. Item vs. Bulk Processes

The review of the different steps of the nuclear fuel cycle has shown that the chemical and physical form of the fuel changes throughout the cycle. What also changes are the handling methods. Namely, the management of NM can use an item-based approach (i.e. involving discrete entities, for instance fuel assemblies), or a bulk-handling approach (like in reprocessing or enrichment). An item is a unique, inseparable, and identifiable object.

Differently, from item-based processes, in bulk-handling processes the nuclear material is held, processed, or used in loose forms—liquids, gases, powders, or pellets—and may change chemical or physical form during processing. Such processes are usually performed in batches. According to the IAEA Safeguards Glossary, a batch is defined as *a portion of nuclear material for which the composition and quantity are defined by a single set of specifications or measurements. One batch may be composed of one item or a number of separate items, or may be in bulk form as a whole* [29]. In bulk-handling processes the risk of gradual, undetectable diversion of small amounts of NM (i.e. *bias defect*), therefore the accountancy for this type of facilities must be based on the accurate assessment of the Material Unaccounted For (MUF).

2.5.2.2. Focus on the MUF for Bulk Handling Facilities

The MUF is defined as the difference between book inventory and physical inventory [29]. The objective is to achieve an amount of MUF that is statistically compatible with the hypothesis "the MUF is zero" (i.e. the discrepancy we see is given only by measurement inevitable inaccuracies and not by the fact that NM is really missing). MUF is calculated for an MBA over a material balance period using the material balance equation:

$$MUF = (PB + X - Y) - PE \qquad 1)\ [5]$$

Where PB is the beginning physical inventory, X is the sum of increases to inventory, Y is the sum of decreases from inventory, and PE is the ending physical inventory [5].

The material balance is usually calculated for each species of interest. In a U-Pu cycle, the isotope of interest would be Pu and a material balance for just Pu is established [30].

For item MBAs, MUF should be zero; a non-zero MUF should be investigated as it's usually of accountancy inconsistencies. On the other hand, in bulk-handling MBAs, a non-zero MUF is expected because of measurement uncertainty and the nature of processing of nuclear material in bulk form.

Therefore, in bulk-handling facilities, a possible proliferation path is to conceal a cumulated bias defect (protracted removal of small quantities of NM) within the MUF. This might or might not be accompanied by an overstatement of the true measurement uncertainty of MUF (As previously stated, the MUF and its uncertainty are derived from statistical methods and are subject to deviations). The diverter may also attempt to prevent its appearance in MUF by the falsification of flow or inventory data. Any of the following techniques or a combination of them can be pursued for this purpose: understate receipts, overstate shipments or declare non-existing ones, overstate discards, or overstate the physical inventory. The optimum diversion strategy is a combination of diversion by inclusion in the MUF and falsification of flow or inventory data.

To counter this, the IAEA has established a standard value, referred-to as International Target Value (ITV), for the expected uncertainty (standard deviation of the uncertainty) of MBA's measurements during PIV. This target value depends on the associated risk of a facility and varies from each step of the fuel cycle, while staying consistent if the applied technology changes (same value for diffusion or centrifuge enrichment, but a different one for enrichment and reprocessing). It also depends on the precision of the available and commonly used instrumentation.

Table 8: Expected measurement uncertainty δ_E associated with closing a material balance [5]

Type of Bulk handling facility	δ_E
U enrichment	0.002
U fabrication	0.003

Pu fabrication	0.005
U reprocessing	0.008
Pu reprocessing	0.010
Separate waste storage	0.25
Separate scrap storage	0.04

The operator's measurement uncertainties associated with each of the material balance areas are combined with the material quantities to determine the uncertainty of the material balance, σ_{MUF}, according to the formula below:

Let σ_{MUF} be the uncertainty on the MUF,

Let TP be the throughput of the facility,

Let INV be the inventory of the facility,

Then:

$$\sigma_{MUF} = \max(TP; INV)*\delta_E \qquad 2)\ [5]$$

For assessing the admissible uncertainty of the MBA closing, the condition is:

$$|MUF| \leq \sigma_{MUF} \qquad 3)\ [5]$$

(The absolute value is taken as MUF can be positive or negative)

A crucial metric for evaluating the detection efficiency of safeguards is the Low Limit of Detection (LLD). Given the standard deviation of uncertainty in the MUF determination, δ_E, then the amount of the diverted special nuclear material that could be detected with a 95 % detection probability and a 5 % false alarm rate (the standard safeguards goal) is:

$$LLD = 3.3\delta_E \qquad 4)\ [5]$$

This result can be derived from the following well-established safeguards formula, by describing the uncertainty as a shift in the mean value distribution for the MUF:

$$\Phi^{-1}(1-\alpha) + \Phi^{-1}(1-\beta) = \frac{SQ}{\sigma} \qquad 5)\ [31]$$

Where α is the false alarm probability, $1-\beta$ is the detection probability, σ is σ_{MUF}, and Φ^{-1} is the inverse Gaussian function. Appendix A of [32] provides detailed calculations.

For α=0.05 and 1−β=0.95, an SQ of 8kg of Pu allows for an uncertainty in measurement of 2.4 kg to detect a diversion. This is further discussed with practical reactor examples in 3.4.4.1.

A detection limit is thus set by the standard deviation of σ_{MUF}, deriving from the uncertainty in measurements and sampling.

2.5.3. The Application of the Safeguards Principles within the Facilities

The application of IAEA safeguards is meticulously tailored to each facility type in the nuclear fuel cycle, with the technical approach and instrumentation dictated by the material's form (item vs. bulk), chemical composition, radioactivity, and strategic significance. The fundamental objective remains consistent: to verify the absence of diversion. However, the tools and techniques vary dramatically. [2] This section is designed to define a reference set of safeguard approaches, allowing for comparison with the approaches or instrumentation required for innovative systems.

2.5.3.1. Item Counting Facilities (e.g. LWRs)

LWRs, which make up the vast majority of operative NES, primarily involve the safeguarding of discrete, identifiable fuel assemblies. The measurement challenge is not primarily a quantitative assay of the nuclear material inside each assembly, but rather the identification and counting of items, coupled with ensuring their integrity and identity (no replacement with dummies). For fresh fuel, verification involves item counting and checking unique serial numbers against records. **The enrichment of the fresh LEU fuel (typically 3-5% U-235) is certified at the fabrication plant and is not re-quantified at the reactor**. More qualitative measures are done instead (e.g. presence of LEU rather than its exact quantification). The key instruments during operation are optical surveillance systems to monitor fuel storage areas and reactor hall access points. Their "accuracy" is binary: they must reliably detect the movement of a whole fuel assembly. For core verification, after refuelling, inspectors might visually count the assemblies in the open core. Tamper-indicating seals are then applied to the reactor’s core lid; these are mechanical or electronic devices whose function is to indicate any attempt at unauthorised access with a very high probability of detection. For SNF, the high radiation field imposes remote handling methods**. A random sampling is followed by a modelling and measurement of the energy and intensity of gamma rays from isotopes like Cs-137 and Cs-134 through a passive gamma emission tomography.** This allows inspectors to verify the cooling time and, through correlation, the burn-up, providing a consistency check against operator declarations. The accuracy of such gamma spectroscopy for burn-up verification is typically within 5-10% [25]. This is combined with the measurement of the collimation

[2] Disclaimer : this section should be considered as illustrative only, not as an actual set of safeguards objectives and measures throughout the fuel cycle. This is particularly true given that the main source [25] might be outdated, but is used given its comprehensiveness.

of the Cherenkov light inside the fuel assembly channels, which allows for the qualitative verification of the presence of SNF. The reactor type and the relative approach are further described in Annex B of [33].

2.5.3.2. On-load Fuelled Reactors

The challenges multiply at on-load fuelled reactors (e.g., CANDU), which contain large inventories of small, natural uranium fuel bundles that are frequently moved. Item counting remains a primary tool, but the sheer number of bundles (tens of thousands) makes full manual counts impractical. Here, the IAEA relies heavily on bundle counters to independently verify the number of bundles discharged from the core. Bundle counters are passive radiation monitors (NDA approach) or optical systems installed on fuel transfer channels. This provides an example of how the NDA signatures can inform on the operational state, with observable peaks in neutron activity after a bundle removal. Their accuracy must be near 100% counting efficiency to detect the loss of even a single bundle. The verification of spent fuel bundles in storage ponds often uses underwater CCTV cameras and scintillation detectors (e.g., NaI) for gross gamma counting to confirm irradiation status.

2.5.3.3. Conversion, Enrichment, and Fabrication Plants

For enrichment plants, Safeguards focus on verifying the mass and enrichment of uranium in cylinders at feed, product, and tails stations. Cylinders are weighed on large, high-precision scales with accuracies of ±0.05% or better. High precision is paramount, given the facility's very nature. Especially in tails that have a concentration of fissile U that can be as low as 0.1%, this enrichment must be verified to the greatest extent possible, and the total mass of a cylinder, if accurate, is greatly informative, given that enrichment results in a difference in mass [25].

The primary instrument for measuring enrichment is the gamma spectrometer; however, for UF_6, the high density requires high-energy gamma rays. The 1001-keV gamma ray from Pa-234 (a decay product of U-238) is used. By comparing the intensity of this line to the 186-keV line from U-235, the enrichment can be determined without opening the cylinder. Modern systems can achieve accuracies in U-235 enrichment measurement of better than 2% relative (e.g., ±0.08% for a 4% enriched sample). DA can be applied via thermal ionisation mass spectrometry to provide the definitive enrichment value with an accuracy better than 0.1%. A comprehensive list of available instruments, with the corresponding target, has been compiled by the ITV [34].

At conversion and fabrication plants handling LEU, the IAEA verifies the operator's measurements through random sampling. High-precision scales are used to weigh UF_6 cylinders, oxide powders, and pellets. Accuracies for large masses (e.g., a full UF_6 cylinder) are targeted to be within ± 0.1% of the gross mass. For verification of NM without opening containers, gamma spectrometers are used. For uranium, the 186-keV gamma ray from U-235 is identified which allows to establish the enrichment. The mass of U-235 is then obtained through weighting. The accuracy of such measurements for determining U-235 mass in UF6 cylinders or finished fuel assemblies can range from 1-5%, depending on the homogeneity of the material and the calibration standards [25]. In facilities handling plutonium, neutron coincidence counters are essential. These devices allow to distinguish the spontaneous fission neutrons from even-numbered plutonium isotopes (e.g., Pu-240), the neutrons from (α,n) reactions, and the subsequent induced fissions. They are essential to determine the NM mass in items like oxides or scrap with an accuracy of 1-3% for well-characterised items, but this can be worse for impure or heterogeneous materials. In these cases, inspectors may use the neutron multiplicity counter (PSMC, Plutonium Scrap Multiplicity Counter), which allows a more "generalized" measurement at the price of a more detailed description of the setup geometry. DA provides the highest accuracy and is the benchmark for calibrating NDA instruments. Samples are taken to off-site laboratories for isotope dilution mass spectrometry, which can determine uranium and plutonium isotopic concentrations and elemental masses with accuracies better than 0.1% [25]. DA after an environmental swipe sampling are of paramount importance.

The difference in mass measurement accuracy targets for UF_6 cylinders—±0.1% at conversion plants versus ±0.05% or better at enrichment plants, is not an inconsistency but a deliberate and risk-informed calibration of safeguards strategy. This variance is fundamentally driven by the distinct strategic role each facility plays in the nuclear fuel cycle and the resulting difference in how the measurement is used within the international safeguards regime [25].

At a conversion plant, the primary Safeguards objective is to accurately track the quantity of natural or slightly enriched uranium as it changes chemical form. The ±0.1% accuracy target for weighing UF_6 cylinders is precise and entirely fit for purpose within this context. It ensures a robust accounting of the total U mass entering or leaving the facility. This accuracy level is balanced against other operational factors and potential sources of uncertainty in the chemical process (rather than solely mechanical). The material, being in a less sensitive form (natural uranium when compared to the conversion from yellowcake to UF_6, has a lower proliferation significance), justifies a stringent but pragmatically achievable standard.

In contrast, the enrichment plant represents one of the most sensitive points in the entire fuel cycle. Here, the precise separation of isotopes creates material of direct strategic significance: low-enriched uranium for fuel and, as a tails stream, depleted uranium. The entire safeguards approach for such a facility is built upon a rigorous material balance, comparing the precise amount of the fissile isotope entering the feed stream against the amount exiting in the product and tails streams. The mass measurement is the cornerstone of this balance. Two critical factors necessitate the stricter ±0.05% accuracy requirement. First, the sheer volume of material processed means that even a tiny fractional uncertainty translates into a significant absolute amount of nuclear material. A 0.1% error on a 10-tonne cylinder represents 10 kg of UF_6, containing approximately 7 kg of U. Reducing this uncertainty by half significantly reduces the potential "noise" in the balance, thereby enhancing the ability to detect a meaningful loss [25].

Second, and most importantly, the mass measurement must compensate for the inherently higher uncertainty of the accompanying enrichment assay. Determining the U-235 isotopic fraction inside a full UF_6 cylinder non-destructively using gamma spectrometry is plagued by the physical effect of self-absorption, where the dense uranium itself absorbs its own radiation. Moreover, given the sublimation temperature of around 54°C, crystallisation or partial sublimation might appear, which impacts the gamma measurements' ability to describe the cask's interior beyond the infinite thickness. The accuracy is between 1% and 5%. To ensure the overall uncertainty in the final calculation of U-235 mass is not dominated by the mass measurement itself, the weight must be known with superior precision. The ultra-precise ±0.05% mass measurement is therefore a technical necessity to mitigate the limitations of the non-destructive isotopic analysis, ensuring the combined uncertainty of the material balance is tight enough to provide credible assurance that no diversion has occurred [25]. To overcome those limitations, current research initiatives have explored new hybrid methods combining laser, gamma ray and neutron techniques [35].

In essence, the higher accuracy standard reflects the heightened proliferation risk at the enrichment stage. The safeguards regime demands a more precise foundation for its material balance because the consequences of an undetected discrepancy are far greater, and the tools available to verify the isotopic content are less precise. The measurement accuracy is thus not a fixed value, but a variable strategically tuned to the risk profile of the facility. This instance is a great example of how Safeguards measures are tailored on a case-by-case basis, with the practical objectives in mind, a great deal is devoted to maintaining resource efficiency.

2.5.3.4. Reprocessing Plants

Reprocessing plants present criticalities due to the inaccessibility of process equipment, the high radiation fields, and the need to measure large flows of liquid and dissolved plutonium with extreme precision. The entire safeguards approach hinges on precise measurements at KMPs, primarily the input and output accountability tanks.

The first step is Tank Calibration. The physical volume of every process tank is meticulously calibrated, often using liquid addition methods, to establish a relationship between liquid level and volume. As measurements of mass can be much more accurate than measurements of volume or flow, a well-defined mass of water is used to assess the volume of the tank. Care is needed for the heels (it is the residual liquid in the bottom of the tank that cannot be completely drained or accurately gauged, even when the tank is considered empty). The sought goal is a volume measurement uncertainty of less than 0.1% [25]. Densitometers and Level Indicators are used in tandem to determine the mass of solution in a tank (when considering reprocessing liquids). Differential pressure manometers measure hydrostatic pressure to determine the liquid level, while gamma densitometers (using a radioactive source and a detector) measure the density of the solution. Combined, they allow the calculation of mass. The target accuracy for the total mass of solution in an input accountability tank must be at least 0.2% [25].

Table 9: Volume and Density ITV table [34]

Measurement	Instrument Type	Uncertainty Component % rel., k=1		ITV (% rel., k=1)	Notes
		u(r)	u(s)		
Volume[1/]	ELTM (Accountability tanks)	0.05	0.1	0.12	2/
	ELTM (Process tanks; high concentration)	0.2	0.2	0.28	3/
	ELTM (Process tanks; low concentration)	1	1	1.4	4/
	ELTM (Accountability tanks)	0.3	0.2	0.36	5/
	ELTM for calibration conditions (Accountability tanks)				7/
Density	ELTM (Accountability tanks)	0.05	0.05	0.07	6/
	ELTM (Process tanks; high concentration)	0.1	0.1	0.14	
	ELTM (Process tanks; low concentration)	0.7	0.7	1	
	VTDM	0.05	0.05	0.07	
	Density for calibration conditions				7/

1/ Volume determinations are made on the basis of level pressure, density and temperature measurements. The volume measurement uncertainties are highly dependent on the homogeneity of the liquid, the quality of the density measurements and of the calibration equation determined in the calibration process. The volume measurements may also involve an absolute error component which has to be taken into consideration when determining the overall uncertainty of volume measurements.

2/ For accountability tanks in newly built large-throughput facilities, uncertainties of 0.05% for u(r) and 0.1% for u(s) at full volume are achievable if: i.) A carefully designed calibration procedure has been implemented under well-controlled environmental and stable temperature conditions; and ii.) Measurements, using high precision electro-manometers, are performed on a well-characterized and homogenized liquid.

3/ Process tanks for high Pu concentration solutions are generally also equipped with high precision electro- manometers, however, the calibration effort and tank design specifications may be lower.

4/ Equipped with standard electro-manometers, lower calibration effort.

5/ The values apply to older facilities where the tank design was not driven by optimized ELTM volume measurement capabilities.

6/ The same comments as given for the volume measurements apply; one additional important calibration parameter is the determination of the probe (dip tube) separation.

7/ Placeholder for ITV still under review.

After determining the mass and volume of a liquid, the next critical step is to assess the Pu and U concentrations. This is achieved through DA. Representative samples are drawn from homogenised solutions and analysed via IDMS. This step requires an accuracy of better than 0.1% [25]. The combination of mass and concentration measurement enables the determination of total plutonium quantity with a target uncertainty of approximately 0.3-0.5% [25]. To further investigate the MUF, gamma scanners and neutron coincidence counters are used to quantify the amount of nuclear material trapped in pipes and vessels (holdup) and in solid waste streams, such as hulls and filters. These measurements are less accurate (often 5-20%) but are crucial for a complete material balance and characterisation of MUF for partial defect identification.

For reprocessing plants that take advantage of process monitoring, a value for false alarms is defined as 5% when a 95% detection probability is sought, as stated earlier.

2.5.3.5. Innovative Nuclear Reactors

For fast breeder reactor fuel cycle facilities, which handle large inventories of Pu and HEU in often-inaccessible forms (e.g., the NM is immersed in liquid sodium coolant), safeguards combine the most stringent elements of all other facilities. The fresh fuel (which can vary a lot from traditional fuel vectors) is verified at fabrication plants using neutron coincidence counters (1-3% accuracy for Pu mass) and gamma spectrometers. **At the reactor itself, in-situ measurement is often impossible, so reliance shifts to C/S**. Traditional seals, surveillance cameras, and fibre-optic seal systems that can assure the CoK. For the SNF of fast reactors, specialised techniques are being developed to verify its identity and quantity through the shielding and cooling systems, though with significantly higher uncertainty than in accessible environments. **Newly developed reactors could have a mixed approach**.

2.5.3.6. Summary on the Application of Safeguards to the Fuel Cycle

In conclusion, the application of IAEA safeguards employs a vast array of instruments, ranging from scales and cameras to mass spectrometers and neutron counters, each chosen and calibrated to achieve the specific accuracy required to provide credible assurance that the NM remains in peaceful use at every step of its journey through the fuel cycle. A further assessment of the preferred assay methodology, NDA, is done below.

2.6. NDA

NDA refers to measurement techniques "applied to nuclear material and other items of safeguards interest to confirm their isotopic composition and quantity without destroying the items" [27]

Six fundamental types of NDA techniques for NMAC and NMA verification can be listed: weight, volumetric, gamma, neutron, calorimetric, and modelling [27].

Commonly, NDA refers to the methods based on radiation-matter interactions, which often take advantage of the NM signatures[3], which are the features that make it identifiable and assessable. In a broad sense NDAs can be subdivided into active NDA

[3] Signature = The intrinsic material characteristics that contribute to the ease of identifying/recognising the nuclear material type and composition [27].

or passive NDA. A special mode of passive assay, self-interrogation, has been recently developed.

In the choice of the appropriate NDA technique and instrumentation, the following considerations are to be made, according to [27]:

- *"What radionuclides could be present?*
  What is the matrix of the material, and is it uniform?
- *What is the geometry of the items to be measured?*
- *What precision and accuracy are required?*
- *What background conditions exist in the measurement area?*
- *Do building or facility constraints exist? How will the measurement system be implemented?*
- *What measurement controls will be necessary?*
- *What is the throughput expectation?*
- *What is the budget?"*

[36] defines passive NDA as "*the properties of the item may be assessed by detection and characterisation of its spontaneous radiation emission (i.e., neutrons). The intensity of this radiation is nearly exclusively a function of the item's isotopic composition and cannot be controlled by a fuel cycle facility/reactor operator, except for the multiplication of the item*". Passive NDA should be preferred as it is generally easier to implement in the field, cheaper, and faster than DA and active NDA [5].

2.6.1. The Relevant NDA modes

2.6.1.1. Modes of Neutron Assay

2.6.1.1.1. Passive Neutron Assay

Radioactive isotopes (generally the pair isotopes of Pu) undergo **spontaneous fission (SF) events, which emit a sequence of generally 2 or 3 neutrons** (according to a probability distribution) that are correlated in time with one another, allowing an observer to identify a SF event. **The quantification of these events allows the direct quantification of the pair isotopes of Pu. This information, combined with the isotopic ratios of the NM, can inform on the total actinide mass.**

However, many relevant thorium, uranium, and plutonium isotopes decay via α-decay. The range of α-particles is too short (~µm; stopped by a sheet of paper) to be of direct relevance when the material is bulk in nature because they cannot escape the measured item and be externally detected. If the actinide atom is chemically bound to, or mixed in a matrix with, material containing a low atomic number element (e.g., Li, Be, O), the emitted α-particle may be absorbed by the atom of the light element, leading to the emission of a single neutron via an (α,n) reaction. These (α,n) neutrons cannot be discriminated from the neutrons from spontaneous fissions using their respective energy. **(α,n) Neutrons are not correlated in time with one another**; therefore, it is possible to identify them and quantify their relative contribution to the overall detected neutron rates by coincidence counting, by the application of appropriate statistical methods. Finally, by subtracting the neutrons generated by these events from the overall count, the neutron signature can be directly related to the even isotopes of the NM.

The passive detection of the neutrons must differentiate between these events to obtain the desired goal of fissile mass quantification.

2.6.1.1.2. Active Neutron Assay

In an active NDA, the properties of the item are inferred from its induced response to a controlled action of the operator/inspector, which is referred to as interrogation. The interrogation induces the item's response, by irradiation with a known particles quantity, to produce a measurable signal. When considering the neutron signatures of NM, the irradiation is typically performed by **an external neutron source that may induce fission within the material, ultimately allowing the assessment of the effective amount of all fissile nuclides present.**

The selection of an appropriate neutron source for nuclear material assay is determined by the required neutron energy spectrum and operational parameters. Neutron generators, utilising deuterium-deuterium or deuterium-tritium reactions, produce monoenergetic neutrons at 2.45 MeV and 14.1 MeV respectively, with commercial systems achieving neutron intensities up to approximately 1×10^9 n/s. These systems offer pulsed or continuous operation modes and can be deactivated when not in use, though they require complex instrumentation and regular target replacement.

Alpha-emitters such as ^{241}Am or ^{238}Pu can be used as neutrons sources via (α,n) reactions, producing a continuous neutron spectrum with energies typically below 5 MeV. These sources require no external power but cannot be switched off, necessitating permanent shielding.

Spontaneous fission sources, particularly californium-252 (^{252}Cf), emit neutrons following a fission spectrum with a mean energy of approximately 2 MeV and a high-energy tail extending to about 10 MeV. These sources produce time-correlated neutrons but have a relatively short half-life of approximately 2.6 years, necessitating regular replacement to maintain consistent intensity, which the inspector must accommodate. Both isotopic sources provide continuous emission without external power but present persistent radiation fields that complicate measurement interpretation and safety protocols [36].

An active NDA method currently used in waste characterization is the differential die-away technique (DDA). Differential die-away is a pulsed neutron interrogation technique in which short bursts of fast neutrons, typically from a deuterium–tritium or deuterium–deuterium neutron generator, are injected into a sample to induce fission in fissile material while simultaneously generating capture and scattering events in surrounding moderating media (polyethylene). The measured time-dependent neutron population in the detector following each pulse exhibits a characteristic decay profile made by both prompt fission neutrons and longer-lived thermal or epithermal neutron populations. By analysing how the number of detected neutrons decreases with time after the pulse (the neutron population "dies away"), the method enables the differentiation between prompt neutrons, SF-emitted neutrons, and delayed neutrons from induced fissions. The difference between these decay rates is highly sensitive to the presence and quantity of fissile isotopes, thus allowing its quantification (not characterisation).

2.6.1.1.3. Self-interrogative NDA

In [36], self-interrogation is described as "*a special mode of NDA that combines certain aspects of both passive and active assay modes. From the perspective of the operator/inspector, self-interrogation is a passive mode of assay because it does not require the introduction of external radiation to induce a response from the item. From the perspective of the item, it is similar to an active assay in the sense that a response is being induced within the item*". In any case, the efficacy of self-interrogation depends on the geometrical configuration of the item and its surroundings according to the two mechanisms employed.

1. Neutrons emitted by the item are reflected by the item's surroundings back into the item to interrogate it from the outside-in. Neutrons emitted by the item can be reflected in a sufficient number only if their energies are moderated to or near thermal levels [27]

2. Neutrons emitted within the item interrogate the item from the inside-out. This case can be considered only if the isotope of interest is in chemical form with other light elements, such as Oxygen or Fluorine, or by the presence of Lithium [36] [30] [27].

2.6.1.2. Methods based on Gamma-rays detection

Regardless of the decay mode of an unstable atomic nucleus, in nearly all cases, the emission of an α- or β-particle is accompanied by the emission of one or more gamma rays. As a matter of fact, the daughter nucleus is most often formed in an excited state and, to reach its ground state, one or more gamma rays are emitted in short succession. A gamma ray can have discrete and unique energies for a specific isotope, since they are tied to the structure of the nucleus and its possible transitions. The resulting energy spectrum of emitted gamma rays constitutes a fingerprint that is unique to each radioactive isotope, making it detectable. However, not all isotopes produce easily measurable gamma rays, and in some instances, only gamma spectra from decay of secondary or tertiary products can provide evidence of a certain isotope's presence within the assayed item.

Gamma-based NDA instruments and methods provide qualitative information that is complementary to the quantitative information provided by neutron-based instruments and methods [36].

2.6.1.3. Calorimetry

The decay heat is an integral measure of the heat produced by a fuel element. It is calculated by combining reactor simulations and irradiation history to estimate the fuel inventory (the inputs of the simulation are decay properties of fission products and actinides, cross-sections and fission yields) [37].

2.6.2. Assay Instrumentation

A few of the main methodologies with expected applicability to innovative NESs are described.

2.6.2.1. Detector's choice

The resolution of a detector is a measure of its ability to resolve two peaks that are close together in energy. The parameter used to specify the detector resolution is the full width of the (full-energy) photopeak at half its maximum height (FWHM). If a standard Gaussian shape (where σ is the width parameter for the Gaussian) is assumed for the photopeak, the FWHM is given by:

$$FWHM = 2\sigma\sqrt{\left(2ln(2)\right)} \qquad 6)\ [27]$$

High resolution (small FWHM) makes the distinction of close-lying peaks easier.

The absolute detector's efficiency is defined as below:

$$\varepsilon_{tot} = \frac{\text{total number of photons detected in the full energy peak}}{\text{total number of photons emitted by the source}} \qquad 7)\ [27]$$

A more detailed definition discriminates between the sources of inefficiencies:

$$\varepsilon_{tot} = \varepsilon_{geom} \times \varepsilon_{absp} \times \varepsilon_{sample} \times \varepsilon_{int} \qquad 8)\ [27]$$

Where:

- The geometric efficiency ε_{geom} is the fraction of emitted photons intercepted by the detector.

- The absorption-efficiency ε_{absp} is the fraction of emitted photons not absorbed in intervening materials. Of particular relevance in shielded samples.
- The sample efficiency ε_{sample} is the fraction of emitted photons that escape the sample without being absorbed.
- The intrinsic efficiency ε_{int} is the probability that a gamma ray that enters the detector will interact and give a pulse in the full-energy peak. In simplest terms, this efficiency comes from the standard absorption formula [27]

The complexity of a gamma-ray spectrum has a significant impact on the selection of the optimal detector; as the spectrum becomes more intricate, higher energy resolution is necessary to resolve closely spaced peaks. High-resolution detectors such as HPGe are used for the assay of Pu, which present particularly complex spectrum structures, due to the more varied isotopic vector. U has a simpler spectrum that can often be adequately assayed using lower-resolution detectors. Efficiency is a critical factor, as it determines the achievable full energy peak count rates and directly affects both the measurement duration required for a set precision and the minimum detectable activity. Higher efficiency is linked with bigger crystal dimensions, thus greater expense, particularly for HPGe. Conversely, thanks to the higher Z of I, NaI detectors can reach acceptable efficiency at a lower cost, although the working principle of such detectors reduces their resolution. Such detectors are therefore used when their resolution is fit-for-purpose while maintaining lower costs.

For a same detector type, a trade-off between efficiency and resolution must still be made: a more efficient (bigger) crystal has worse charge collection, implying a lower resolution. The detector has to be once again fit for purpose. For Pu, planar HPGe (low efficiency, high resolution) is used, and for U coaxial HPGe (lower resolution, but higher efficiency) is used instead.

Considerations such as portability, cooling, and available space further complicate detector choice and often require compromise among conflicting priorities. Therefore, once the most suitable detector type has been selected, it is imperative to communicate all essential technical requirements to suppliers in detail to guarantee that the delivered instrument meets the intended application.

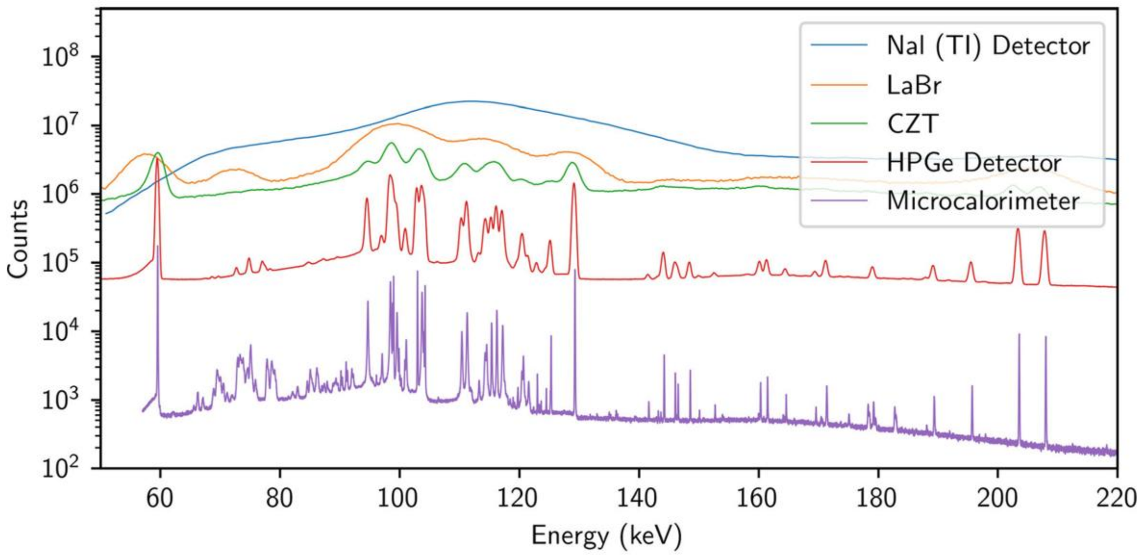


Figure 9: Low-burnup plutonium spectrum obtained with several different detectors. [27]

2.6.2.2. Neutron Detectors

The neutron detection technologies traditionally used in safeguards applications include primarily ^{3}He-based proportional counters, which have been a gold standard of nuclear safeguards for the past decades, and can detect neutrons by producing an electrical pulse for each neutron that interacts with the gas. Fission chambers are based on the fission of U-235 from incident neutrons.

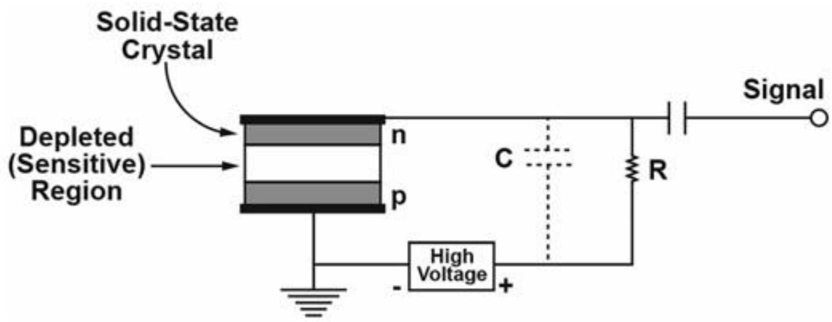
Neutron proportional counters coupled with pulse train analyses constitute a neutron coincidence counter. The system measures not only the total neutron count rate but also detects nearly simultaneous pairs ("doubles") and triples of neutrons, both emitted by SF events. This ability to identify time-correlated neutrons (after data analysis) allows us to distinguish neutrons based on their generation process. As described above, by analysing the ratio and timing of these events, the neutron coincidence counter can accurately quantify and help identify nuclear materials. ^{3}He-based counters provide high neutron detection efficiency to support quantitative nuclear material assay.

Fission chambers are typically used in applications with a high gamma background. They represent the typical technology used in SNF applications, and can operate at dose rates of 10^{4}-10^{6} Gy. The detection efficiency is low, in the order of 0.01% against ~0.3% for ^{3}He, which is adapted to the high-dose environment. However, the fission chambers present shipping and regulatory control issues because of the fissile material content [30].

The miniHDND [30] is a promising technology being developed in the USA (along with the HDND [38]). The key advantage of this application is its improved gamma dose tolerance compared to ^{3}He. As mentioned, it is based on boron-lined proportional counters (a detection cell). Thermal neutron detection in a boron-lined proportional counter is based on the ^{10}B (n,α) ^{7}Li reaction, which yields higher energy reaction products (2.3 MeV reaction energy) than in of ^{3}He (765 keV reaction energy). The pulse height spectrum (measured signal) from a boron-lined detector, therefore, results in an improved energy discrimination to prevent gamma-ray pileup in high dose environments and allows for the attainment of a usable neutron signal, all while having 10x the efficiency of fission chambers [38].

2.6.2.3. HPGe detectors

High-purity Germanium detectors (HPGe detectors) are the current best solution for precise gamma and X-ray spectroscopy. HPGe detectors operate as semiconductor diodes with a p-i-n structure, where the germanium crystal serves as the sensitive volume to ionising radiation. When gamma photons enter the crystal, they interact through photoelectric absorption, Compton scattering, or pair production. Compared to silicon detectors, Ge has a better radiation detection capability due to its atomic number being higher than Silicon's and a lower average energy required to create an electron-hole pair, which is 3.6 eV for silicon and 2.9 eV for germanium. Due to its higher atomic number, Ge has a much larger linear attenuation coefficient, which leads to a shorter mean free path, hence higher detection efficiency, thus resolution. Moreover, silicon detectors cannot be thicker than a few millimetres. Ge can have a sensitive thickness of centimetres and, therefore, these two factors combined result in a total absorption for gamma rays up to a few MeV. Under the influence of an electric field, electrons and holes travel to the electrodes, generating charge pulses whose amplitude is a direct measure of the incoming photon energy, thus enabling high-resolution gamma spectroscopy. The number of such pulses per unit time also gives information about the intensity of the radiation.



Typical arrangement of components in a solid-state detector. The crystal is a reverse-biased p-n junction that conducts charge when ionization is produced in the sensitive region. The signal is usually fed to a charge-sensitive preamplifier for conversion to a voltage pulse

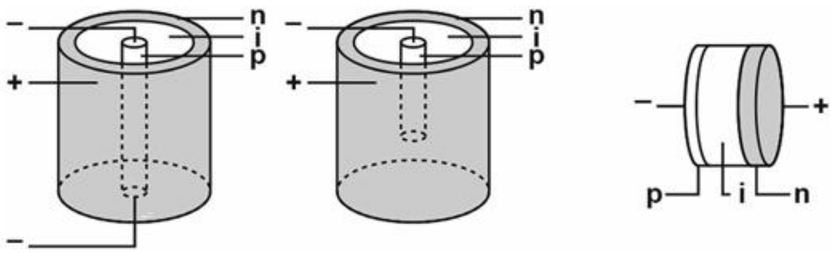


Illustration of various solid-state detector crystal configurations: (**left**) open-ended cylindrical or true coaxial, (**middle**) closed- ended cylindrical, and (**right**) planar. The p-type and n-type semi-conductor materials are labeled accordingly. The regions labeled i are the depleted regions that serve as the detector sensitive volumes. In the context of semiconductor diode junctions, this region is referred to as the intrinsic region or a p-i-n junction

Figure 10: Solid state detector components and configurations [27]

The detector requires cooling to cryogenic temperatures, typically around 77 K using liquid nitrogen, to reduce thermal noise caused by germanium's low band gap (0.67 eV) and to maintain the purity and charge collection efficiency of the crystal.

2.6.2.4. hybrid K-edge densitometry

Hybrid K-edge densitometry (HKED) is a technique for the quantification of actinide elemental concentrations in NM solutions, combining K-edge densitometry (KED) with X-ray fluorescence (XRF) spectroscopy. HKED has become the workhorse NDA technique for nuclear safeguards applications at reprocessing and fuel fabrication facilities worldwide.

KED relies on the characteristic drop in X-ray transmission through a sample due to X-rays' resonant absorption at the K-shell electron binding energy. X-rays with energies above the K-shell energy are capable of being absorbed by the atom and thus liberating the K-shell electron by photoelectric effect. At the K-edge energy (115.606 keV for uranium and 121.797 keV for plutonium), the X-ray absorption coefficient exhibits a sharp discontinuity, and the magnitude of this step change is directly proportional to the elemental concentration. KED characterises the total (elemental) U density of a sample. KED is currently employed for measuring aqueous U solutions from SNF reprocessing, where typical concentrations are 150–300 g/L. In general, an accuracy of less than 1% in the U density (generally around 0.2%–0.3%) is achievable for U solutions of 100–500 g/L [39].

The combination of the two measurement techniques into a single measurement is done by employing two HPGe detectors with an X-ray generator, simultaneously measuring both x-ray transmission through a sample (as described for KED) as well as the relative abundances of actinides via XRF. Taking the absolute density of U in the analyte from KED, the densities of other actinide species can be inferred based on their abundances relative to U.

HKED allows for rapid, high-precision measurements at reduced operational costs compared to traditional wet chemistry. Measurements can be completed in timeframes ranging from 300 to 3000 seconds, providing near-real-time accountability data [40] [41] [42].

2.6.2.5. Microcalorimetry

Microcalorimeter (microcal) detectors are sensitive devices that can produce extremely high-resolution spectra by exploiting material properties at very low temperatures (< 0.1 Kelvin). A microcalorimeter is a device that consists of essentially two components: an absorber and a sensor. The absorber is engineered such that a single photon of the energy range of interest that strikes it will increase its temperature by a measurable amount. The sensor, thermally coupled to the absorber, is designed to be sensitive to the temperature increase of the absorber and convert the heat pulse into an electrical signal. The more energy the photon deposits in the absorber, the larger its temperature increase and the larger the electrical pulse produced in the sensor. A gamma-ray spectrum from a microcal detector can readily have a resolution an order of magnitude better than a HPGe detector. The technology has become quite mature and has been identified as an important technique when exceptionally precise gamma-ray measurements are required [27]. However, costs and implementation issues (related to the extremely low operating temperatures) must be justified.

3. Molten Salt Reactor Technology, Applications, and Safeguards

Molten Salt Reactors (MSRs) are a class of nuclear fission reactors where molten salts serve as the reactor coolant, but also possibly as fuel, and/or moderator. This fundamental design variation from water-cooled reactors has a variety of technical and non-technical implications that are making MSRs relevant for near-future development.

3.1. History and Context

3.1.1. History of MSRs' Development

The development of the Molten Salt Reactor (MSR) concept was driven by the demand for a compact, high-power-density engine for military propulsion, specifically for long-range strategic aircraft. The initial push for the technology came from the U.S. Air Force's Aircraft Nuclear Propulsion program, which sought a reactor capable of operating at extremely high temperatures to power a jet engine, thereby providing the necessary thrust for a bomber with unlimited range. This military imperative provided the direct funding and urgency that led to the design and construction of the Aircraft Reactor Experiment (ARE). The ARE, conducted at Oak Ridge National Laboratory (ORNL) in 1954, was a pioneering proof-of-concept that successfully demonstrated the fundamental viability of a high-temperature molten salt reactor but also revealed significant materials challenges. Fast reactors using chloride salt were also considered in the early stages, but the relatively high neutron capture cross-section of chlorine-35 led to the focus on fluoride salts. Its primary finding was the achievement of criticality and operation at an unprecedented power density and outlet temperature of over 800°C, proving that a molten fluoride salt mixture could serve as both an efficient fuel carrier and an effective coolant. The experiment provided crucial validation for the reactor's unique safety characteristics, including a strong negative coefficient of reactivity, which was demonstrated when the reactor smoothly shut down in response to a controlled cooling of the salt. However, the ARE also exposed the critical issue of severe corrosion and radiation damage to its metal components. The stainless-steel alloy used could not withstand the corrosive, high-radiation environment, suffering from intergranular cracking. This key finding underscored the absolute necessity of developing entirely new, corrosion-resistant structural materials, a challenge that directly spurred the creation of the nickel-based Hastelloy-N alloy. Therefore, while the ARE conclusively proved the neutronic and thermodynamic soundness of the MSR concept for high-performance applications, its most enduring legacy was in starkly identifying the paramount materials science obstacle that would define much of the subsequent MSR research and development. Following the cancellation of the aircraft program, the scientific success of the ARE convinced researchers at ORNL that the molten salt concept held even greater promise for civilian nuclear power, leading to the development of the Molten Salt Reactor Experiment (MSRE) to explore its potential as a thermal breeder reactor. MSRE successfully demonstrated the stable, continuous fission reaction in a fluoride salt fuel (a LiF-BeF2-ZrF4-UF4 mixture) at high temperatures (650°C) for extended periods, proving the core nuclear viability. The MSRE also pioneered the handling of molten salts, demonstrating the effectiveness of specialised nickel-based alloys, notably Hastelloy-N, which was developed to resist corrosion in the molten fluoride salt environment. Furthermore, it successfully tested online fuel processing by circulating the salt fuel and continuously removing gaseous fission products, a unique feature of the MSR concept. However, the experiments also revealed paramount issues. A key finding was the unexpectedly high radiation-induced embrittlement of Hastelloy-N caused by the high-energy neutron flux. Shielding and improved alloy design were pursued to alleviate the embrittlement vulnerability of containment. The shielding approach added an interior graphite lining of approximately half a metre to the interior of the reactor vessel to prevent a significant neutron flux from reaching the vessel wall. The graphite itself was subject to radiation damage and dimensional change. The improved alloy design approach was based on creating large numbers of helium traps (finely dispersed carbides) within the microstructure of Hastelloy N [37]. Another significant issue was the need to separate protactinium and uranium from the molten fuel and finally prevent tritium permeability into components and eventually the steam cycle, which could pose a hazard. Both problems were solved during the MSRE and led to the development of separation technology and tritium capture technology still needed in today’s designs. Thus, while the U.S. program conclusively proved the neutronic and basic chemical feasibility of the MSR concept, it also left a legacy of specific materials science and chemistry challenges that must be definitively solved for commercial deployment.

3.1.2. Today’s Context

Today's Context is one of an upcoming energy crisis. On one hand, the worldwide energy production still heavily relies on fossil fuels, which are notoriously a finite source of energy and expected to run dry sooner than later, posing an inevitable struggle to find alternative energy sources. The current commercial nuclear fleet accounts for only 4% of the world primary energy consumption [43]. On the other hand, carbon emissions from energy production (beyond electricity) are severely fuelling climate change. Approximately 40% of global CO2 emissions are emitted from electricity generation through the combustion of fossil fuels to power a steam turbine [44].

For the sustainable continuation or development of civilisations, it is a worldwide priority to find alternative energy sources to fossil fuels. The green energy takes advantage of renewable energy sources like solar or wind, but each presents challenges on its own. Both renewable energy production facilities have significant life-cycle emissions, but the main problem remains the intermittence of the energy source. This intermittence is particularly problematic for electricity production, by destabilising the grid and unbalancing the energy markets. There is still a need for a safe, secure, and reliable means for producing the base load of electricity consumption. Nuclear Energy presents itself as a comprehensive solution, since it constitutes a very stable means of producing electricity in the long term, with very low carbon emissions.

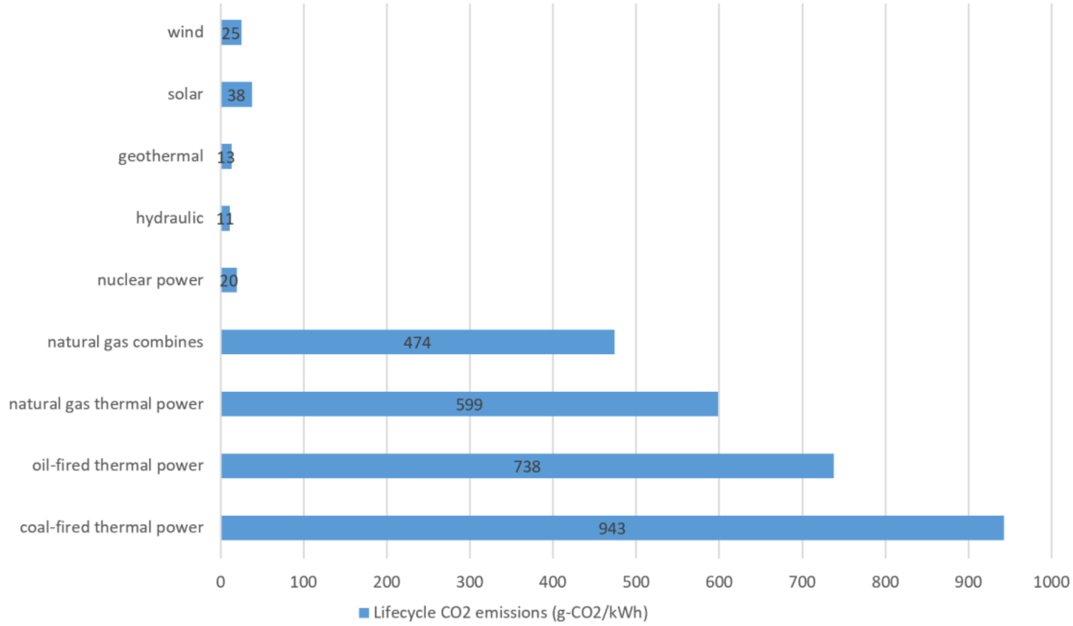


Figure 11: CO2 emissions of several power sources per kWh produced. Adapted from [44]

The widespread adoption of nuclear power has been hindered by challenges related to cost, public perception of safety, long construction timelines, and the management of radioactive waste.

However, the global energy crisis extends far beyond the electricity sector, encompassing a critical shortfall in the provision of industrial heat, transportation fuels, and feedstocks for chemicals and fertilisers, all of which remain heavily dependent on fossil fuels. This multifaceted crisis, driven by geopolitical volatility, resource scarcity, and the climate imperative, is amplified by a soaring global demand for these non-electric energy forms. Consequently, there is an urgent need for new, more efficient technologies that can decarbonise these harder-to-abate sectors. Advanced nuclear energy, particularly Generation IV designs, offers a transformative solution by generating high-temperature process heat for industries such as steel and cement manufacturing, enabling thermal storage for grid stability, and supporting the production of hydrogen and synthetic carbon-neutral fuels for transportation and agriculture. These technologies offer a pathway to replace hydrocarbon-based systems with inherently safer, more efficient, and dense sources of clean energy, thereby enhancing energy security and industrial competitiveness while meeting ambitious climate goals. The development of such innovative nuclear applications is therefore not merely a matter of electrical generation but a fundamental prerequisite for a comprehensive and sustainable energy transition across the entire global economy.

These societal factors, the apparent advantages of the applications of MSR technology, combined with the proof-of-concept from the 20th-century experiments, constitute a fertile ground for MSR to make a comeback.

3.1.3. Advantages of MSR Technology

A very different set of features characterises the distinction between this class of nuclear reactors and traditional water-cooled reactors.

3.1.3.1. Technical Advantages

A key safety feature common to all MSR technologies is the operation of the primary system at near atmospheric pressure, which drastically reduces mechanical stress on structural materials, lowers the risk and potential consequences of accidents, and minimises the possibility of radioactive releases. Essentially, the loss of coolant accident risk is minimised. The chemical and thermophysical properties of the molten salt itself provide intrinsic safety; it acts as an exceptional solvent that strongly bonds and retains fission products, creating a powerful barrier against radioisotope release. Furthermore, the salts are chemically inert and non-flammable, eliminating risks associated with exothermic reactions, causing hydrogen explosions or sodium fires. MSRs, if liquid-fuelled, enable the continuous removal of gaseous fission products during operation, preventing pressure buildup within the core. The technology boasts a strong negative reactivity temperature feedback mechanism, providing intrinsic core stability and enabling passive shutdown in response to temperature rises. Because if the fuel is liquid, it expands when heated, thus slowing down the rate of nuclear reactions and making the reactor self-regulating [45]. Additionally, this aspect is enhanced by the fact that the heat is produced directly in the heat transfer fluid; thus, there is no heat transfer delay and very fast thermal feedback. Moreover, the fuel is homogeneous, which prevents the need for a loading plan. Finally, molten salt is an ionic liquid, and irradiation damage or mechanical failure of the fuel does not occur, unlike in solid fuel pellets, in which it causes a limitation of the irradiation potential. This aspect is coupled with better neutron economy due to the absence of parasitic neutron-absorbing structural materials, which is further enhanced by the possibility of employing semi-continuous fission product removal. Thus, liquid-fuelled MSRs are not constrained by fuel performance limitations (or by a fixed amount of fissile material that can be loaded at a given interval), and MSR fuels can be irradiated to much higher burnups: 60 GWd/tHM (Gigawatt-days per ton of Heavy Metal) is typical for PWRs compared with anything up to 200–300 GWd/tHM for fast-spectrum MSRs. [46]. A reactor core with liquid fuel can host all actinide elements and enable their continuous recycling. Since the fuel manufacturing stage can be omitted, even elements producing intensive decay heat can be recycled with the liquid fuel. This facilitates flexible fuel cycles, including the ability to breed new fuel from thorium-232 or uranium-238 or to burn existing transuranic waste stockpiles. The liquid fuel form allows for continuous in-situ recycling of all actinides, omitting the complex and costly need for solid fuel fabrication [47].

The molten salt, often combined with a large graphite moderator mass, serves as a massive heat sink to absorb decay heat after shutdown. For most MSR designs, the liquid form of the fuel provides a simpler way to remove heat by circulating the fuel in a heat exchanger. This, as opposed to water-cooled reactors, eliminates the risk of circulating coolant fluid in the core. The exceptionally high boiling point of the salts (with a margin of up to 700°C above operating temperatures) ensures that cooling capability is maintained even during severe temperature transients.

3.1.3.2. Economic Advantages

MSRs offer compelling economic advantages that are intrinsically linked to their enhanced safety profile, as the reduced need for complex engineered safety systems lowers capital costs. A primary economic driver is the ability to operate at high temperatures exceeding 600°C, which enables very high thermodynamic efficiency of up to 50%. This efficiency can be harnessed through conventional steam Rankine cycles or more advanced gas Brayton technology, such as upcoming supercritical CO2 cycles that promise dramatically smaller machinery, allowing for miniaturisation and versatility. Furthermore, this high-temperature output opens the vast non-electrical energy needs that have been previously mentioned, which are inaccessible to water-cooled reactors limited to around 300°C. MSRs also exhibit highly effective load-following capabilities, making them ideal for modern grids and coupling with intermittent renewable sources. This is again due to the rapid, negative reactivity feedback inherent in liquid fuel systems, which allows the reactor to automatically and swiftly adjust its power output to match grid demand without relying on control rods. Indeed, as soon as the fuel salt temperature varies because the power extracted has changed, the quasi-instantaneous variation of the salt density modifies the power generated [37]. The economics of the fuel cycle are also superior. MSRs achieve high resource utilisation through the high burnup and a closed fuel cycle that burns minor actinides and existing transuranic waste, minimising final waste volumes. As previously mentioned, the fuel fabrication cost is also reduced. Finally, the optimal use of the core volume in liquid-fuelled designs and the potential for modular construction facilitate a compact form factor, supporting decentralised power production and offering a flexible, attractive technology for future energy systems.

3.2. The MSRs' Underlying Technologies and the Reactor Families

To understand the MSRs at the nuclear level, a brief description of the neutron nuclear interactions is needed. The key nuclear reactions are neutron capture (n, γ), neutron scattering (n, n), and neutron-induced fission (n, f). The resulting reaction is determined by the energy of the incident neutron and the inherent nuclear properties of the specific isotope, with the probability of each reaction occurring being quantified by its neutron cross-section, measured in barns (1b=10^{-24} cm^2). The cross-section of an event is the fraction of the incident particles which undergo the specified interaction, divided by the number of target particles per unit area (of a thin target) [48].

Neutron capture occurs when the nucleus absorbs a neutron without fissioning, transmuting to a heavier isotope of the same element, accompanied by a gamma-ray emission to reach a ground state of excitement. A neutron scattering reaction occurs when a target nucleus emits a single neutron after a neutron-nucleus interaction. This can happen by the collision of the neutron into the nucleus or a close interaction in the nuclear forces. In fission, the nucleus absorbs a neutron, which causes an unbalance in the fundamental nuclear forces, and splits the nucleus into two lighter FPs, with emission of neutrons and/or alpha, and gamma.

At the macroscopic level, one of the key characteristics of MSRs is the significant variation between designs. A classification becomes necessary, especially when considering that reactors of the same family could have a similar safeguards approach. Within the ENDURANCE project, all reactors share a common characteristic, which is the use of liquid fuel salts. Their other characteristics vary greatly. **In the optic of finding a common safeguards approach for reactors with technical similarities, the question of how to group together these reactors comes forward.**

Several classifications for MSR families are possible based on the technology involved.

Some of the parameters that have been considered for MSR classification fuel state (liquid, solid); neutron spectrum (thermal, epithermal, fast, hybrid, variable); moderator (graphite, hydrogen based, deuterium based, beryllium based, other); salt type (fluorides, chlorides, mixture); salt purpose (fissile, fertile, coolant, moderator); core criticality (critical, subcritical); primary circuit layout (loop, pool); core structure (homogeneous, heterogeneous); use of neutron leakage (reflector, blanket, multizone core); neutronics design (burner, converter, breeder, breed-and-burn); make-up fuel (enriched uranium, existing spent fuel, none); fuel cycle (open, partially closed, closed); breeding (Th–U cycle, U–Pu cycle, combination); fuel reprocessing location (in situ, ex situ); and fuel refilling/removing (continuously, batchwise, at once, none) [37].

Two approaches are proposed in [37] and [49]. The former focuses on the technologies employed *stricto sensu*. The latter regroups MSR designs based on the PR-relevant macro-characteristics.

3.2.1. ENDURANCE

ENDURANCE ("EU kNowleDge hUb foR enAbling MolteN Salt ReaCtor safety development and dEployment") is a project led by Politecnico di Milano, coordinated by the Department of Energy, and funded by Euratom. The project is a cooperation between the European Commission's Joint Research Centre (EC JRC) and several other academic and industrial partners. The goal of the project is to help the (European) designers of MSR to increase the TRL of their designs through cooperation with the different stakeholders. ENDURANCE designs are exclusively liquid-fuelled, and thus this thesis will focus on liquid-fuelled designs. Given that pebble-bed reactors are similar to HTGRs, their safeguardability and safeguards approach are similar and should be studied jointly.

This thesis evolves in the Phase 1 (M1-12) of the ENDURANCE project, task 5.5 on waste treatment and processes safeguardability for which the goals are to:

- Compile a comprehensive list of publications, reports, and guidelines from reputable sources such as the IAEA, GIF, and nuclear research institutions
- Analyse the general considerations of proliferation resistance and safeguardability in relation to reactor design and operation, and identify those that might be of interest for MSRs
- Identify specific features of MSRs that may influence proliferation resistance and safeguardability, considering different fuel forms and design options
- Gather a list of required data as per the IAEA's safeguards Design Information Questionnaire (DIQ)

A table summarising the main characteristics of the Endurance's designs is presented abovebelow.

A more thorough description of the designs is done in [50], and only briefly reported in this work in **Erro! Fonte de referência não encontrada.**.

Table 10: List of EU MSRs identified [50]

Concept	Fuel carrier	Moderator	Neutron spectrum	Fissile matter / Fuel cycle	Power [MW_{th}]	Use
ARAMIS-A (CEA + ISAC project)	Chloride	None	Fast	Pu + Am	300	Electricity production + Waste reduction
ARAMIS-P (CEA / Orano)	Chloride	None	Fast	Pu	300	Electricity production + Waste reduction
CMSR (Seaborg)	Fluoride	Graphite	Thermal	U-based	250	Electricity production
Copenhagen Atomics MSR	Fluoride	Heavy water	Thermal	Th-based	100	Heat production + Waste reduction
MSFR-F (CNRS + EU projects)	Fluoride	None	Fast	^{233}U – $^{enrich}U$ – Pu + MA / Th fuel cycle	3000	Electricity production
MSFR-Cl (CNRS + SAMOSAFER)	Chloride	None	Fast	Pu + MA / U fuel cycle	3000	Electricity production + Waste reduction
RAPTOr (CNRS/Orano)	Chloride	None	Fast	Pu + MA	300 / 500	Waste reduction + Electricity production
Stellarium (Stellaria)	Chloride	None	Fast	U-based or Pu-based	250	Energy + Waste reduction
Thorizon One (Thorizon)	Chloride	None	Fast	Th and Pu or U based	250	Electricity and possibly heat production + Waste reduction
XAMR (Naarea)	Chloride	None	Fast	Pu-based	80	Electricity and/or heat production + Waste reduction

3.2.2. The IAEA's taxonomy

This taxonomy is based on the fact that graphite is the only known moderator that allows direct contact with the fluoride salts and thus constitutes a discriminatory factor.

Class I

The choice of graphite as a moderator has several design implications and can constitute an effective means of grouping the designs into a family, Class I.

- In power reactors, graphite usually has a higher temperature (700°C) than moderators based on ^{1}H or ^{2}H (water or heavy water). Hence, the neutron energy spectrum peak is shifted towards higher neutron energies.
- To slow down a neutron from 2 MeV to 1 eV with a graphite moderator, 92 collisions are necessary, which is significantly more than with ^{1}H or ^{2}H, and at the same time a 10 times smaller scattering cross-section. However, its use in the reactor core is justified by the 200 times smaller capture cross-section than water, (see the neutron capture probability in Figure 12B) which improves the

neutron economy and counterbalances the previous effect. To achieve breeding, thermal neutrons are preferrable (not epithermal that result in a resonance zone), and features such as a blanket, reflector, layout with two zones or a bulky core, or a combination of these features, need to be used to improve the interaction probabilities with fertile nuclei. Graphite-moderated MSRs always act undermoderated [37].
- The initial fissile load is lower for graphite-moderated MSRs. At the same time, a reactor with very low fissile content in the core is more sensitive to the parasitic absorption of neutrons by fission products.
- In reactors cooled by fluoride salt, graphite acts as a fuel matrix. The graphite's useful life is severely impacted by the radioactivity (as described in the MSRE) and is replaced regularly, creating a volume of activated waste. [37]

When referring to solid fuel, it usually is Tristructural-Isotropic (TRISO) particles that are small spheres of graphite composites containing the fissile isotopes. As fuel is segregated within pebbles, online isotopic separation is not possible as in liquid fuel, but differently from the fuel salt, pebbles can be discerned and discarded according to burnup level. This constitutes an advantage to traditional fuel pellets, in addition to the increased solidity of the pebble (in comparison to pellets). More discussion is done in 3.2.3.

Class II

Class II is Homogeneous MSRs. These reactor designs are characterised by the absence of structural materials in the core that separate the fuel salt from a coolant or a moderator. The core is thus filled solely by the fuel salt. Homogeneous MSRs adopt a fast neutron spectrum (no moderators). The two major families (within this class) are determined by the type of salt used in the MSR (fluoride or chloride). The implications of the choice of Cl vs. F salts are varied.

Firstly, the neutron energy spectrum is impacted. Fluoride salts, given the cross sections of Fluorine, can be much more moderating. This is enhanced when coupled with Beryllium, which is a heavily moderating element (see the neutron capture probability in Figure 12B, which is low for Beryllium; thus it does not act as a neutron poison but scatters the neutrons back into the neutron flow, with reduced energy, i.e. moderated). This affects the breeding capabilities since the resulting neutron energy relatively matches the Th capture probability for thermal neutrons, allowing a thermal Th-based breeding cycle.

Chlorine has two stable isotopes, ^{35}Cl and ^{37}Cl, with natural abundances of 76% and 24%, respectively. Practically all designs require enrichment in Cl-37 due to the higher neutron capture probability for Cl-35 (see Figure 12B), impacting negatively the neutron economy. As can be seen in Figure 12B, F-19 has a lower neutron capture probability than Cl-37 and is much lighter. These two factors combined increase its moderating capabilities with respect to Cl-37: the interacting neutron thus needs about twice as many collisions to slow down from 2 MeV to 1 eV. See Figure 13.

However, Cl-37 generally has a low neutron absorption section, and combined with a low scattering cross section, it leads to having fast spectra [37]. Therefore, neutron spectra for Chloride fast reactors are much harder than for fluoride fast MSRs (and belong to the hardest spectra of all fast reactors [37]). This is also the most appealing feature of Chloride-based MSRs.

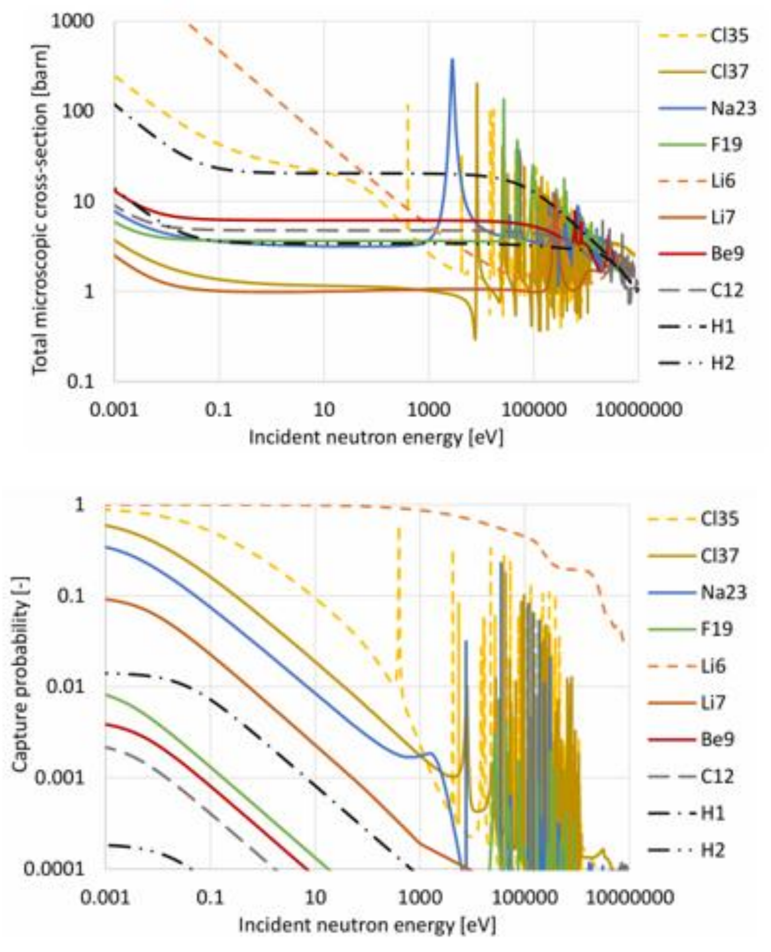


Figure 12 A and B: Microscopic cross sections of selected nuclides (total and capture) [37]

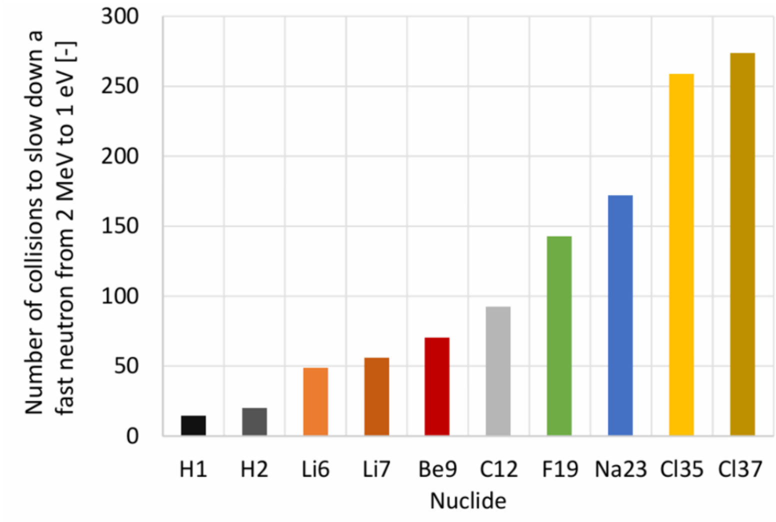


Figure 13: Number of collisions needed for effective moderation (from 2MeV to 1eV) for selected nuclides [37]

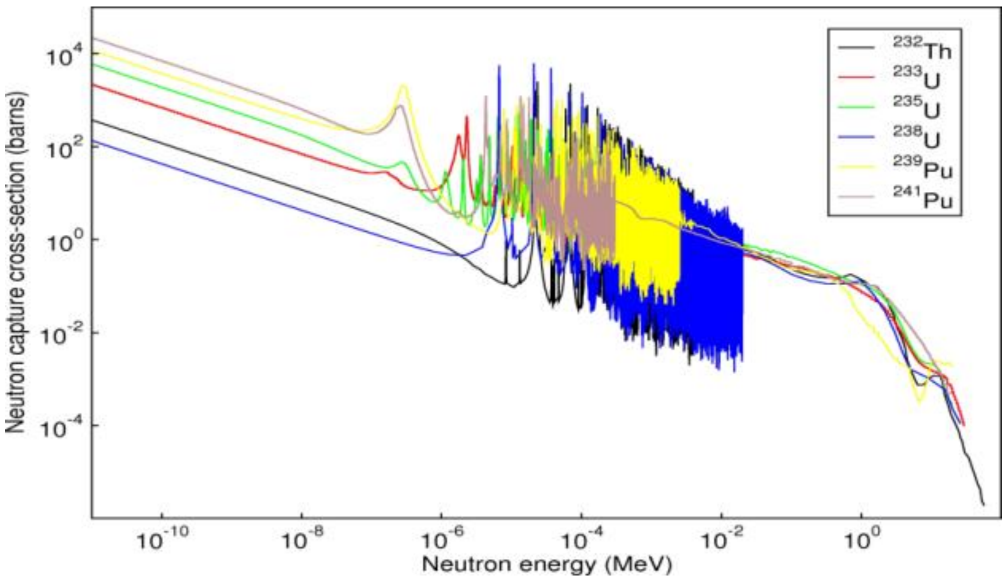


Figure 14: neutron capture cross-sections for actinides of interest with respect to the incoming neutron energy [51]

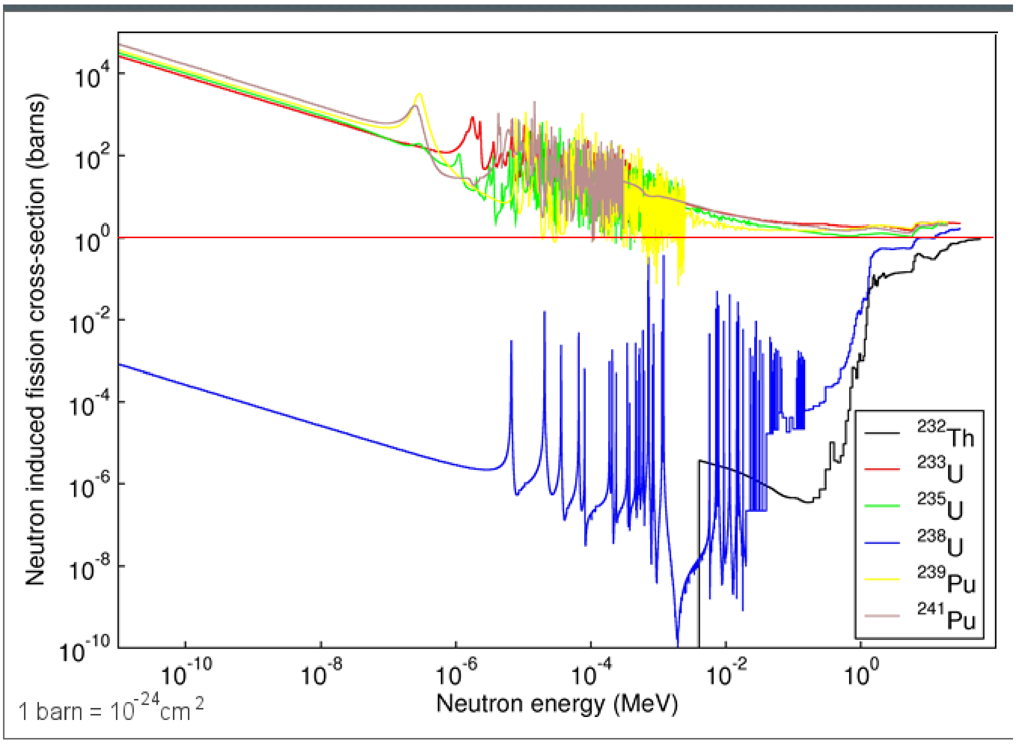


Figure 15: Nuclear fission cross-sections for actinides of interest with respect to the incoming neutron energy [52]

Finally, all factors combined, Cl-37 enables efficient breeding in both Th-U and U-Pu closed cycles (given operation in the high cross-section zone of fertile isotopes for neutron capture, implying breeding), and operation in an open breed-and-burn U-Pu cycle [37]. Chlorine isotope separation appears to be both technologically feasible and not cost-prohibitive, which further motivates the R&D in this technology [47]. At the same time, the solubility of actinides or lanthanides in the fuel salt can be higher than in fluoride-based MSRs, which is beneficial for avoiding precipitations [37].

The hard spectrum is, however, accompanied by the core transparency for neutrons, meaning there is much more neutron leakage that must be compensated for by larger cores, reflectors, and/or blankets [37]. Secondly, the components' durability is impacted by the neutron spectra. A harder spectrum means more radiation damage, which is reduced by a factor of 5 to 7 in fluoride salts [53]. However, there are chemical implications to consider for both salts in the impact on the component's durability. For instance, LiF necessitates an enrichment to 99.995% ^{7}Li to reduce ^{3}H (tritium) production via neutron capture through the reaction $^6\text{Li} + \text{n} \rightarrow {}^3\text{H} + \alpha$ that has a cross section of ≈980 b for incident thermal (0.025 eV) neutrons. For fast spectra, the reaction with ^{7}Li is possible for incident neutrons of energy greater than 2.5MeV (but the cross section is much smaller, amounting to circa 0.3b [38]. It is possible to track activation products that could be indicative of corrosion (e.g., ^{24}Na, ^{51}Cr, ^{59}Ni, ^{65}Ni, and ^{60}Co). A table summarising the chemical implications of the carrier salt choice is presented below.

Table 11: Chemical implications of the carrier salt choice [53]

Element produced	Problem	Fluoride Salt	Chloride Salt
^{36}Cl produced via $^{35}Cl(n,y)^{36}Cl$ and $^{37}Cl(n,2n)^{36}Cl$	Radioactivity $T_{1/2}$ = 301000y		10 moles / y (373 g/y)
^{3}H produced via $^{6}Li(n,\alpha)$ t and $^{6}Li(n,t)$ α	Radioactivity $T_{1/2}$ = 12 years	55 moles / y (166 g/y)	
Sulphur produced via $^{37}Cl(n,p)^{34}P(\beta\text{-}[12.34s])^{34}S$ and $^{35}Cl(n,\alpha)^{32}P(\beta\text{-}[14.262\ days])^{32}S$	Corrosion (mainly located in the grain boundaries)		10 moles/year
Oxygen produced via $^{19}F(n,\alpha)^{16}O$	Corrosion (surface of metals)	88.6 moles/y	
Tellurium is produced via fissions and extracted by the online bubbling	Corrosion (cf. Sulphur)	200 moles/y	200 moles/y

Class III

Class III are Heterogeneous MSRs. These are characterised by the presence of structural materials in the core that separate the fuel salt from a dedicated coolant or from a moderator not directly compatible with the fuel salt. It could be the case of Chlorine-based designs that use graphite as a moderator, requiring cladding. Non-graphite moderated MSRs utilise fluoride fuel salts diluted in fluoride carrier salt and are characterised by a thermal spectrum. There are some moderators (e.g. based on beryllium or deuterium) that can ensure better neutronics performance than graphite but require a physical separation from the salt [37]. Such concepts can be potentially better breeders or waste burners than graphite moderated MSRs (see 3.3.2 for more details). Heterogeneous chloride fast MSRs use chloride fuel separated from the coolant by the structural material. The coolant can be another salt or liquid metal, typically lead [37].

Class IV

Class IV is made up of reactors that do not fall into these three categories or share features of them.

Families

The four classes are divided as described into families according to the main design characteristics. The third layer of the taxonomy is the classification of each family into different types.

A comprehensive table of features is shown below.

Table 12: Comparison of parameters for the six major MSR families [37]

Class	I. Graphite based MSRs		II. Homogeneous MSRs		III. Heterogeneous MSRs	
Family	I.1. Fluoride salt cooled reactors	I.2. Graphite moderated MSRs	II.3. Homog. fluoride fast MSRs	II.4. Homog. chloride fast MSRs	III.5. Non-graphite moderated MSRs	III.6. Heterog. chloride fast MSRs
Fuel state	Solid	Liquid	Liquid	Liquid	Liquid	Liquid
Spectrum	Thermal	Thermal	Fast	Fast	Thermal	Fast
Salt type	Fluorides	Fluorides	Fluorides	Chlorides	Fluorides	Chlorides
Neutronics performance	Burner, converter	Burner, converter, breeder	Burner, converter, breeder	Burner, converter, breeder, breed-and-burn	Burner, converter, breeder	Burner, converter, breeder, breed-and-burn
Actinides	Enriched U, TRU, Th as semi-inert matrix	Enriched U, TRU, closed Th–U cycle	Enriched U, TRU, closed Th–U cycle and U–Pu cycle	Enriched U, TRU, closed Th–U cycle and U–Pu cycle	Enriched U, TRU, closed Th–U cycle	Enriched U, TRU, closed Th–U cycle and U–Pu cycle
Irradiation induced issues	Limited burnup of fuel in graphite matrix	Limited graphite moderator lifespan	Limited vessel lifespan	Limited vessel lifespan	Limited vessel and structural material lifespan	Limited vessel and structural material lifespan
Fuel extensive pumping	No	Yes	Yes	Yes	Yes (no if cooled by moderator)	No
Heat transport medium	Fluoride coolant salt	Fluoride fuel salt	Fluoride fuel salt	Chloride fuel salt	Fluoride fuel salt (or moderator)	Molten salt or lead coolant
Primary heat exchange	In-core	Ex-core	Ex-core	Ex-core	Ex-core (in-core)	In-core

In summary, the MSRs are divided into three major classes according to the type of materials present in the core. Accordingly, the active core of the first class includes graphite and salt, the second class relies solely on salt, and the third class includes structural material that separates the fuel from the coolant or moderator.

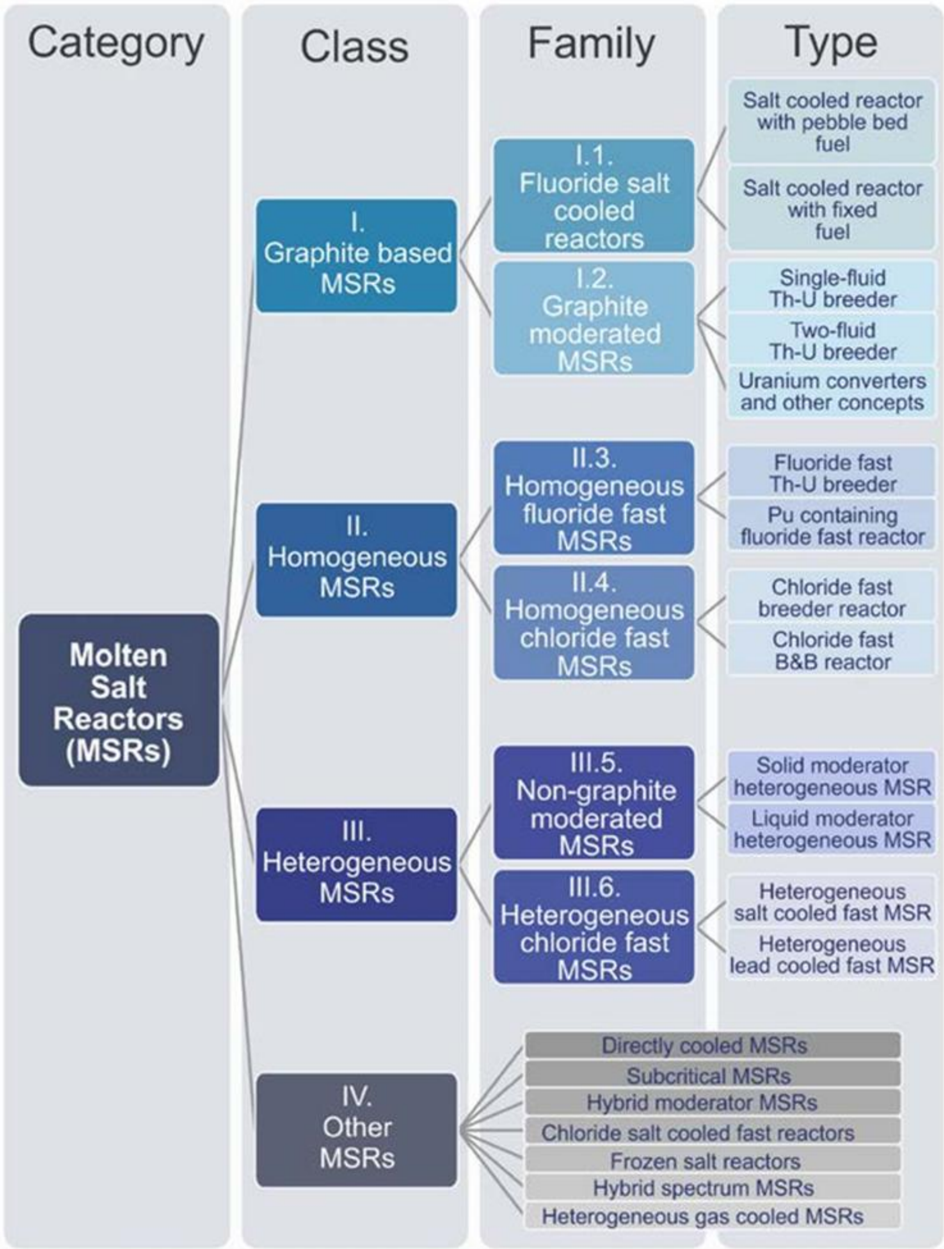


Figure 16: Block diagram for the IAEA's taxonomy [37]

3.2.3. GIF's taxonomy

The GIF, through its PR&PP workgroup, characterise MSR designs based on the relevant accountancy (and PP) challenges. From the variety of fuel salt configurations and fuel cycle implications, the categorisation boils down to three classes with similar safeguards approaches.

- Liquid-fuelled with integrated salt processing: onsite salt processing is part of normal operation, and fissile materials are both removed from and reintroduced to the active salt circuit. While these designs do have considerable variation in fuel choice, one or two fluids, core neutronics, and power density, they all have liquid-fuelled cores and propose to perform fissile material separation on-site to enable achieving a closed fuel cycle

- Liquid-fuelled without integrated chemical fuel salt processing: only the materials that inherently separate from the liquid fuel salt (gases and the non-soluble, noble metals) are removed, but additional dissolved elements are not chemically separated onsite as part of the normal operation. In some designs, the entire fuel salt (or entire core) will be replaced periodically, and the salt may be subsequently processed at an external site.
- Solid-fuelled with salt coolant: the third primary category of molten salt reactors (MSRs) utilises solid fuel while employing molten salts strictly as a coolant. In these designs, the reactor core is typically fuelled by TRISO particles, often arranged in pebble beds, and cooled by a salt mixture. This configuration supports efficient neutron economy and ensures an overall negative reactivity and temperature feedback coefficient. To achieve these, however, the moderation by the graphite in the pebble is not sufficient, and needs to be complemented by a moderation from the salt. This implies having fluoride as the coolant salt, without ^{6}F due to its high capture cross section, and ^{9}Be for its high scattering cross sections (FLiBe salts) [37]
- A further characterisation that is not proposed by the taxonomies, that is very relevant for PR in general, but also for establishing a safeguards approach, and impacting on the safeguards process effectiveness, is the fact that few vendors plan to implement barge-based MSRs that would be both floating on water, and mobile (see Seaborg).

3.3. MSR fuel cycles peculiarities

From the presentation of the past experiments, the taxonomy, and the introduction to ENDURANCE’s designs (**Erro! Fonte de referência não encontrada.**), it is shown that MSRs can have different purposes than mere electricity generation. Another evidence is that to achieve their purposes, whether it is energy generation, waste burning, or breeding, MSRs can vary a lot in their fuel and fuel cycle. Adaptation to the new fuel cycles might generate several challenges both in supply-chain management and Safeguards applications over the fuel cycle. The reference fuel cycle and fuel supply chain has been previously described/described in Appendix.

3.3.1. The ^{232}Th-^{233}U cycle

In [36], the Thorium (Th) fuel cycle is defined as "an entire nuclear fuel cycle in which at least one of its stages utilises or depends on the use of natural thorium or ^{233}U created by Thorium irradiation." One step with Th is sufficient to characterise a Th fuel cycle, while some steps might not include Th or ^{233}U at all.

Thorium is, on average, three times more abundant in nature than U. However, Th is less likely to be exploitable through geological deposits. Nevertheless, Th constitutes an attractive material for pursuing a nuclear fuel cycle for countries with large thorium reserves, and particularly for countries with limited access to natural uranium deposits or enriched uranium markets. The use of Th fuels complements that of U and helps ensure the long-term sustainable use of nuclear power through the increase in the strategic reserve of nuclear fuel.

The literature recognises a few advantages to the Th-U cycle when compared to the U-Pu cycle.

- Thorium fuel cycles have thermal breeding capability. The key metric is η, defined as the average number of neutrons produced per neutron absorbed in the fissile fuel. Crucially, the η value of ^{233}U when fissioned by thermal neutrons is significantly higher ($\eta \approx 2.3$) than that of ^{235}U or ^{239}Pu ($\eta \approx 2.1$). This means that when a ^{233}U is fissioned, one neutron goes on to pursue the fission chain reaction, one neutron can breed ^{232}Th into ^{233}Th (which would eventually decay to ^{233}U), and the remaining 0.3 neutrons on average are enough to compensate for the parasitic losses (due to the absorption of structural material and FPs, which should be minimised) and neutron leakage (which can be reduced as well by employing neutron reflectors, as described in the design of the MSFR). Instead, the 2.1 value of η for U/Pu is not enough to reliably perform breeding, even in a liquid-fuelled thermal MSR and the associated reduced parasitic losses. η increases to 2.5 and above when considering a fast-spectrum reactor. On the other hand, the thermal capture cross-section of ^{232}Th is about three times larger than that of ^{238}U, and a much higher starting fissile enrichment is required for ^{232}Th-fuelled cores in thermal spectrum reactors to begin the self-sustaining reaction chain. Similarly, its fission cross section is non-existent for neutron energies below 0.0015eV, which complicates the start but simplifies the control of the chain reaction [36].
- SNF from thorium fuel cycles has lower radiotoxicity over time (but increased in the short-term) [36]. The mass number of ^{232}Th is six units less than that of ^{238}U; thus, many more neutron captures are required to transmute Th to the first TRUs, which are highly radiotoxic. [45]

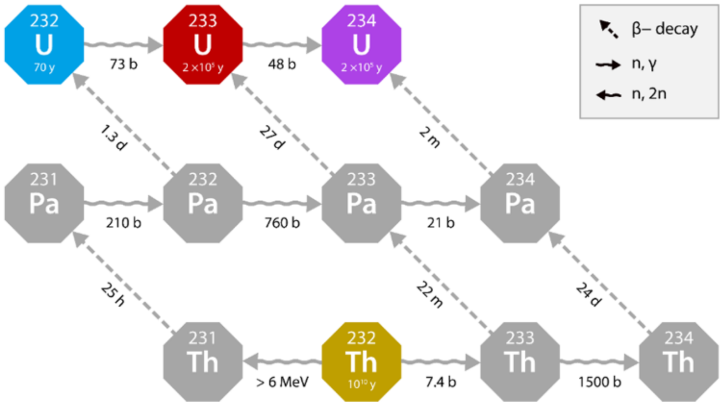


Figure 17: The production of U-233 from Th-232 [36]

Given the experience of use of the Th-U cycle applied to a liquid-fuelled MSR gained from the MSRE [54], and the peculiar possibility of FP cleaning of the salt, the liquid-fuelled MSR technology is particularly suited to the Th-U cycle and is thus one of the main candidate designs for application to this cycle.

A key aspect of the Th-U cycle is the co-production of ^{232}U initiated by an energetic neutron (n, 2n) reaction (reaction Q value -6.43 MeV [49]) in ^{232}Th according to the evolution below:

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

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

Figure 18: Creation of U-232 from Th-232 [49]

One of the daughter radionuclides of ^{232}U, Thallium-208, is a strong emitter of ionising highly energetic (2.6 MeV) gamma rays with high absolute emission probability, which poses serious health risks and the need for remote handling. From a PR point of view, the presence of Thallium makes the Th-U cycle more proliferation resistant as it acts as a self-protecting mechanism against human intervention.

Figure 19: The figure shows the decay chains of U-232 and U-233 (m for months) [36].

3.3.2. Actinide Management (and Breeding) Designs

The management of long-lived radioactive actinides, particularly minor actinides (MAs) such as Neptunium (Np), Americium (Am), and Curium (Cm), constitutes a fundamental challenge within the nuclear fuel cycle. These transuranic (TRU) elements, generated through neutron capture and decay processes in conventional reactors, are the primary contributors to the long-term radiotoxicity and thermal load (decay heat) of SNF, necessitating geological isolation timescales exceeding 100,000 years. The management of MAs hinges on understanding their interaction with neutrons.

Transmutation is often undesirable for waste management because it transforms one long-lived actinide into another, sometimes even more problematic, isotope. The desired outcome for "burning" or "transmuting" MAs is fission, more favourable in a fast neutron spectrum for most MAs.

The current open fuel cycle approach, based on direct disposal, presents significant environmental and political challenges. MSR technology, particularly liquid-fuelled configurations, enables a transformative shift in actinide management strategy, from passive storage to active transmutation, working towards the closure of the cycle. In solid-fuelled systems, valuable neutrons are lost (parasitically absorbed) by structural materials, cladding, and fission product poisons that accumulate irreversibly in the fuel. Moreover, the degrading ceramic in which actinides are constrained (UO2 pellets), due to its mechanical stresses, poses a limitation on the achievement of deep-burning. Instead, a liquid-fuelled MSR has no fuel cladding and minimises structural material in the core. More importantly, it can continuously (or semi-continuously by batch) remove neutron-absorbing FPs. For instance, Xenon-135, which has a severely impacting thermal capture cross-section, can be removed by flotation due to its low solubility in molten salts from the fuel salt via the online gaseous extraction processing system. The choice of whether to remove other FPs, like lanthanides and noble metals, depends on the intended goal of the design. For actinide burners, the noble metals will plate out onto salt-wetted surfaces and tend to diffuse into the substrates and may provide some corrosion protection, so their removal could result in non-beneficial consequences [49]. For breeder designs, an additional semi-continuous salt processing unit may be included for fission product removal and actinide separation [49].

This contributes to the maintenance of optimal neutron economy, meaning that every neutron has a higher probability of interacting with a fissile or fertile atom, including the minor actinides, which is crucial for transmuting heavier actinides that require higher neutron fluxes. The liquid state ensures a homogeneous mixture of all actinides, eliminating localised power peaking problems associated with heterogeneous fuel assemblies in solid-fuelled systems. Furthermore, fresh actinide materials, whether derived from reprocessed spent fuel or dedicated targets, can be introduced continuously to maintain optimal reactivity levels throughout the operational cycle. MSRs in an Actinide Burner Configuration are designed with a conversion ratio below 1 (of the fertile isotope’s breeding) to prioritise the destruction of existing actinide stockpiles. The efficiency of actinide transmutation is critically dependent on the neutron energy spectrum, with distinct advantages emerging for different MSR configurations. Fast Cl-based MSRs exhibit a higher probability for fissioning MAs with high fission-threshold cross-sections, such as Am-241 and Cm-244. This tendency for fission over neutron capture prevents the net accumulation of heavier actinides, enabling more complete waste destruction [37].

Graphite-moderated, fluoride-based MSRs operating with a thermal neutron spectrum remain effective for Pu destruction and management of MAs when homogeneously mixed with fissile material (see Copenhagen Atomics’ configuration). In a thermal reactor, MAs predominantly act as neutron poisons because at low energies, the nucleus formed after neutron absorption is more likely to de-excite by emitting gamma radiation than to undergo the large deformation required for fission. Most MAs exhibit lower fission probabilities for thermal neutrons but the efficient neutron economy of thermal MSRs can partially compensate for this limitation. While the probability that an interaction results in fission (rather than capture) may be lower than in a fast spectrum, the MSR ensures that a significantly greater number of interactions occur per neutron generated, thereby increasing the absolute rate of actinide destruction compared to traditional reactors. Finally, the MSR effectively uses neutron capture as the first step in a multi-stage transmutation pathway that ultimately leads to fission. It turns the reactor into a nuclear "assembly line" where initial capture is not a dead-end but a necessary step toward eventual destruction.

An assessment performed in [55] found that when the MSFR-Cl design, when operated on spent MOx fuels, would reduce the amount of such elements at the cycle end by 216t, while also reducing the needed U.

3.3.3. Salt Reprocessing

Even though it has been extensively addressed throughout the whole chapter, further details can be provided on salt reprocessing, given its impact on the designs.

Reprocessing can be done fully onsite or partially offsite, but the first option is more representative of the peculiarities of the reactor class, while the latter is more reminiscent of today's reprocessing schemes.

Pyroprocessing methodologies constitute the core of liquid salt management. They have been developed as an alternative to PUREX extraction (see Reprocessing) and originally involve the use of molten salts as a medium for performing a reductive separation of actinides. For this reason, they are particularly suited to use in MSR as the fuel can be directly taken from the core and reprocessed. The first advantage is that the absence of water avoids accidental criticalities (moderation). The online reprocessing of hot MSR fuel is considered a complicated radiochemical technology [56] which has a great number of limitations caused by the corrosivity of the liquid MSR fuel, its high radioactivity and heat generation and finally by the direct link to the reactor chemistry itself. Even though the online reprocessing technology was never realised, there is a significant background from the ORNL experiments [54]. In MSRs, the central objective is the extraction of FPs while ensuring that actinides are retained and recycled back into the primary circuit. The key pyrochemical technologies derived from early ORNL work and current research efforts are electrochemical separation, fluorination, and molten salt/liquid metal extraction (reductive extraction), each suited to specific separation tasks, probably performed in sequence as shown in the design of the MSFR (see **Erro! Fonte de referência não encontrada.**) [57]. Salt volatilisation (fluorination) is designed to remove elements that can achieve high gaseous oxidation states, functioning as an off-line chemical step that oxidises components in the salt to produce gaseous products; this technique is specifically suited for extracting FPs such as niobium, tellurium, iodine, molybdenum, chromium, and technetium, along with actinides like U, Pu, and Np. In the MSFR, only the actinides are sought to be separated through this method by converting, for instance, liquid UF_4 to gaseous UF_6, which is then extracted and then reconverted to UF_4. Electrochemical separation is highly favoured because it avoids introducing external chemicals into the molten salt and separates FPs based on their redox potential. Electrorefining extracts metallic U on a cathode, typically made of graphite. Electrowinning extracts TRU metal on a liquid cadmium cathode while noble metal FPs remain on the anode [56]. Electroreduction of molten salt/liquid metal extraction involves contacting the molten fuel salt with a liquid metal, such as liquid bismuth, to perform crucial chemical partitioning. Because Bismuth can dissolve Th and U, as well as rare earths, it allows the selective extraction of these metals from the salt phase into the bismuth phase, especially when combined with reducing agents like lithium, which are also soluble in Bismuth. For these reasons, this method is particularly suited for the effective separation of actinides from lanthanide FPs [31]. This is the method used in the MSFR reprocessing, as seen on the schematics [49]. The electrical separation techniques can be performed in series, too [31].

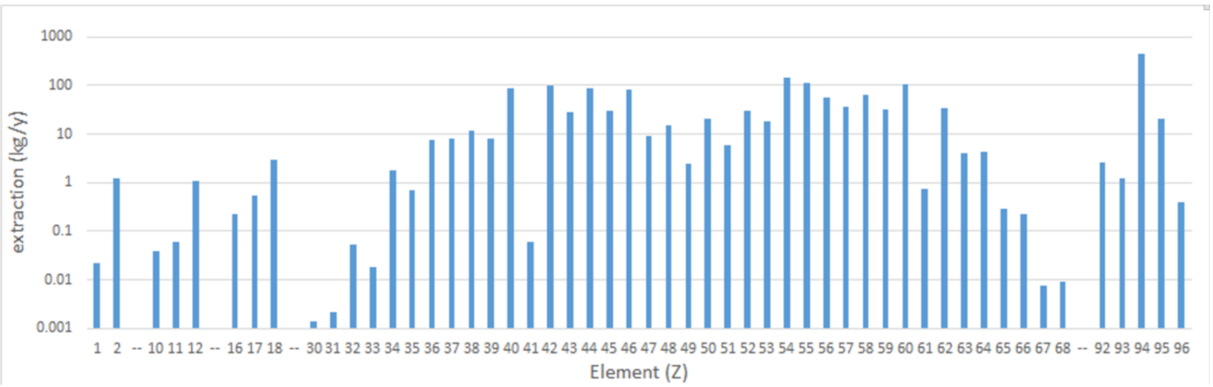


Figure 20: total (fuel and fertile) extraction rates of elements for the MSFR-Cl fuelled with spent MOX TRUs (average 50y). [55]

[55] studied in an MSFR-Cl fuelled with spent MOX's TRUs, the flow rates (kg/year) of elements that must be processed by the reprocessing unit every year, considering both the online gas removal system and a reprocessing system of 100L/day capacity. (This is higher than the F-fuelled MSFR, but it is consistent with the fact that the total salt volume for the Cl version is significantly higher). A significant amount of fission products produced are in gaseous form, with Xenon (Z=54) being the most produced (about 150 kg/y). This highlights the importance of the gas removal system to maintain a low pressure.

The SCALE suite, developed and maintained by ORNL [58], serves as a depletion modelling tool (that calculates decay after a given burnup is obtained). It has been extended with MSR-specific capabilities through tools like ChemTriton and enhanced TRITON sequences. ChemTriton, a Python-based material tracking script, enables the simulation of continuous material feeds and removals while tracking isotopic compositions throughout the fuel cycle. TRANSFORM (Transient Simulation Framework of Reconfigurable Models, ORNL-developed) can track the entire, comprehensive inventory of fission products and actinides. The model includes the behaviour coupling of fission product movement in the system with reactor kinetics and thermal hydraulics. However, the level of model complexity determines the computational expense, which can become impractical for real-time tracking [38]. These tools represent significant advances in MSR modelling capability, but they also highlight the fundamental challenges in adapting solid-fuel reactor codes to liquid-fuel applications.

Table 13: Summary of extraction and reprocessing methods and MSR applicability (preliminary evaluation) (Source material: Pavel Souček, European Commission, Joint Research Centre Karlsruhe, 2025 [59])

Method	Principle	Property governing selectivity	Separated elements	MSR applicability
Gas extraction process (He bubbling)	Bubbling or sparging the salt with inert gas	Solubility of gas in the melt	Noble gases (Xe, Kr) and insoluble noble metals	YES
Fused salt volatilization	Fluorination / chlorination of the melt	Volatility	Uranium, possibly Pu in fluoride melt	YES
Electrorefining	Electrochemical dissolution and deposition in molten salt	Electrochemical stability (red-ox potential, activity coefficient in the melt)	U, Pu, MA	NO
Electrowinning	Electrochemical reduction of species dissolved in molten salt		U, Pu, MA	YES with limits
Reductive Extraction	Chemical reduction of species dissolved in molten salt	Chemical stability (activity coefficient)	U, Pu, MA, RE	YES with limits
Oxidative Precipitation	Oxidation of dissolved species to form insoluble oxides	Solubility of oxides in molten salt	Actinides, RE in chlorides	YES
Melt crystallization	Gradually solidification under controlled temperature	Solubilities in the salt and melting points	Sr, Cs	YES
Aqueous solvent extraction	Transfer of species between aqueous and organic phases	Affinity to aqueous and organic phases	U, Pu MA, RE – under development	Is evaluated

3.3.4. Supply chain issues

The shift towards the deployment of liquid-fuelled MSRs introduces several peculiar supply chain challenges distinct from those encountered in solid-fuelled systems. Firstly, a robust commercial capacity for producing the MSR fuel salts is currently non-existent; while minute quantities of pure salts are produced at the laboratory scale for property measurement and feasibility tests, scalable technologies necessary for chlorinating and fluorinating metallic and oxide feedstocks to produce commercial quantities of fuel salt are still under development [60]. While fissile or fertile materials, such as enriched U, Pu, TRUs, or Th, must be dissolved in the molten salt carrier to form the fuel, scalable technologies necessary for converting metallic or oxide feedstocks into commercial quantities of purified fluoride or chloride fuel salt are still undergoing development [61] [45]. Furthermore, ensuring the extremely high purity of the fuel salt is paramount for mitigating corrosion and guaranteeing compatibility with structural materials, given that corrosion rates are dependent upon impurities such as oxygen, water, and sulphur [37]. Yet, the specialised technologies required for the removal of critical impurities from chloride and fluoride salts remain largely experimental and necessitate considerable scale-up before commercial viability [61] [45]. This challenge is compounded by the design requirements of many MSRs to utilise isotopically separated carrier salts, notably enriching lithium to the stable isotope ^{7}Li (often to 99.995%) to minimise tritium production and reduce its permeation in metals and its neutron absorption, necessitating substantial infrastructure and scalable production methods that are currently lacking [58]. Finally, the qualification and licensing process is inherently complicated by the absence of consensus standards regarding acceptable purity levels, permissible contamination limits, and comprehensive component specifications, all of which are critical for the procurement of fuel salt and the certification of new suppliers [37] [60]. This situation is further exacerbated by the fact that consensus standards for property measurements applied to molten salts are not yet available, and the application of conventional nuclear standards directly to the specific environment of the MSRs is notably challenging, hindering the efforts to qualify the required supplier base [37].

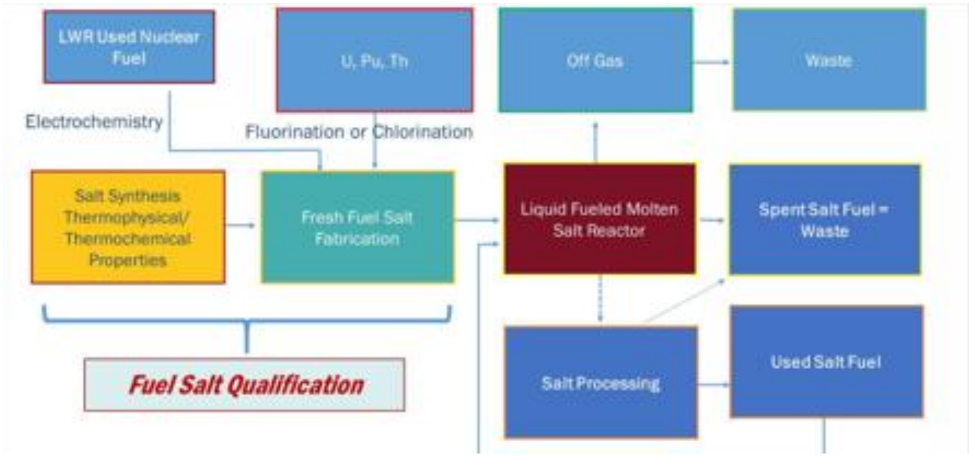


Figure 21: Generic Fuel cycle for a liquid-fuelled MSR [60]

Even when assuming scalability of production, few of the new reactor developers provide information on where the fuel salt synthesis is intended to occur. It is not clear whether uranium or thorium metal will be transported to the site and then converted to a fuel salt, or fuel salt will be synthesised elsewhere and then transported to the site. The recent and ongoing development of non-aqueous, organic fuel salt synthesis methods under mild conditions increases the likelihood of nearby fuel salt synthesis. Storage locations and quantities are also unknown [49].

3.4. MSR: Safeguards Challenges and Opportunities

The deployment of MSR technologies presents a paradigm-shifting challenge for international safeguards systems, raising fundamental questions about the readiness and adequacy of the current verification approaches. As the diverse family of MSR designs gains international momentum towards deployment, the IAEA and the broader international safeguards community face unprecedented verification challenges that current frameworks seem ill-equipped to address. These challenges might not merely require incremental improvements to existing approaches but might pose the need for a fundamental re-evaluation of safeguards concepts, technologies, and implementation strategies. In this sense, the international community should ask itself, and answer, the following questions, explicitly stated in [47]:

• *Have the appropriate safeguards and approaches been determined for MSRs?*

• *Are the safeguards approaches for one MSR design valid for another design?*

• *Is the International Atomic Energy Agency (IAEA) and international safeguards system ready for MSRs? If not, what steps should be taken to prepare?*

• *Are the safeguards inspection regimes of today valid for proposed MSR designs and the associated fuel cycles?*

• *Is the safeguards technology of today sufficiently mature to meet the verification challenges posed by MSRs and their associated fuel cycles?*

• *Is safeguards technology readily fieldable? For example, are non-destructive assay technologies and other measurement instruments ready for deployment to meet these new verification challenges?*

While exploring the answers to these fundamental questions, it becomes clear that the intrinsic properties of the MSR technology, up to the sub-atomic level, offer both significant challenges but also opportunities for exploration of new technologies and verification methods. The criticality lies in the establishment of the requirements and the methodology for a substantially different approach to NMA.

The challenges in the establishment of an NMA approach are multifaceted and stem from MSRs' fundamental differences from an item-based accountability. Beyond pebble bed reactors, which have countable numbers of semi-distinguishable items, more stringent nuclear material accountancy measures will likely be required to verify the quantities, locations and movements of the nuclear material, including fuel flow monitors, seals, video surveillance, the use of sensors to trigger other sensors, more accurate NDA measurements and sampling plans that select additional items for verification [62]. Liquid-fuelled MSRs do not contain their fuel in assemblies, making traditional item counting and visual accountability impossible; instead, these facilities resemble bulk accounting facilities like reprocessing plants, where inventories must be determined based on measurements of the actinide content in the salt [63]. [46] states that *"modelling is a vital part of the safeguards, especially at the early concept stage of development. This includes an assessment of the inventories (masses, isotopics, source term, gamma and neutron signatures, etc.), their locations in the core and associated fuel cycle, and safeguards approaches and instrumentation needs"*. It is expected that modelling will play a crucial role throughout different domains in the challenges and opportunities of safeguards in MSRs.

3.4.1. The Impact of Uncertainties

3.4.1.1. Uncertainty in Inventory Determination

To accurately perform Nuclear Material Accountancy (NMA), control of the composition of the declared NM has to be performed through verification of these compositions. To be able to determine whether the declared NM conforms to expectations (rather than declarations at this step), a rigorous modelling and assessment of the depletion and decay of the fuel has to be performed to assess the expected inventory isotopic composition.

Unlike bulk-handling facilities, MSRs do not have high material throughput, and both create and consume fissile materials, making the codes already in use unapplicable, and thus presenting a challenge to assessing what a correct inventory would contain at a given point in time. This is further complicated by the fact that nuclear cross sections and reactor physics in molten salts (especially fast spectra or Chloride-based) have sufficient remaining uncertainty to prevent an accurate evaluation of the intended fissile inventory, even if such an inventory were static [63]. Complicating material accountability further is the continuous variation of isotopic concentrations within the fuel salt, which are driven by burnup, transmutation, plating out of materials on surfaces, and online chemical reprocessing. The modelling of these phenomena requires sophisticated multi-physics coupling capabilities that account for the relationship between neutronics, thermal-hydraulics, and fuel chemistry (ChemTriton and TRANSFORM have been mentioned in Salt Reprocessing). The liquid fuel's movement through the reactor core and primary loop creates strong coupling between traditionally separate physics domains. Temperature feedback directly affects neutron flux distributions, while the spatial distribution of delayed neutron precursors changes as the fuel salt circulates through the system [32]. Different elements are fed into and removed from the system at varying rates, influenced by passive or active chemical processing methods, making it necessary not only to predict the isotopic composition at each time step but also to model the continual feed and removal of specific elements, tracking their locations and eventual decay products throughout the system. The capability to model the material being removed or fed into the molten salt is not necessarily difficult, but then it introduces large approximations because it reduces physical processing schemes and efficiencies into simpler removal actions. The MUF becomes heavily reliant on computed addition and removal terms rather than straightforward physical measurements, leading to the accumulation of prediction errors [63] [46]. A known challenge to the application of safeguards to pyroprocessing is the lack of an accountability tank to determine the NM concentration [31], emphasising the inventory determination challenge of such processes with respect to established technologies (PUREX). FPs that release neutrons upon decay can move away from their birth location and potentially decay outside the core. The impact on reactivity control and spatial power distributions requires new approaches to neutron kinetics modelling that traditional reactor physics codes cannot adequately address [32].

To inventory the NM in bulk form, the mass or volume of the NM must be measured [63]. Measuring the total salt volume at any given time across the entirety of the MSR process flow with varying temperatures and salt conditions is likely to be considerably more difficult than in present bulk processes (such as the previously mentioned reprocessing tanks, which still require thorough calibration) [63]. Similarly, the determination of total salt mass is hindered by the distinct geometric and structural configuration of MSRs, made up not only by the core, but also the heat exchangers, piping, and the salt processing subsystems, elements in which the fuel is present at any time and thus make precise modelling, but also the measurement, of the total salt mass extremely challenging [63] [46] [49], finally making this determination method unavailable or not informative (due to the reprocessing, knowing the mass might not be informative on the isotopic content). On the other hand, given the liquid nature of the fuel salt and the fast circulatory speeds (3-4s for the MSFR), the material is homogeneous, and small samples are representative of the isotopic and chemical composition of the entire fuel inventory [37]

In summary, the reliability of NMAC/NMA practices in MSRs fundamentally depends on the validation of the computational codes used to predict nuclear material quantities [64]. However, owing to the limited operational history of MSRs, there is a lack of experimental data needed to verify and validate the advanced Multiphysics computational models [63] [64]. These difficulties are further exacerbated by the MSRs' environment that regroups high temperatures, intense radiation, and corrosiveness, imposing significant technical barriers to the applicability and fidelity of sampling methods for DA characterisations, NDA verifications, and other instrumentation for physical inventory assessments [46], making the collection of data for model validation impractical in the short term. Until the intrinsic uncertainty in multiphysics codes is evaluated and reduced, relying on material balances within the challenging-to-access reactor confinement area remains highly problematic. This could be compensated by reliance on robust material control measures, containment, and surveillance [63] (within the USA's safeguards framework, control is associated with containment and surveillance measures).

3.4.1.2. Uncertainty of Measurements

NDA and DA measurements encounter severe complications, with nuclear material accountancy systems needing to verify that any MUF is within IAEA allowable ranges through high-accuracy measurement systems to meet safeguards timeliness goals. The main challenges are the high radiation and high corrosivity. The high radiation makes for a high-background noise environment, necessitating instruments with high resolution but adjusted efficiency, perhaps through additional shielding.

The highly corrosive environment of molten salts makes the maintenance of safeguards monitoring instrumentation particularly challenging, with online instruments in direct contact with salt experiencing short operational lifetimes. This can be partially counterbalanced by the plate-out of rare earth elements and noble metals in reactor components and piping, but it would present additional measurement challenges (interferences and reduced signal intensity) and increased inaccuracies [49] [63].

Sampling and DA of molten salts remains technically challenging due to the unique combination of high temperature and high radiation environments present in the salt fuel, which should be sampled and measured shortly after the core's exit (particularly with the ^{232}U contamination of ^{233}U). This significantly impacts the ability of human inspectors to perform safeguards verification measurements, impacts the durability of the instrumentation, and poses the need for remote inspections and measurements (with robotic inspection units, for instance) [36].

The challenge of accurately measuring the NM is further compounded by the fact that the NM may plate out or accumulate (see DPA) in various system components. Holdup is difficult to quantify and a challenge to NMA [65] [63]. A signature of interest is zirconium (Zr). Zr is soluble in the fuel salt and chemically follows the molten mixture. Therefore, detection of ^{95}Zr would indicate fuel salt deposition. Other isotopes of Y, Cs, Ce, Ba and La were also expected to track with the salt [38] (adapted from the findings on the MSRE [54]). The initial holdup in a new facility, which constitutes the hidden inventory, can be from 1 to 10% of the initial input of material [65], and is made up of NM that cumulates, for instance, in the porosity of the metallic structure, and is not recoverable by cleaning actions. It is, however, stable in time and should be readily quantifiable (even if the exact location of such inventory cannot be assessed). [4]

Two paths must be undertaken simultaneously to close the gap in the uncertainty of measurement. On one hand, instruments with better resolutions should be developed and used, such as microcalorimetry, when applicable. On the other hand, a thorough investigation into what these instruments seek to recognise must be done. In other words, the NM signatures have to be modelled and verified. Modelling challenges are present once again, stemming from the varying isotopic composition of MSRs, which can be significantly different from conventional LWRs.

[4] designers can refer to these documents for guidance on the reduction of holdup, which can impact their performance: "*NRC Regulatory Guide 5.8, "Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations," May 1974; NRC Regulatory Guide 5.25, "Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations," June 1974; NRC Regulatory Guide 5.42, "Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations," January 1975.*"

3.4.1.3. Uncertainty, Incompleteness and Changes in Signature Modelling and Applicability of NDA techniques

Unlike conventional reactors, where fuel assemblies have relatively predictable isotopic compositions based on burnup history, MSR fuel compositions depend on complex interactions between irradiation, processing, and transition of the fuel outside of the reactor's neutron flux [46]. In present-day LWR spent fuel, the gamma-ray emissions are dominated by ^{137}Cs, and the measurements are usually performed after a few years of cooling time, allowing the decay of short-lived FP. In MSRs, the circulating nature of the fuel allows and imposes the need for immediate measurements, perhaps minutes after the removal from the core. Both neutron and gamma signatures would be drastically changed, and new predictions have to be made.

The issue of signature prediction stems from the inventory determination issue. The material balance for MSRs will require a calculated component that also contributes to the overall material balance uncertainty. Analysis using the scale code system shows that the lower bound for expected computational uncertainty is around 4% for thermal spectrum designs and 0.25% for fast spectrum designs that are considered. The uncertainties in nuclear data (like cross-sections), together with the computational models' approximations, will lead to a compounded uncertainty in the results (further information on what the impact on material balances of a 1% inaccuracy can be is found in the MBA definition) [32].

Efforts have been made to model both the neutron and gamma signatures for MSRs' SNF [46] [37] [38] [66].

Neutron Signatures

The Pu neutron signatures can be obscured by the contribution from other actinides (e.g. Cm) and neutrons from (α,n) reactions.

The SF rate from ^{242}Cm is almost 2 orders of magnitude higher than that from the strongest contributing Pu isotope (^{240}Pu). Due to its short half-life (163 days), ^{242}Cm does not typically provide a significant contribution to neutron emissions in the LWR spent fuel assemblies due to the long cooling times typically required (this is evident in the assessment of SF emissions in a MSR SNF, in which the ^{242}Cm emission is not relevant after 1 year) (Table 14). The Cm/Pu ratio is obtained by DA on a sample taken from the feed materials, which will require an operative reactor. The Cm/Pu ratio would allow SF emitters identification.

The total neutron emission rates increase slightly with increasing CT, which is in stark contrast with UOX and MOX fuel types. This may be attributed to the buildup of nuclides like Am241 and their tendency to undergo (α, n) with increasing CT.

The presence of the light elements (previously discussed) in the salt will yield orders of magnitude more neutrons than in the case of LWR fuel (where the most abundant light element is oxygen). Neutron contribution from (α,n) reactions is approximately 2 orders of magnitude larger than the neutron emission rate from SF, possibly obscuring them and excluding this means of discrimination [30]

Table 14: neutron emission rates from different isotopes [30]

Isotope	Neutron emission rate from (α,n) [n/s]			Spontaneous fission neutron emission rate [n/s]		
	936 [days]	1872 [days]	2880 [days]	936 [days]	1872 [days]	2880 [days]
^{242}Cm	8.65E+03	1.11E+05	4.13E+05	1.53E+02	1.95E+03	7.29E+03
^{239}Pu	1.96E+03	3.25E+03	4.28E+03	4.73E-03	7.86E-03	1.04E-02
^{240}Pu	7.28E+02	1.90E+03	3.10E+03	2.23E+01	5.81E+01	9.45E+01
^{238}Pu	1.92E+02	1.77E+03	7.53E+03	1.38E-01	1.27E+00	5.41E+00
^{241}Am	1.21E+02	1.15E+03	3.91E+03	1.98E-04	1.89E-03	6.44E-03
^{244}Cm	2.83E+00	1.30E+02	1.22E+03	1.35E+00	6.19E+01	5.82E+02
^{241}Pu	1.42E+00	6.74E+00	1.51E+01	2.66E-04	1.26E-03	2.82E-03
^{238}U	6.64E-01	6.65E-01	6.65E-01	8.04E-01	8.05E-01	8.05E-01
^{236}U	3.14E-01	5.69E-01	8.02E-01	4.44E-04	8.06E-04	1.13E-03
^{235}U	2.78E-01	2.45E-01	2.16E-01	8.03E-04	7.09E-04	6.25E-04
^{242}Pu	8.08E-02	7.79E-01	2.74E+00	3.28E-01	3.17E+00	1.11E+01
Total	**1.17E+04**	**1.19E+05**	**4.33E+05**	**1.78E+02**	**2.08E+03**	**7.98E+03**

LEU-fuelled MSR salt will pose significant safeguards challenges as the Pu neutron emissions will be largely obscured by the Cm and (α,n) neutron emissions.

Regarding the Th-U cycle, challenges are presented from the lack of experience in safeguarding this cycle, but several opportunities arise from the advanced material characteristics. ^{233}U has an induced fission probability that is at least 1.1 more than that of ^{235}U and up to 1.7 times more, depending on the incident neutron energy. ^{233}U can be assayed by active NDA, and the knowledge of assay in ^{235}U should be transposable, allowing techniques such as differential die-away (previously described) or coincidence counters. The challenge is to be able to differentiate the two isotopes, and further R&D is needed, since gamma measures may not be applicable in the case of heavy shielding [36].

^{233}U-bearing materials in both oxide and fluoride chemical forms, emit significant amounts of (α,n) neutrons (approximately 5 per gram for oxides and an estimated 100 times more for fluorides) enabling for instance effective self-interrogation for relatively thin items, if and only if such items are surrounded by an environment capable of reflecting and thermalising the emitted neutrons back into the emitting item. ^{233}U thermal fission cross-sections have already been established as similar to and higher than those of ^{235}U [16] [36]. The prospects for self-interrogation are significantly enhanced for $^{233}UF_4$ samples compared to $^{233}UO_2$ due to their approximately 100 times stronger internal neutron source term [36]. Self-interrogation can be ruled out for metallic ^{233}U [36]. The table below summarises the feasibility of neutron NDA techniques for the Th-U cycle.

Table 15: Summary of the evaluation of the feasibility of neutron NDA techniques [36]

Assay target	Material type	Passive assay	Active assay	Self-interrogation	Note
^{232}Th	^{232}Th metal	Not feasible	Difficult, fast neutrons only (>1.5 MeV)	Not feasible	
	ThO_2	Not feasible	Difficult, fast neutrons only (>1.5 MeV)	Not feasible	
	$(^{nat}U,Th)O_2$*	Not feasible	Not feasible	Not feasible	Th signature likely masked by ^{nat}U signature
	$(^{233}U, Th)O_2$	Not feasible	Not feasible	Not feasible	Th signature likely masked by ^{233}U signature
	(enriched U, Th)O_2*	Not feasible	Not feasible	Not feasible	Th signature likely masked by U signature
	ThC*	Not feasible	Difficult, fast neutrons only (>1.5 MeV)	Not feasible	
	Th(FLiBe) salt*	Not feasible	Difficult, fast neutrons only (>1.5 MeV)	Not feasible	
^{233}U	^{233}U metal	Not feasible	Feasible, similar to ^{235}U	Not feasible	
	$^{233}UO_2$	Not feasible	Feasible, similar to ^{235}U	Difficult, potentially feasible	
	$^{233}UF_4$	Not feasible	Feasible, similar to ^{235}U	Likely feasible	

*Material type not included in the simulations or measurements.

The comparison of passively obtained self-interrogation results with results of active assay could enable discrimination between ^{235}U and ^{233}U in the case of inconclusive or absent complementary gamma measurements.

Neutrons are measured through ^{3}He detectors; however, Gamma rays can transfer energy to electrons (by Compton scattering or photoelectric effect) in the detector fill gas or the walls, yielding high-energy electrons that in turn produce a column of ionisation as they traverse the detector. This perturbs measurements. Hence, when choosing a ^{3}He detector for high gamma background applications, such as that of MSRs' SNF, the detector wall material, gas pressure, and quench are important considerations. In high-dose environments, Aluminium cladding, low internal gas pressures, and low-Z quench gases are preferred [30].

Gamma signatures

The short decay times between irradiation and potential measurement in MSRs create unique opportunities for signature analysis but also present modelling challenges. Fission products with half-lives from hours to days can provide information about recent reactor power levels and fuel processing activities. However, accurately predicting these signatures requires detailed modelling of fuel salt circulation patterns and the processing system [46].

Simulated spectra have different characteristics than commonly measured in LWR SNF safeguards (where the main gamma emitters are ^{134}Cs, ^{137}Cs, and ^{154}Eu, not in order of importance). The simulated gamma spectra by [46] of the MSR fuel salt are shown below for cooling times (CTs) of 0, 10 minutes, 12 hours, 7 days, and 30 days.

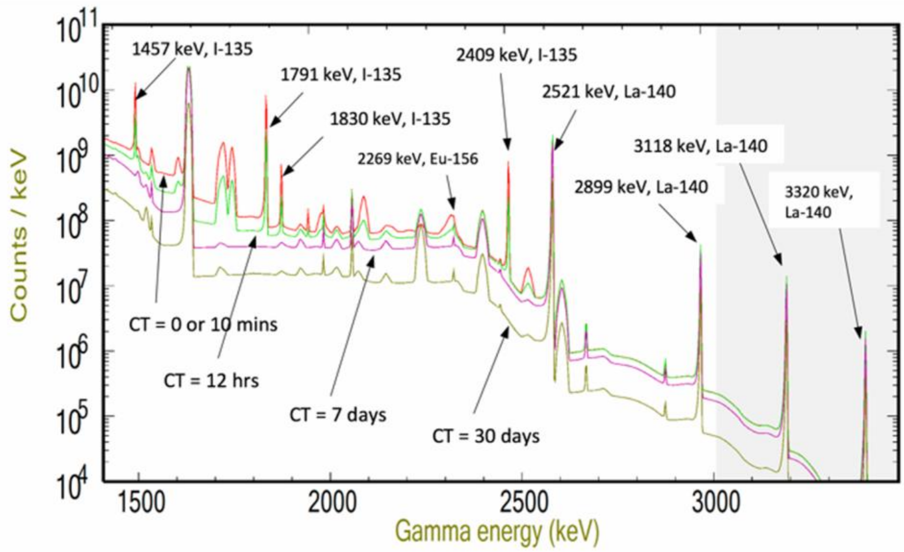


Figure 22: Simulated gamma signals seen by an HPGe detector for MSR fuel using GADRAS [46]

The detectable high-energy gamma peaks are mainly from ^{135}I, ^{140}La, and ^{156}Eu. Since all these three isotopes have short half-lives (from 6 hours to 15 days) and their concentrations mainly depend on the recent reactor powers, they can be used to monitor the recent (< 7 days) power levels of MSRs. These high-energy gamma lines also provide insight into the source of fission power from the MSR. Since ^{239}Pu produces 8 times more ^{156}Eu than ^{235}U, measuring one of the gamma lines from ^{156}Eu will provide a means to discern whether the reactor has been primarily fissioning ^{235}U or ^{239}Pu in the past few days, which may be useful for safeguards applications [46].

In the assessment of SNF signatures of the ENDURANCE's Seaborg CMSR [66] (with 10 to 20% enrichment), at a CT of 0, the main contributors to the gamma spectrum were ^{239}Np, ^{239}U, ^{140}La, ^{135}I, ^{143}Ce, ^{132}Te, ^{135}Xe, ^{133}Xe, ^{95}Nb, ^{95}Zr. For short CTs (between 1 and 10 years), ^{95}Nb, ^{106}Ru and ^{134}Cs are isotopes of interest. After one year, ^{137}Cs remained the main contributor to this specific fuel mixture and burnup [66]. Note that these signatures might be inexistent in design with online reprocessing, as the FPs are removed.

In Th-U cycles, the presence of ^{232}U might provide unique signatures to also help detect the diversion of uranium even in very small quantities. The activity of 1g of ^{232}U, related only to the γ-ray and assuming equilibrium among decay products, increases by 0.3 GBq per day during the first three months of the reactor operation, with the maximum activity reached after 10 years equal to 270 GBq (from the ^{208}Tl activity) [49]. This would generate a distinct signature for the U contained in the fuel salt and in the fertile salt (1 Bq = 1 radioactive decay per second: It measures the rate of radioactive decay, i.e., how many atoms disintegrate per second). As the fuel burnup increases, the concentration of ^{232}U increases faster than the ^{233}U concentration (even though on an absolute scale it remains typically three to five orders of magnitude lower), and the gamma detection would indicate the burnup level. As shown in Figure 8, the decay chains of ^{232}U and ^{232}Th are intertwined. Discrimination between the two is possible using high-resolution detectors only, only in non-fresh fuel, thanks to the possibility to identify ^{228}Ac, a daughter of ^{232}Th [36]. The need for high-resolution detectors impacts the efficiency of the safeguards system through increased costs and complications, like the cooling of the instrumentation.

3.4.1.4. Summary on the Uncertainties

In summary, the ability of the modelling tools to accurately capture the gamma and neutron signatures is key to being able to design the safeguards instruments and determine their requirements in terms of accuracy, response, and lifetime [46]. [36] states that *"one of the greatest general NDA challenges with respect to ^{232}Th is quantitative assay of its total mass. One of the greatest general NDA challenges with respect to ^{233}U is associated with its prompt identification, particularly if significantly shielded"*.

The material balance equation will require modification to include an estimate of actinide changes due to burnup and depletion occurring during normal operation. Two different inventory terms appear in the equation: one observed inventory derived from direct measurement and a second inventory that is estimated using computational burnup codes. Measurement conditions, incomplete knowledge of core conditions needed for modelling, and nuclear data uncertainty are among the factors creating discrepancies.

These two terms are defined as follows:

- $I_{m,t}$ - The measured (NDA and/or DA) inventory at time t
- $I_{c,t}$ - The calculated (from burnup code) inventory at time t

$$\begin{cases} I_{m,t} = B_{m,t}\, C_{m,t} \\ I_{c,t} = B_{m,t}\, C_{c,t} \end{cases} \qquad (9)\ [32]$$

Note that $B_{m,t}$ is the bulk salt measurement at time t, whereas $C_{m,t}$ is the measured concentration derived from NDA or DA measurements $C_{c,t}$ is the calculated concentration based on a burnup code. Then the MUF formula for an MSR becomes:

$$MUF = (I_{m,t} - I_{m,t-1}) - (I_{c,t} - I_{c,t-1}) \qquad (10)\ [32]$$

3.4.2. Machine Learning and Datasets

Recent developments in MSR safeguards have begun incorporating machine learning approaches to enhance detection capabilities. These methods can potentially identify patterns in complex, multi-dimensional data that traditional statistical approaches might miss. However, the application of machine learning to MSR safeguards modelling faces significant challenges related to training data availability and model validation.

[66] [67] produced a dataset of entries of different combinations of salts and the associated quantities of interest, like nuclide inventories, and nuclide-specific gamma source strength (in photons/second), decay heat (in Watts), (α, n) and SF neutron emission rates. The goal of this dataset establishment is to allow the analysis of the fuel salt properties and to allow for the implementation of ML algorithms trained on safeguards-relevant signatures.

Again, the lack of comprehensive operational data from MSR facilities limits the ability to train and validate machine learning models. But studies on new safeguards approaches can be done on simulated data, which, as all models do, may not capture all the real-world complications, but allow perhaps to set lower boundaries of uncertainty [30, 46, 32]. Other new techniques could take advantage of digital twins or similar digital technologies [57].

3.4.3. Process monitoring

Previously tested safeguards technologies from pyroprocessing processes research offer applicable measurement techniques for determining actinide content in molten salts, though radiological doses will be much higher since samples may come directly from the reactor. Decay tanks in some designs provide time for short-lived fission product decay, making post-tank sampling more optimal for process monitoring measurements [49]. Analysis of the sample material could allow determination of the quantities of fissile materials and actinide concentrations through DA and NDA techniques implemented in situ in near real time. [67] mentions the possibility of coupling pumping power signature diagnostics with neutron noise analysis for primary flow measurement, a combination of the operator's instrumentation with the inspectorate's to obtain a single shared valuable information. Another instance of process monitoring is the monitoring of the reactor temperature. Due to the negative feedback coefficients in the MSFR (but generally in the class of MSRs), the core temperature can be linked to the fissile inventory present in the core and the overall reactivity. Studies show that a disappearance of 1 kg of ^{233}U leads to a reactivity variation of 9.5 pcm, with feedback coefficients of approximately -5pcm/°C resulting in a 2°C temperature decrease, meaning diversion of one ^{233}U SQ would cause an observable 16°C temperature decrease [49].

Much available instrumentation could be applied to MSRs, especially the one adapted from the process monitoring of pyroprocessing or conventional reprocessing plants [55] [36] [30] [38] [31] [39]. Process monitoring practices from on-load reactors like CANDU could be adapted to on-load liquid-fuelled reactors. Research has demonstrated that HKED can be deployed at strategic locations to measure special receipt material concentrations in fuel salt streams entering and exiting various components such as the reactor core, processing systems, and storage tanks, with high TRL and expected high measurement accuracy [39].

Flow-Enhanced Electrochemical Sensors (FEES) and optical spectroscopy approaches are promising developments for in situ, real-time monitoring of actinide concentrations in MSR fuel matrices, but current sensor methodologies face significant technical limitations with respect to safeguards-grade measurement accuracy. FEES can achieve uranium quantification with relative errors near 5%, which is insufficient for international safeguards requiring uncertainties well below 1% for validated inventory statements [38]. In comparison, the HKED achieves sub-1% absolute accuracy for U and Pu mass determinations in reprocessing plants, far more quickly and at a lower cost than chemical methods. However, adapting HKED to the high temperatures and corrosive chemistry of MSR salt requires transforming molten salt samples to a stable aqueous phase without compromising sample representativity, while addressing complex transport phenomena like solute migration and salt freezing within sampling hardware. The sampling process itself risks breaching the integrity of otherwise sealed reactor salt circuits, but HKED’s physical decoupling from the salt stream offers substantial operational resilience, as the instrument is placed outside of the degrading environment [38].

Optical spectroscopy enables detailed resolution of uranium redox chemistry and speciation states. Both Raman and UV-vis spectra can be measured simultaneously to monitor U in different oxidation states and in the presence of interfering species, which leads to complicated chemistry under molten salt conditions. Limits of detection were determined to be in the millimolar range for the various U oxidation states within the chloride melts explored. Optical methods can capture transient chemical behaviours and material trends, though propagated volumetric and density errors currently prevent achieving the sub-0.1% accountability targets, particularly after propagating errors from volume and density measurements. Nevertheless, these optical techniques are highly effective for process monitoring to detect anomalous chemical changes indicative of protracted actinide diversion, and for identifying phenomena such as plating or precipitation that may perturb the temporal evolution of material inventories [63].

Laser-induced breakdown spectroscopy (LIBS) is being investigated for solving the real-time accountancy challenge, both in the off-gas system and in the molten salt system. While demonstrated on hydrogen isotopes, this methodology can be extended in the future to additional isotopes of interest, such as uranium and lithium [68].

Research is being done to characterise the signatures of diversion, to have a benchmarked signal to recognise with process monitoring. Advanced dynamic mass accountancy frameworks relying on equilibrium thermodynamic models have demonstrated utility in tracking the coupled behaviour of U, Pu, and MAs under continuous MSR fuel recirculation, though such models inherit the systemic uncertainties of depletion calculations and remain susceptible to compounding errors arising from uncertain operating conditions and changing salt compositions [69] [70].

[71] explores the system frequential response of the kinetic parameters (e.g. reactivity) as a means of identifying a diversion (not performing accountancy). Specific patterns for Pu and U removal are identified, providing a new means for detecting a diversion in the bulk system. However, the studied case is that of an abrupt removal of Pu, which would lead to a temperature decrease, which can be masked by a control rod movement. This technique must be complemented by control rod tracking when they are present. Additionally, the challenge of biased defects is still not resolved through this technique, and further investigation is required.

3.4.4. NMA Approach Establishment and Diversion Path Analysis (DPA) Impact

3.4.4.1. MBA definition

The definition of appropriate MBAs and internal control areas becomes critical to localise potential material loss, particularly in systems where residual holdup material left on drained components can amount to substantial quantities. The usual MBA for a NES is the whole facility or reactor building, but it is not necessarily the case, as an MBA should be as small as needed for identifying material losses. The choice of MBA boundaries directly impacts diversion detection sensitivity because smaller, more tightly controlled MBAs enable more frequent material balances with reduced inventory uncertainties, while overly broad MBAs may make detection much harder with dilution of evidence. However, it should be known that the multiplication of MBAs in the day-to-day is accompanied by a huge multiplication of declarations to the inspectorate (e.g., each receipt and shipping must be notified, even between MBAs of the same facility). An engineering-based approach should provide actionable knowledge and take into consideration the real inspections' day-to-day.

MBA establishment, tied to the possible change of the NM form within the facility, poses consistency issues, as one might prefer to have the same material form throughout an MBA rather than having to take into account the transition, especially since it might affect the methodology for NMA. For instance, solid fuel salt delivered to an MSR nuclear power plant before melting can be treated similarly to UO_2 fuel pellets and counted as batch items [37], while it could be a continuous flow of molten salt during operation. In any case, items are always to be preferred to bulk, and batch is always preferable to continuous flows for safeguards purposes [72].

The previously described accumulation of uncertainty and errors, when put together with the generally large fissile inventories intrinsic to MSR designs, amplifies the effect of uncertainty in the Inventory Difference, such that even with high-precision measurements and respect of the IAEA's goals, detecting material losses on the scale of multiple SQs becomes statistically impossible in some modelled scenarios [63]. Regardless of design, large fissile inventories create a challenging environment to detect material loss using statistical testing on material accountancy data alone. Generally, detection of material loss on the order of multiple SQs is difficult in MSRs using obtainable measurement errors. When investigating the MUF (or ID), USA national safeguards evaluate the σ_{MUF} (referred to as the SEID, the Standard Error of Inventory Difference) [63]. σMUF/SEID is the standard deviation (or square root of the variance) of the inventory difference, formally incorporating measurement uncertainties, systematic and random errors from all involved measurement processes, integrated with the quantity of NM present. Mathematically, the total variance includes the summed variances of all common measurement terms contributing to the inventory difference, regardless of their sign, following standard error propagation practices. It is reminded that the σ_{MUF} depends on the uncertainty in measurements δ_E, as previously addressed in 2.5.2.2. The 1% uncertainty value, or $\delta_E = 0.01$, is consistent with the expected standard deviation in closing an MBA for Pu reprocessing [5] [25].

A table summarising the modelled σ_{MUF} for 6 MSR designs is presented below

Table 16: LLDs based on current σ_{MUF}/SEID values [63]

Design	Design Power (MWth)	Material	Average Nominal σ_{MUF} at 1.0% Uncertainty (kg)	Lower Limit of Detection at 1.0% Uncertainty (kg)
MSDR	750	Total Pu	26.11	85.65
MOSART	2400	233U	33.60	110.22
MOSART	2400	Total Pu	141.164	463.02
MSFR	3000	233U	63.20	207.29
REBUS	3700	Total Pu	242.08	794.05
MSCFR	6600	Total Pu	87.97	288.56

It is also shown that the LLD (meaning the amount of protracted material that, given 1% uncertainty, will be detected with 95% probability), derived from the previously defined formula, is dozens of times an SQ. For instance, on the MSFR, as long as the MUF is within 63.2kg of U-233 (of which, the reader is reminded, an SQ is 8kg), it is within the specific goal of the IAEA for Pu reprocessing MBAs. 207.29kg of U-233 would have to be removed for a detection within an MBA's closure. Parallelly, a mechanism for setting limits on material measurement accuracies and discrepancies will need to be developed for MSRs, with the critical concern that the volume or mass of salt needed to divert 1 SQ of NM can be as little as less than 1% of the total core salt volume, informing the accuracy requirements for safeguards instruments and inspection frequencies [37] [46]. This further demonstrates the need for improved accuracy in the estimations, but also the measurements. If measurement uncertainties do not satisfy IAEA requirements, NESs will potentially necessitate more frequent PIVs or separation of materials into smaller throughput streams [47], which could be a way of mitigating the large fissile inventory present by reducing the target of the measurement. Another opportunity for achieving this result could be a core optimisation during the early design stages that can reduce equilibrium fissile inventories [64]. **New instruments might be needed**.

3.4.4.2. DPA

The intrinsic characteristics of MSRs create new diversion paths that need to be covered, impacting the inspectorate efficiency (not necessarily the effectiveness) in designing an adapted safeguards system, and the costs to do so. It can be required that at least two independent elements be used to prevent or detect every diversion path, which impacts the efficiency of the safeguards approach establishment [63]. The creation of new diversion paths is intrinsic to the new technologies adopted, tied to the salts, salts reprocessing, or the Th-U cycle, when used. The characteristics of the fuel and the inherent non-zero MUF make partial and biased defects particularly challenging in this family of reactors. It is reminded is the removal of a small part of NM that amounts to an SQ has to be repeated and protracted in time and may or may not be accompanied by a concealment action. The detection probability then has to be achieved on a longer term than for gross defects, but a single biased defect could very easily go undetected. [63] cites a few preferable modes of diversion: *"gross diversion of a storage container of fresh fuel, protracted diversion from a container with concealment methods, diversion of SNM through a salt sampling process stream, diversion of SNM from filters, material substitution, or diversion of SNM as residual material (holdup) within equipment being replaced."* Regarding salt reprocessing, [49] states that designs that minimise or eliminate separation of fissionable species may be more proliferation resistant. This is implied by the elimination of locations in which the fissile material can be found, et cetera. However, being proliferation-resistant does not directly imply that safeguards are more effective. An instance of this is the ^{233}U recovered from SNF, which is considered "self-protecting" due to the presence of ^{232}U (and its decay chain as previously described), which could result in exposure to a fatal dose of radiation to a potential human diverter. However, the concept of self-protection is not relevant to international safeguards, especially since a truly determined rogue actor would accept the risks [49]. Still, the highly energetic gamma radiation should pose the need for remote handling, automation, and containment by design. These features would reduce the accessibility and degrees of freedom in action and making the detection (through specific signatures) and the application of C/S measures easier. Similarly, a HEU would provide a more recognisable gamma signature, making it more safeguardable (through easier identification), but it is clearly less proliferation resistant.

The form of the fuel can be beneficial from a safeguards standpoint in both a gross and partial defect event since the resulting fissile density may decrease compared to LWR fuel assemblies/rods, resulting in a large amount of salt needed for an SQ, increasing detectability.

Table 17: Inventory of potential NM in a 3GW MSFR in terms of SQs and the corresponding litres of salt [49]

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

A particularly concerning diversion path is created in designs in which the breeding of ^{233}U is achieved through the decay of ^{233}Pa outside of the neutron flux of the core (The Molten Salt Breeder Reactor – MSBR program relied on this to achieve a net breeding gain). In the case of the MSFR concept, all isotopes of Pa are quickly sent back to the core, but a pure stream of ^{233}U can be produced in other designs, which reduces the quantity of ^{232}U present and thus the radiation-associated health hazard [36]. For fuel cycles in which the protactinium isotopes are removed on a 10-day cycle and not reintroduced into the fuel (as in the MSBR design) [49], production of ^{232}U is significantly suppressed due to the removal of intermediate isotopes. The idea is to isolate Pa from the salt (fuel or fertile) by chemical processing and then let the ^{232}Pa decay into ^{232}U before separating the remaining ^{233}Pa from other isotopes. After 3 weeks of storage, the fraction of ^{233}Pa remaining is 58% but the amount of ^{232}Pa has been reduced by a factor of 4×10^4. After 4 weeks, this factor becomes 1.5×10^6, and the remaining ^{233}Pa is still 49% of the initial amount. Obtaining an SQ of ^{233}U would require the processing of about 10 m3 of fertile salt or 2.7 m3 of fuel salt (detectable quantities) [49]. Even if the decay of ^{232}Th into ^{228}Ra is in the order of 10^{10} years, not being able to separate it fully from ^{233}Pa in reprocessing would still leave traces and a potential signature [49]. There is also the potential for misuse of the reactor by modifying its fuel salt composition to produce more ^{233}U [47], for which fresh fuel checks must be implemented.

[49] identifies a diversion path through the fuel salt reprocessing. Taking advantage of the gaseous form of UF_6, after extraction of the U by fluorination and before

incorporation back to the fuel salt (UF_4). The diversion will upset the actinide balance of the fuel salt and should be readily detectable.

A particular issue for liquid fuel MSRs, with or without on-site fuel salt reprocessing, is the heel of fuel salt left on components that have been drained. Liquid fuel designs will likely employ a flush salt to reduce the amount of fuel left in replaced components. However, the flush salt itself will progressively acquire additional radionuclides and will need to be stored in containment when not in use. MSR plants will need an effective capability to transfer worn-out components into secure, cooled storage and to perform fissile material inventory as components are transferred out of containment. This issue is exacerbated in salt reprocessing units, as for pyroprocessing, one of the main issues is the inability to flush out the entire system and entirely clear the inventory [73], further building on the holdup issue.

[65] identifies potential holdup locations of a generic liquid-fuelled design that should be inspected as potential diversion path sources.

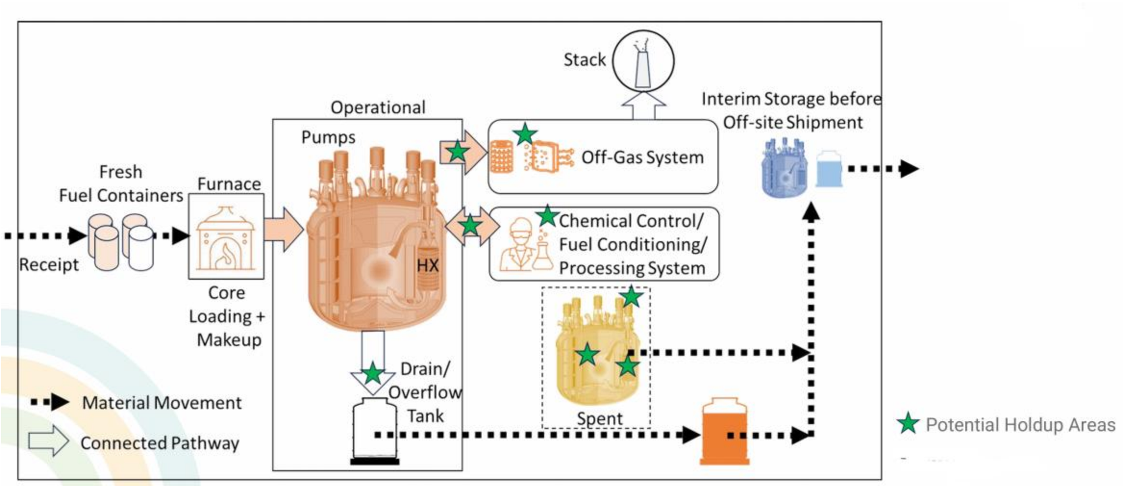


Figure 23: potential holdup locations of a generic liquid-fuelled design [65]

Liquid-fuelled MSR designs with off-gas sparge systems for noble gas and metal removal present pathways for material loss through waste streams or filter systems, with sampling ports on fuel salt loops presenting direct access to molten fuel salt and thus potential additional pathways for material theft. Generally, FPs tend to be concentrated, mainly in low-flow areas of piping. Longer-lived isotopes were identified in the off-gas piping, which indicated deposition of the noble gas parents; however, ^{95}Zr was not detected in the off-gas piping, indicating that no holdup of NM (from the fuel salt) is occurring in that location, effectively demonstrating the low-risk separation capacity. A table summarising the typical holdup values is presented below (Originally taken from [74]).

Table 18: typical holdup values in equipment [63]

Equipment	Holdup	Equipment	Holdup
Glovebox prefilters	2 to 100 g	Pipes (after destructive cleaning)	0.3 g/m
Final filters	10 to 100 g	Ducts (no cleaning)	1 to 100 g/m
Equipment interiors (after routine cleaning)	10 to 50 g/m²	Annular tanks	1 to 10 g
Furnaces	50 to 500 g	Furnaces	50 to 500 g

Several other issues arise from the temporary storage of SNM outside of the reactor core. The on-site salt processing particularly poses the potential to have fuel inventory present outside the reactor containment vessel. New components like decay tanks and storage of isolated fissile isotopes (see the MSFR schematics) are added to designs, providing new sensitive opportunities for diversion. For instance, in the MSFR, 99% of U (including ^{233}U) and Np and 90% of Pu are extracted by fluorination and stored in a temporary tank waiting for reintroduction to the core. This step is followed by actinide extraction with Pa and final reductive extraction to separate Th from lanthanides for waste disposal [49]. At each step, possibilities for misuse are present by tailoring the pyroprocessing voltages.

Additional proliferation concerns stem from the carrier salts. ^{35}Cl has a substantially higher Sulphur production cross-section than ^{37}Cl. Sulphur is extremely corrosive to engineering materials in liquid halide salts, and its presence promotes the formation of uranium sulphide, which can precipitate from the fuel salt and thus provide an additional diversion pathway compared with fluoride salts. [49] (additional design issues, such as the radiotoxicity of chloride salts due to the activation of ^{35}Cl into ^{36}Cl, which is a long-lived isotope ($t_1/_2$ = 301,000 years) that can be mobile in geologic repositories, thus allowing a possible extraction [49]. A solution to both issues is the chlorine isotopic separation with enrichment in ^{37}Cl.

Design Information Verification (DIV) and Containment and Surveillance (C/S) face particular challenges primarily due to high radiation environments that prevent traditional inspector access and require radiation-hardened equipment capable of extended operation in extreme conditions. Remote and unattended monitoring will likely be required, which will have to be developed [46]

3.4.5. Regulatory and Implementation Challenges

The regulatory acceptance of MSR modelling tools for safeguards applications presents significant challenges that extend beyond purely technical considerations. Nuclear regulatory authorities require extensive validation and verification of computational models before accepting their use in safety-critical applications. The limited experimental database for MSR validation compounds this challenge significantly [75]. Current regulatory frameworks, developed primarily for solid-fuel reactors, may not adequately address the unique modelling challenges associated with liquid-fuel systems. There are no current formalised IAEA goals for the Type I error probability for an advanced fuel cycle concept [31]. For the Th-U cycle, a sensitive aspect is the inclusion of MA (e.g., Np) and relevant isotopes into safeguardable materials. For instance, ^{233}Pa is not safeguardable, but as previously shown, it offers dangerous proliferation paths.

The safeguards technical objectives could be revised to increase the needed accuracy for the problems linked to MUF and σ_{MUF}.

The IAEA and other national and regional regulatory bodies must develop consistent approaches to model validation, uncertainty quantification, and acceptance criteria. The diversity of MSR designs under development worldwide adds complexity to this coordination challenge.

3.4.6. Summary of the Sub-Section

The need for change in paradigm is evident from the fundamental incompatibility of traditional item-based safeguards approaches with liquid-fuelled MSR characteristics. As previously stated in current safeguards practices, conventional approaches rely on item counting and containment/surveillance of discrete fuel assemblies, whereas MSRs require bulk material accountancy approaches similar to reprocessing facilities but with unique characteristics that most existing instrumentation cannot address, necessitating significant research and development efforts [62]. The paradigm shift requires a transition from periodic static inventories to continuous ones, coupled with process monitoring approaches and with sophisticated modelling and simulation tools for inventory prediction and verification. The deployment of MSR will require a Safeguards approach able to reliably detect partial defects, utilising diversion path analysis (DPA), and through the design of measurement techniques suited for high-radiation, high-temperature, corrosive environments. Instead of traditional PIV inspections, the IAEA may need to rely on technologies related to process monitoring to accurately quantify materials [46]. The peculiarities of the fuel cycles offer both challenges and opportunities in the evaluation of the NM's NDA signatures, which need to be evaluated after the inventory determination issue is resolved.

3.5. NDA Instrumentation Applicability to MSRs

The timeline for any new safeguards technology development is around 10 years from the lowest technology readiness level (TRL) to the highest (i.e., commercial adoption and field implementation) [36]. Given the expected deployment for some MSRs as early as the 2030s, and given the previously established challenges in performing

NMA on these designs (both in the non-applicability of some techniques, and the accuracy of the instrumentation tied to others), there is a real need to assess the deployability of NDA instrumentation to liquid-fuelled MSRs to direct the R&D efforts towards either incremental improvements or paradigm shifts. A preference should be given for redirecting already available techniques, rather than having to develop new instrumentation [36]. This challenge is particularly relevant for the Th-U cycle, given that the current IAEA Complementary Access Toolkit uses a detector (FLIR's identiFINDER HM-5) not capable of discriminating ^{232}Th from ^{232}U, nor identifying shielded ^{233}U [36].

[39] is provided as a reference for the matter. NDA techniques, instrumentation, and required geometry are described, which should be considered further on in Safeguards-by-Design, for adaptation of the designs' characteristics to the needs of the instrumentation (e.g., shielding and mean free path for passive gamma measurements).

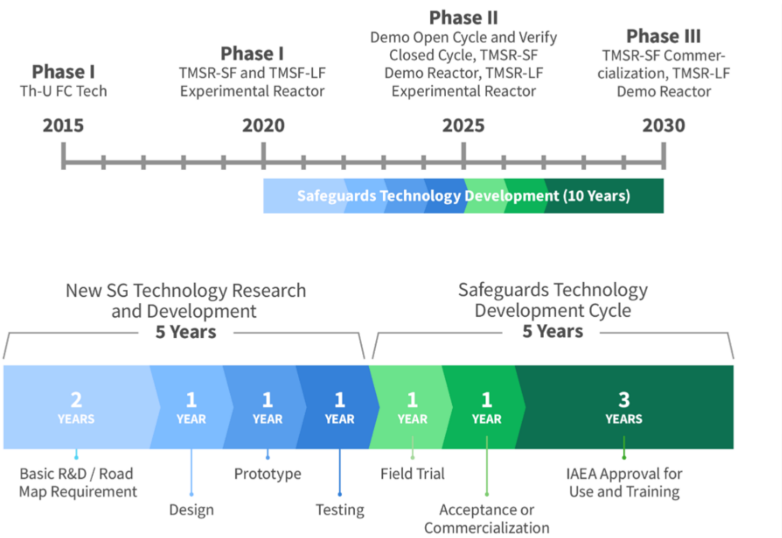


Figure 24: Th-U implementation timeline (top) vs. safeguards technology development. [36]

Table 19: TRL and MSR applicability of different NDA instrumentations, as of 2021. Adapted from [38].

Technology	Measurement	Detection Basis	TRL	Molten Salt Tested	Radioactive Salt Tested	Safeguards Utility
High-Dose Neutron Detector	Radiometric – neutron	Moderation and ^{3}He tubes	4–5	N	N	M
Online Actinide Sensor	Actinide concentration	Voltage applied to the electrode	3	Y	N	M
In Situ Actinide Monitor	Actinide mass/composition	Measurement of melting point (temperature)	2–3	N	N	M
Wide Bandgap Alpha Detector	Radiometric – alpha spectroscopy	Semiconductor charge collection	3	Y	N	L

Microcalorimeter	Radiometric – photon spectroscopy	Induced heat from incident radiation	4–5	N	N	M
Triple Bubbler	Salt density and depth	I–V curves of independent probes	2–3	Y	N	M
Voltammetry	Salt composition (molecular)	I–V curve w/ electrode	2–3	Y	N	M
High-Throughput Micro-Sampling	Direct salt sampling	Pneumatic salt droplet	2–3	Y	N	M
Hybrid K-Edge Densitometer (HKED)	Radiometric – U and Pu concentrations in samples	K-Edge transmission and X-ray fluorescence	9 (COTS)	N	N	M
Fission Chamber[7]	Radiometric – neutron	Gas-based charge measurement	9 (COTS)	N	N	M
In-core, Ex-core, and self-powered Neutron Detectors[9]	Radiometric – neutron	Gas-based charge measurement	9 (COTS)	N	N	M

HKED is to be adapted from reprocessing. Fission chambers and neutron detectors from LWRs' safeguards technology.

3.6. Proposed approaches

The international community consensus is that no fit-for-all detailed solutions that apply to every liquid-fuelled MSR design can be conceived [37] [47] [63].

In identifying the right safeguards approach for a given design, two key processes should be done:

- Early in the project, identify the high-level NMA objectives across the facility that would be necessary to prevent or detect diversion of material.
- Perform a DPA to identify potential specific paths of diversion from process streams [63]

Given the difficulties in mass/volume determination, uncertainty in models, and inability to detect diversion from statistical methods only, [63] states that implementing a traditional inventory-determination-based NMA system, whether adapted to bulk-handling facilities or not, in liquid-fuelled MSRs *"requires a high level of resources devoted to MC&A and is, in the author's opinion, both inefficient and not necessary to prevent or detect diversion, which is the overall goal of an MC&A program"*. This point of view builds on the temporary necessity of radically changing the safeguards approach, but it should be taken carefully, as it might be strongly biased by the US national (domestic) safeguards perspective, which is that of an NWS.

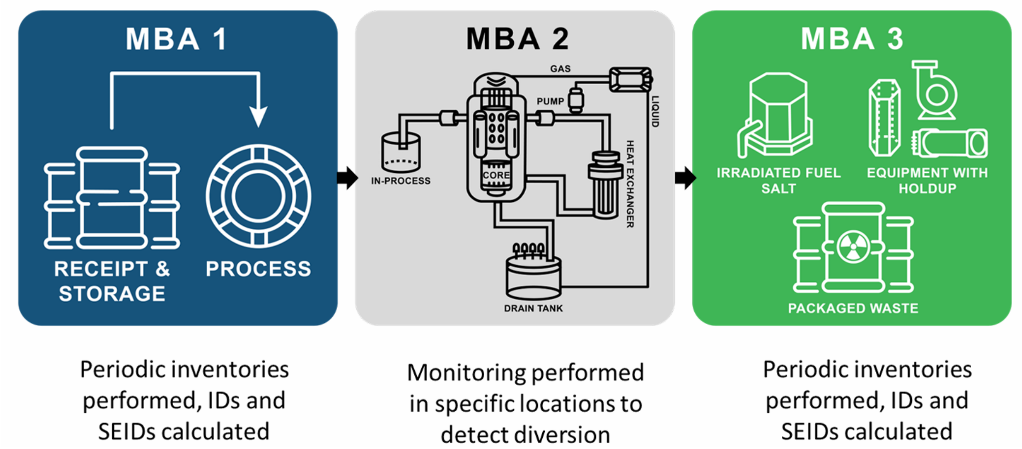


Figure 25: A proposed MBA structure for a terrestrial-based liquid-fuelled MSR [63]

The Inventory-Monitor Containment-Inventory (IMCI) acknowledges the dual nature of both nuclear reactors and bulk material facilities. This methodology strategically divides reactor facilities into multiple MBAs, with different accountability measures applied based on material accessibility, form, and radiological conditions [63]. Under the IMCI framework, periodic inventories are conducted on accessible NM located outside reactor confinement areas. This approach is defined to circumnavigate the problem of inventory definition in the reactor, as long as uncertainties are not reduced. Periodic inventories would not be performed on NM in areas in which the inventory is not well-determined, and where high-radiation and high-temperatures make measurements impractical. The approach is to consider the reactor and primary loop, in which the depletion and most transmutations happen, as black boxes for which only inputs and outputs are measured. In these confinement zones, Safeguards measures rely primarily on C/S. The boundaries would be immediately before (or after) the points in the process flow where the NM changes form [63].

For a liquid-fuelled MSR with online salt reprocessing, the proposed MBA structure for liquid-fuelled MSRs encompasses three distinct zones, each with distinct NMAC approaches. MBA1 handles fresh fuel salt storage. The NM can be subjected to traditional accounting methods (batches can be item counted for MSRs without fuel fabrication processes in the facility [64] [63]. The NM is received under seal, measured, and the batch weights are recorded and compared with shipper declarations. Finally, the seals are verified for tampering indications. The second area constitutes reactor confinement, where irradiated fuel salt becomes highly radioactive and largely inaccessible during operations. The holdup is considered an output. The third area manages irradiated fuel salt and waste storage, where materials leaving the reactor system require quantification through NDA measurement techniques. MBA 1 and 3 could potentially be combined into one MBA if the NM is fed and withdrawn from the same access point [64]. This approach recognises the practical impossibility of performing PIVs within operating reactor systems where NM inventories undergo continuous transmutation and depletion [63]. Table 3 of [63] provides an overview of the expected safeguarded material to be found and measures to be taken in the 3 MBAs.

Building on this black-box approach, [76] proposes to calculate the inputs and outputs through the measurements of the flow rate (m3/s) and total time of flow (△t). Along with the quantification of mass flow in and out of the reactor, burnup monitoring through the detection of the passive neutron emissions from Cm-242 and Cm-244 is proposed to determine fissile content in the flowing fuel salt at every KMP. To measure the fuel burnup, counting the passive neutron emissions of specific elements was considered. HKED is suggested to complement the burnup monitoring and can increase the accuracy of nuclear safeguards by simultaneously measuring the presence and concentration of the fissile isotopes.

Another proposed approach is conceived on the radiological environment issue, in addition to avoiding the inventory definition issue. Because of the radiation conditions within any proposed MSR design, an MBA can be defined to encompass the entire facility, thus avoiding the problem of a small MBA made of just the confinement zones, in which measurements and an accurate inventory definition are complicated. The safeguards approach would then heavily rely on C/S, but IAEA objectives must still be respected [38].

3.7. Summary

To sum up, MSRs are an emerging technology with an established proof-of-concept from early experiments conducted in the USA in the 20th century. Because of the several technical features of MSRs, they are a promising technology for exploration in the 4th generation of NESs. Primarily, the increased safety and reduction in waste are appealing, also to improve the public’s perception about nuclear energy, perhaps the main problem to the adoption of nuclear energy worldwide. Non-energy applications and increased efficiencies are appealing in the context of the Energy crisis and climate change. International bodies such as the IAEA and the GIF are studying MSRs' designs and creating taxonomies to start the regulatory process, by grouping the MSRs into families, based on their technical features and the fuel cycle peculiarities. MSR technology couples compelling performance benefits with distinctive fuel-cycle behaviours that fundamentally reshape how nuclear material moves, changes, and is measured during operation, which in turn reframes safeguard priorities from static inventories to dynamic, process-aware verification. No fit-for-all safeguards solution exists for MSRs, given the vast differences between designs and reactor families. In some designs, the continuously circulating liquid fuel and potential online processing create measurement and modelling uncertainties that challenge conventional signature interpretation, material verification, and MBAs’ closures. This is particularly the case for the designs employing the Th-U cycle. These technical struggles motivate targeted advances in neutron/gamma NDA, transport modelling, and process monitoring R&D in the Safeguards, to establish an approach capable of maintaining accuracy and timeliness despite the evolving isotopics and chemistry. Accurate modelling and validation are perhaps the most urgent challenge to MSR safeguards R&D. The MBAs will have to be defined such that even biased and partial defects can be statistically detected, and a thorough DPA needs to be performed on designs to address the new proliferation paths stemming from the technical peculiarities of MSRs. The proposed approach found in literature heavily relies on C/S rather than NMA, considering the reactor confinement (and auxiliaries) as a black-box of which only the inputs and outputs are measured. Experimental data will have to be gathered to reduce the underlying uncertainties in the models. However, these approaches should be carefully considered in the context of international safeguards. The significant needed change throughout the whole safeguards approach needs to be addressed as early as possible, at a moment of maximum leverage and freedom of action, rather than hoping for post hoc retrofits. Consequently, it follows that Safeguards must be embedded in the design considerations, so that regulatory assurance scales with MSR innovation instead of trailing it [47].

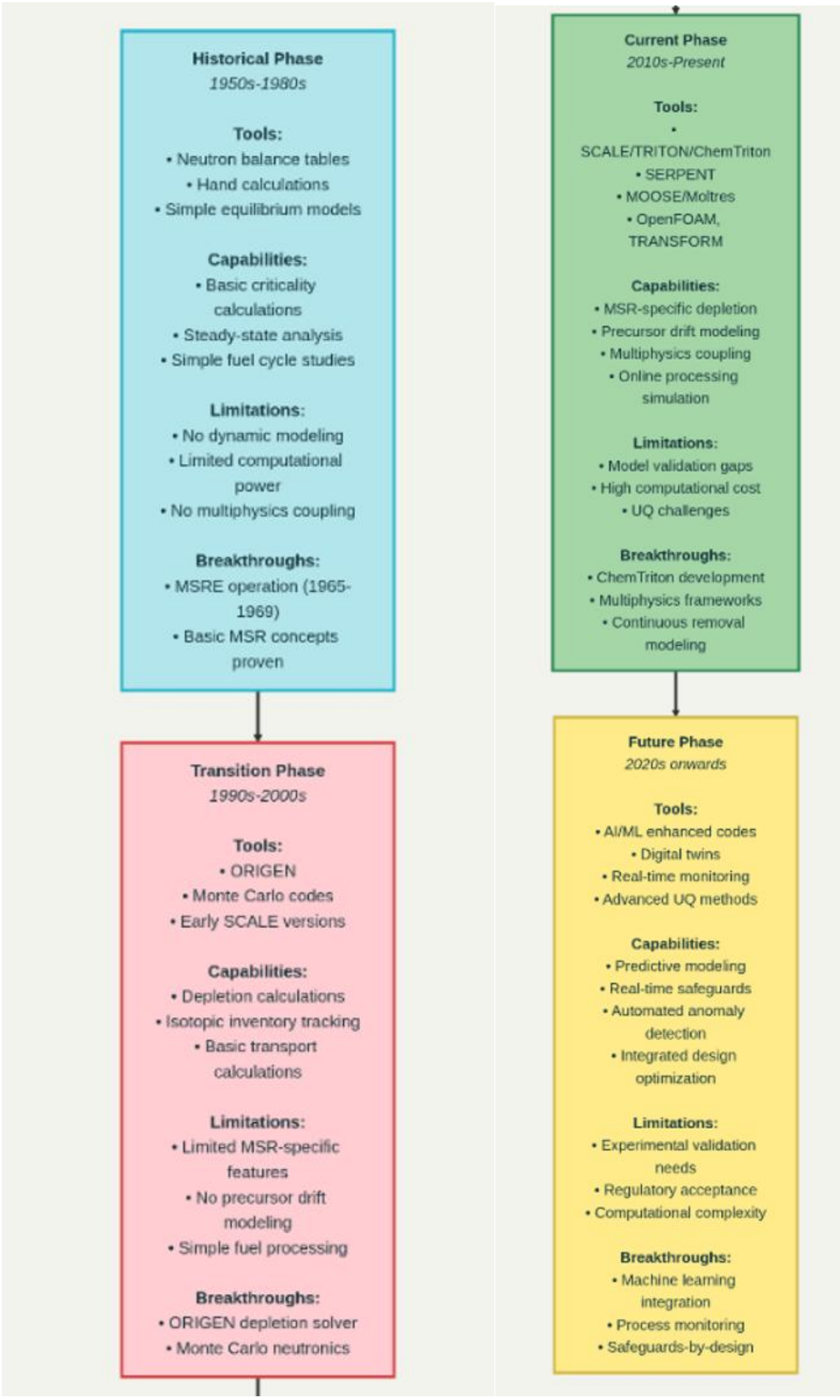


Figure 26: Phases and available tools of MSR designs and safeguards

4 Safeguardability of (liquid-fuelled) MSRs through the Development of an Assessment Tool

Even if slowed by the public's perception of the dangers associated with it (e.g. Germany's ruling out of NES usage), Nuclear Energy is undergoing a recognisable expansion as one of the most stable, safe, and carbon-emission friendly sources of Energy. Nearly 70 reactors are currently under construction worldwide. The IAEA keeps rising its nuclear power projections, forecasting that global capacity could more than double by 2050. Moreover, with the advent of Small Modular Reactors (SMRs) and Advanced Modular Reactors (AMRs), the number of reactor units worldwide is set to grow exponentially. Many designs of SMRs are conceived to be deployed in different-than-usual ways (both in terms of siting and energy production), some of them are meant to be transportable (either land or marine based). The new systems will have innovative fuel cycles, material flows, and operational characteristics. **The combination of increasing reactor numbers, greater technological heterogeneity, and limited inspectorate resources creates a challenging environment for the Safeguards regime**. The traditional safeguards approach based on adapting verification systems to mature designs seem to be not-suited to innovative designs such as MSRs [77]. The most effective and efficient path forward appears to be an integration of safeguards needs into the design at the earliest possible stage, and in any case before a high level of design maturity is reached.

4.1. Safeguards-by-Design, Safeguardability, and alternative views on the Safeguards implementation

The traditional approach to implementing international nuclear safeguards has been "reactive", with safeguards requirements addressed late in facility design or even after construction completion. This retrospective approach has led to numerous challenges, including costly retrofits, project delays, operational inefficiencies, and suboptimal safeguards effectiveness. Historical experience demonstrates that addressing safeguards requirements after facility construction can significantly exceed the cost of early integration [33], with retrofit costs sometimes reaching several percentage points of total capital investment compared to 0.1% when properly integrated from the beginning. [62, 78, 33, 79]. This current approach has been sustainable only given the limited design variations worldwide, since the vast majority of reactors are LWR. This is not going to be sustainable in the future, as previously established: only regarding MSRs, no fit-for-all approach could ever be conceived. **Many of the NES design features are necessary and intrinsic to the design itself. However, some of them could potentially be altered to incorporate safeguards and achieve synergies, resulting in a better effectiveness of the safeguards system or better efficiency in establishing the safeguards approach**. For example, fresh fuel inventory could be limited or stored in one secure location that promotes easier inventorying by the IAEA [63]. Several complications might arise tied to the new technology needs. For instance, the possibility of applying remote monitoring is not only an IT issue, but accommodations must be made, such as perforations into the reactor confinement. These accommodations can be extremely expensive and challenging, or even impossible, when done after construction. Another instance is the characterisation of holdup in MSRs' confined areas. Designers need to plan the inclusion of holdup measuring devices "now" [65].

4.1.1. Safeguards-by-Design

The inclusion of safeguards considerations in the design of a NES is addressed by the Safeguards-by-Design (SBD) process. The IAEA defines SBD as "***the process of including the consideration of international safeguards throughout all phases of a nuclear facility project, from the initial conceptual design to facility construction and into operations, including design modifications and decommissioning***" [62]

4.1.1.1. The "By Design" Concept

The "by design" concept encompasses two complementary meanings. First, it signifies incorporating safeguards considerations into the technical design of the facility itself: the physical layouts, process flows, instrumentation, and infrastructure that will exist in the constructed facility. Second, it means incorporating safeguards into the management of the design and construction project. This dual meaning is essential because **successful SBD requires both appropriate technical solutions and effective project management**. Technical excellence in the safeguards design cannot compensate for poor project coordination that leads to late consideration of requirements. Conversely, project management cannot overcome fundamental technical limitations in the safeguards instrumentation. **The by-design concept is not a new legal requirement for designers**, but undergoing an SBD process could be a market winner when operators buy a design. The inspectorate has the legal obligation to safeguard a facility, at its own expenses, but **the facility cannot be operated until any retrofit is done**, which poses a potentially significant delay. On the operators' side, they already have obligations under their safeguards agreements to provide design information, permit facility verification, and facilitate safeguards implementation. **SBD provides a structured process for meeting these existing obligations efficiently** [78].

4.1.1.2. Fundamental Objectives of the SBD process

Safeguards by Design pursues objectives that benefit all stakeholders in nuclear facility development [80]:

- **Facilities are designed so that the diversion of nuclear material is technically difficult and easier to detect**: Beyond facilitating verification activities, SBD should facilitate implementation of safeguards and the inclusion of intrinsic barriers to proliferation. These include process characteristics that make material diversion more difficult, facility layouts that enhance C/S, and design features that facilitate the detection of discrepancies and anomalies through the transparency of operations.
- D**esign new civil nuclear facilities that meet national and international nuclear safeguards requirements and thus avoid costly and time-consuming retrofits**. SBD eliminates these risks by resolving potential issues during design phases when changes are least expensive, by complying with the two objectives above [80].
- **Make safeguards implementation more effective and efficient.**

Designers are most familiar with nuclear safety, and while the implementation of SBD appears logical and compelling for the points noted, the lack of experience, the complexity of the NES projects, and the complicated dialogue with the stakeholders are forcing the NES designers to face fundamental questions that should be addressed very early in the design process, formulated in [33] as follows: "

- *What are the relevant safeguards requirements?*
- *What are the risks to the project of not meeting these requirements?*
- *What are the potential nuclear material diversion paths and facility misuse scenarios for the facility and/or process?*
- *What are the barriers that prevent the diversion of nuclear material and/or misuse of the facility for the undeclared production of nuclear material?*
- *What is the design of the safeguards system to support domestic, as well as international Safeguards?*
- *How can the facility and/or process be designed to enhance the barriers to diversion and facility misuse?*
- *If the facility design is altered to minimise the risk of nuclear material diversion, what are the associated trade-offs, in terms of reduction in operating efficiency or increased cost?*
- *Can the facility design be optimised to meet the objectives of the facility designer, owner/operator, national nuclear regulator, and the IAEA simultaneously?"*

The latter objective of SBD refers to the dual-action needed in the process. Not only the designers need to adapt as much as possible their designs to the existing safeguards approaches, but the inspectorates need to start early the development of new safeguards approaches and techniques to effectively put under safeguards a new plant. Refer to Figure 24: Th-U implementation timeline (top) vs. safeguards technology development.

In summary, **the SBD process fosters dialogue among the various stakeholders involved to achieve shared objectives that complement each other**. The inspectorate needs an effective safeguards approach and wants to be efficient by minimising the efforts to achieve effectiveness. For the designer, collaborating with the inspectorate in achieving these goals will result in no retrofits and no delays in the NES project.

4.1.1.3. SBD and the Project's Phases

Each facility project goes through a series of phases before operations (whether in the nuclear industry or not). Understanding how the SBD process should be integrated in these phases is paramount to successful SBD implementation.

Figure 27: Sequence of Design and Construction Stages for a typical project [78]

Table 20: Description of the maturity of design and the corresponding uncertainty of judgement. Adapted from [5].

Stage of development of a NES	Level of maturity of an INS (or component)	Level of uncertainty of judgement
No theoretical or experimental showstoppers.	Pre-conceptual	Very High.
Most important (not all) components have been theoretically demonstrated or experimentally verified.	Conceptual feasibility established.	High.
All components of the INS have been theoretically demonstrated and, where necessary, experimentally verified.	Feasibility demonstrated.	Moderate.
All components of the INS have been designed in enough detail to prepare a bid. If needed, a pilot plant (reduced size) was built and is operating successfully.	Developed and demonstrated.	Low.
First-of-a-kind plant (full size) built and operating.	Commercially proven.	Lower.

A series of plants was built and operated.	Full commercial exploitation.	Lowest.

The **pre-conceptual stage** typically represents 8% of total project cost, yet decisions made during this brief period commit approximately 70% of eventual project costs. This phase is crucial for project management decision-making. Decisions taken in this stage have severe repercussions and dependencies throughout the project, and thus, the early and active inclusion of Safeguards experts in this phase is vital to ensure that safeguards requirements are embedded in system specifications [78].

Analysis at this stage should consider the use of proven safeguards measures (by the inspectorate), requirements for new measure development (by the inspectorate and policymakers), and potential for intrinsic design features to enhance the compatibility with safeguards (in a dialogue between the inspector and the designer). For NWSs, the state has to choose whether the facility will be placed on the Eligible Facilities List for potential IAEA inspection. For NNWS, a notification to the IAEA of the intention to construct the facility is required [78].

In the **Conceptual Design Stage**, critical design documents are created by the designer (such as the Process Flow Diagrams, Piping and Instrumentation Drawings, and facility layout), so that the inspector can use them to design the safeguards requirements and approach. Safeguards-specific documentation includes the Design Information Questionnaire (DIQ) and the draft Facility Attachment (FA) [78]. Preliminary definition of MBAs and KMPs can be done so that provisions can be made by the designers to include specific established safeguards necessities (e.g. wiring).

Inspectors' evaluation of the safeguards approach is necessary for understanding whether it would meet the safeguards goals, even at this early stage. The established safeguards goals should be respected (see 2.3), but the NM diversion should also be mitigated through design improvements [33]. The evaluation is performed through tools such as DPA to identify potential vulnerabilities and compare the effectiveness of alternative measures. For complex or novel facilities, the Facility Attachment may continue to evolve during subsequent design stages. However, the industry strongly emphasises the need for FA finalisation no later than the end of preliminary design, since this is when facility layout and infrastructure are frozen for detailed design. Industry input indicates this finalisation should occur during preliminary design: continued uncertainty into final design creates unacceptable project risk [78]. See Table 20.

The **Preliminary and Final Design Stage** encompasses development of detailed engineering specifications and construction packages. During preliminary design, the facility is adapted to site-specific requirements and detailed design criteria are established [78]. It is reminded that inspectors maintain the sole responsibility for the safeguards approach establishment and thus provide a detailed specification of safeguards infrastructure requirements, including physical space for equipment, electrical power (both mains and backup), data transmission capabilities, cable routings, and penetrations for equipment installation. Operators (or designers) develop operational procedures to facilitate the safeguards approach in accordance with the inspector's need, including procedures for PIT, interim verifications, inspector access, and equipment operation [80, 78].

The **Construction Stage** involves actual facility building according to specifications, with ongoing monitoring to ensure as-built condition matches the design intent. Safeguards activities during construction include:

- Installation of domestic MC&A and PP equipment required by national regulations or facility operations, as well as accommodations to include future IAEA (or Euratom) equipment.
- IAEA's DIV activities, including inspection and photography during key construction phases (such as concrete pours) to confirm the facility is being built as declared.
- Assessment of design changes occurring during construction to ensure safeguards performance is not compromised [78].
- Testing and commissioning of safeguards equipment before nuclear material is introduced to the facility [62, 78]

The Operational Stage begins when the facility receives its operating license and introduces nuclear material. The first nuclear material provides an opportunity to calibrate and test safeguards equipment under actual operating conditions. SBD benefits become fully realised during operations through:

- Implementation of the safeguards approach developed during design, without the need for operational disruptions due to inadequate safeguardability
- Efficient verification activities due to appropriate infrastructure, clear access paths, optimised measurement points, and compatible inspection procedures
- Reduced operator burden because safeguards, equipment, and procedures were integrated with operational systems rather than retrofitted [78]
- Flexibility to accommodate evolving safeguards techniques through infrastructure designed with margin for future equipment changes [80, 62]

The Decommissioning Stage occurs when the facility ceases operations. Safeguards obligations continue until all nuclear material is removed and the IAEA confirms termination conditions are met. SBD considerations for decommissioning include the design features facilitating eventual material removal and facility cleanup [62].

A summarising table of the SBD activities to perform in alignment with the design phases is presented below.

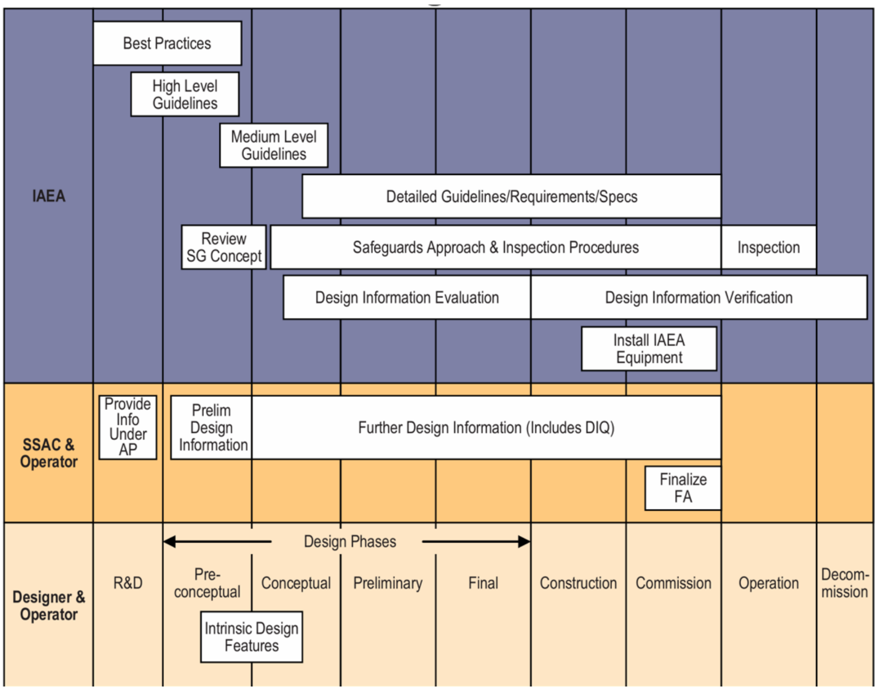


Figure 28: Summary of Safeguards activities timelines for the different Safeguards stakeholders [77]

4.1.1.4. Safeguardability and Safeguardability-by-design

The safeguardability of a facility is a qualitative measure of *"the degree of ease with which a nuclear energy system can be effectively and efficiently placed under international safeguards"* [33]. It is important to note that safeguardability doesn't have a broadly accepted quantitative score: its qualitative nature cannot be readily estimated numerically. There is however a relation between the qualitative approach posed by the safeguardability concept and the quantitative estimation of the efficiency and effectiveness of a safeguards system, of which the evaluation metrics have been defined by [49] (as previously mentioned in 1.3, DP is an index of the effectiveness of an implemented Safeguards approach, DE the efficiency of the resource utilisation). [16] states that the goal of a safeguardability assessment is *"To provide an alternative to the quantitative estimation of the DP and DE measures where not enough information about the implemented Safeguards approach is available; To provide system designers with a list of attributes to be taken into account at very early design stages in order to facilitate the implementation of International Safeguards"*. Therefore, safeguardability should be tackled at the very early phases of an SBD process, and as [73] points out, together with the establishment of the legal basis.

Safeguardability is operable in all aspects of Safeguards, with features increasing the efficiency or effectiveness of the safeguards approach in these domains. The safeguardability attributes can be seen grouped in three categories corresponding to three main aspects of safeguards (DIV, NMA, C/S), and through several phases. A Safeguardability analysis is usually an important part of a SBD process, and thus it would be safe to speak of Safeguardability-by-design. This refers to the specific implementation of SBD aspects that increase the ease of establishing and operating the safeguards system. The Safeguardability phases and project phases are not necessarily connected, and a safeguardability phase can act across design phases. The three phases are defined in [81] and described as:

- **First phase**: basic design features intrinsic to the process, defined in the pre-conceptual phase (with PR and safeguards relevance)
- **Second phase**: Design features related to safeguards equipment installation, linked to the actual planning and layout of the facility, and definition of the safeguards model. The interaction between the stakeholders is crucial in this phase. Typical choices relate to cabling, MBAs/KMPs establishment, PIT, access for inspections, transfer routes, and remote automated handling. An early discussion on the instrumentation to be installed later and its requirements should be done in this phase.
- **Third phase**: Design features related to safeguards efficiency and effectiveness. The safeguards equipment is installed. The nuclear material's level of radioactivity, or signature, chosen in the first layer influences the choice of the detector type and its efficiency and effectiveness. Joint-use equipment and remote data transmission should also be considered [81]

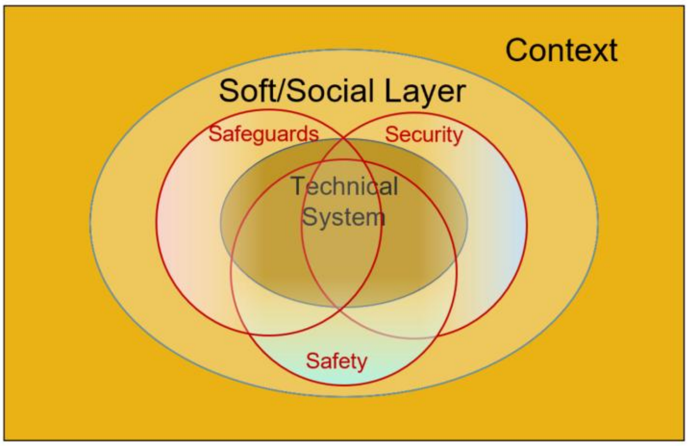
4.1.1.5. 3S and 3S-by-design

Safeguards by Design operates within a broader context of integrated facility design that simultaneously addresses safety, security, and safeguards, namely "3S". The IAEA Safety Standards [82] establish that "*safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another*". This integration offers multiple benefits. Safety, security, and safeguards share common design considerations, including access control, monitoring systems, material accountancy, and facility layout. Synergies exist where a single design feature can serve multiple objectives. For example, robust containment serves both safety and safeguards purposes. However, potential conflicts must also be recognised and resolved, such as when security measures that limit physical access might constrain inspector verification activities [78, 33].

The successful integration requires that safeguards, safety, and security experts collaborate from the earliest design stages, participate in systems engineering processes, and serve on configuration control boards to ensure that design changes preserve compliance with all requirements.

For achieving a successful implementation of the 3S-by-design concept, a project management paradigm shift is necessary.

The Doing-It-Differently System Thinking approach aims at rationalising this management approach by recognising that the 3S domains are deeply interdependent subsystems within a single socio-technical whole rather than isolated system requirements. Nuclear facilities represent complex systems embedded in a social "soft" layer provided by the broader society context (e.g. public's mistrust in nuclear energy). This is perfectly aligned with a Safeguards vision that must recognise the political aspect of Safeguards, and not only the technical requirements. In this soft layer, human operators, technical infrastructure, organisational policies, regulatory frameworks, and physical materials interact continuously.



Framework adapted from:
BLOCKLEY, David I.; GODFREY, Patrick. ***Doing it differently: Systems for rethinking construction***. Thomas Telford, 2000.

Figure 29: Visual representation of the interaction between the 3S in their Soft layer and Context [83]

Research and early application on SMRs highlight that 3S integration enables early identification of leverage points that simultaneously reinforce safety, minimise security vulnerabilities, and support robust safeguards. This can mitigate threats more effectively than isolated measures.

"2x2S" refers to the safety/security and security/safeguards linkages, or the absence of a well-defined (or necessary) safety/safeguards interface. [67]

The similarity between these visions of a more effective integration of the 3S at a very early stage in the design process further emphasises the importance of taking this path forward.

4.2. DIQ MSR (deliverable 1)

The Design Information Questionnaire (DIQ) is a legal document that the designer or operator who wants to license a NES must fill out to inform the inspectorate of the design's characteristics, so it to grasp the relevant safeguards parameters that impact the measures to implement. It is through the DIQ that the inspectorate collects the information that it is going to verify during the DIV. In terms of the Euratom Safeguards, designers must submit the Basic Technical Characteristics (BTC) to DG Ener for verification (The goals and roles of the two entities are perfectly aligned).

The BTC is conceived as facility-specific, so each facility in the fuel cycle has a suited BTC. For power reactors, the current BTC is tailored to item-based reactors for which the fuel is constrained in assemblies (Annex IA of Commission regulation n° 302-2005 [84]). Looking at the SBD and the Project's Phases, the time at which the BTC/DIQ should be addressed is exactly situated at the current design phase for most MSRs, between the conceptual and pre-conceptual phase. However, it is important to adapt the DIQ to the characteristics of bulk handling power reactors, but in particuler to liquid-fuelled MSRs. For these purposes, attributes from the BTC of bulk handling facilities (Annex IC of [84]) have been incorporated into the BTC of power reactors (coloured in blue). Other attributes had to be modified to consider the characteristics of the liquid fuel (coloured in yellow, e.g. the unit for the fuel item had to be changed to kg, L, or volumetric mass). A safeguards approach could be designed with less information that what is asked in this DIQ, which would, however, be insufficient for a safeguardability analysis.

In the ENDURANCE project's consisting in the evaluation of PR and safegurdability, the policymakers from the European Commission seek to accompany the designers in the licensing process, and the creation and the communication of the adapted BTC to the partners works towards the obtention of the design-specific information and fulfilment of the phase 1 project goals.

The adapted BTC can be found in Appendix C.

4.3. Safeguardability Assessment

4.3.1. Evaluation of the SBD's Performance

After having established the need for implementing an SBD process, if not a 3S-by-design one, having put forward the reasons why there is a need for a paradigm change, and after having provided a tool for collecting the information for a safeguardability assessment, the problem of evaluating the performance of the process emerges. While SBD can be evaluated mainly through project management evaluation factors, in respecting the timelines and favouring the talks between the stakeholders, safeguardability is a much more peculiar aspect to assess. Safeguardability is evaluated through the design of the nuclear installation, the nuclear material involved, the process and the way the installation is operated, and the State's commitments [16], aligned with the Fundamental Objectives of the SBD process. The evaluation of the safeguardability allows for capturing a tangible result of the SBD process, and high safeguardability will translate into a high effectiveness of the safeguards system [16].

The qualitative nature of evaluating the "ease" represents the subjective expert opinion, and the challenge of the quantitative evaluation of the safeguardability of a design is emerging, in the scope of evaluating the performance and success of the SBD process. Even though safeguardability is qualitative by nature, a quantitative representative is more actionable. It allows for better identification of gaps in performance, progress, and last but not least, it enhances communication between the stakeholders.

A literature review is necessary to extract a model to evaluate the safeguardability of innovative NES. The evaluation of the MSRs from a Safeguards' perspective is incredibly challenging due to the variety, but also the incompleteness of the available information in this early design phase (which is one of the purposes of doing a safeguardability assessment rather than the evaluation of the safeguards approach through the calculation of a DP, as described in Safeguardability and Safeguardability-by-design). Identifying the gaps in knowledge and needed R&D for a satisfactory safeguards approach is paramount for a double-sided safeguardability approach, to provide actionable knowledge to both the designers and the inspectorate. The identified model should have a demonstrated applicability and discerning power between the various designs. This is aligned with the goals of phase 1 of ENDURANCE:

- Compile a comprehensive list of publications, reports, and guidelines from reputable sources such as the IAEA, GIF, and nuclear research institutions
- Analyse the general considerations of proliferation resistance and safeguardability in relation to reactor design and operation, and identify those that might be of interest for MSRs

4.3.2 Literature review of preexisting models

4.3.2.1 International Safeguards in the Design of Nuclear Reactors, IAEA

Appendix 4 of [62] defines the safeguardability details to take into account by the designer for consideration. These parameters are useful when considering very practical implications of the implementation of an efficient and effective safeguards system, such as considering the need for wiring when designing the reactor confinement, which can hardly be modified afterwards. 54 safeguards details are listed. [33] calls these parameters "infrastructure guidelines" in its facility safeguardability assessment tool, acknowledging the need for including them, but recognising that these details are of a level of detail and a maturity of design too deep for an early application in the SBD process, and of day-to-day use of the safeguards approach rather than a broader first-time implementation. This same view is shared in this thesis, as many attributes are deemed irrelevant to the sought level of detail and are not retained.

4.3.2.2 GIF Molten Salt Reactor PR&PP White Paper [49]

The document is the main reference in the evaluation of PR&PP for liquid-fuelled MSRs. It provides, based on [22], 17 PR design features that intrinsically enhance the PR or facilitate the safeguards process in innovative NES. These parameters are tailored to MSRs. Many of these parameters are included in the INPRO; however, they differ slightly. These parameters include:

- Features reducing the attractiveness of the technology for nuclear weapons programmes (e.g., enrichment needed)
- Features preventing or inhibiting diversion of nuclear material (accessibility of NM)
- Features preventing or inhibiting the undeclared production of direct-use material
- Features facilitating verification, including keeping the continuity of knowledge

The parameters that are not strictly PR and can be reworked to obtain a safeguardability parameter are to be considered for the safeguardability assessment.

4.3.2.3 INPRO/PRADA/PROSA, IAEA [5, 85, 14]

Previously described in 1.3), INPRO provides a holistic sustainability assessment methodology. It is used as a national self-assessment (the preferred user is a State) and designer aid to test whether a proposed system meets User Requirements and Criteria with acceptance limits, identifying gaps early for cost-effective improvements. The methodology applies a structured checklist- and indicator-driven process (logical and numerical indicators) to verify whether Basic Principles and User Requirements are met, including specific requirements for safeguards, facilitation and proliferation resistance. The attributes are evaluated on a semi-quantitative basis, from very weak (VW) to very strong (VS), which would be easily translated into an approximate numerical performance.

The INPRO Collaborative Project “Proliferation Resistance: Acquisition/Diversion Pathway Analysis” (PRADA) was initiated to deepen the INPRO PR area by developing practical methods to identify and analyse high-level acquisition/diversion paths. It is used to perform pathway identification and barrier evaluation for specific fuel cycles or scenarios, to understand how intrinsic and extrinsic barriers combine and where vulnerabilities lie. The evaluation lies in the multiplicity and robustness of barriers using logic trees and qualitative/quantitative analyses tied to safeguards goals. Preferred users are technical assessors and designers.

The INPRO "Proliferation Resistance and Safeguardability Assessment" (PROSA) initiative provides simplified tools for national self-assessors to evaluate whether a NES aligns with INPRO's basic principle for PR and facilitates effective and efficient IAEA safeguards. The process has different steps that each evaluate a different aspect of the NES. Different questions are asked if the NES is with "proven technology", "country embarking on nuclear power with proven technology", or "experienced country with novel technology". The different steps are the evaluation of the material quality, the legal basis, the facilitation of safeguards, a DPA to assess the coverage and robustness, and a summary of findings [85]. A further development should acknowledge the need for obtaining this information.

4.3.2.4 Facility Safeguardability Analysis in Support of Safeguards-by-Design, INL [80], its overview [79] and Trial Application to the Nuscale SMR Design, PNNL [33]

The approach proposes that in the pre-conceptual phase, the designer themselves perform the FSA and provide a document with the assessment of safeguard relevant features and solutions. The process is based on a DPA and the performance comparison with the safeguards approach for an existing safeguarded facility. For new designs, the advice is to analyse each MBA**. For radically new technologies, it seems impossible that the designer proposes a safeguards system**. The applicability to liquid-fuelled MSRs is thus limited, but the model can be useful in building a general tool as a starting point.

As stated in the document, the goals of the FSA are: "

- *Compare and evaluate the effectiveness of nuclear safeguards measures*
- *Optimise the designer's proposed safeguards approach*
- *Objectively and analytically evaluate safeguardability*
- *Design the safeguards system (inspector)*
- *Compare, evaluate, and optimise barriers within the facility and process design*
- *Systematically evaluate cost and operating efficiency trade-offs between variations in the safeguards system and facility design*
- *Provide a documented paper trail and foundation for the national nuclear regulator and IAEA to confirm that the proposed safeguards system and facility design will meet national regulations and international safeguards requirements."* [80]

Regarding efficiency, evaluating the cost of implementation of safeguards vs. design modification seems relevant and a complicated task for an inspector, pointing out the need for cooperation, and a crucial step to include for a double-sided comprehensive safeguardability assessment tool.

The methodology is reviewed in [79], which stipulates the screening questions. The IAEA republished these screening questions in Annexe III of [62], for "Identifying safeguardability issues.

PNNL applies the FSA approach to a NuScale reactor, which is a variation of the PWR design. The document summarises lessons learned and suggestions. This application puts forward the limit in the application of the FSA to radically new facilities, for which no comparison facility is available.

Appendix C of the document evaluates the safeguardability of the new facility in comparison to the existing safeguard system for a safeguarded PWR facility. The result of the comparison is a "more than" or "less than" evaluation. However, the process is relevant for identifying critical points in the design. Appendix C focuses on the effectiveness of the safeguards approach.

Appendix D revolves around the "infrastructure functional guidelines to ensure safeguardability" based on [62], and is related to efficiency.

4.3.2.5 Evaluation Methodology for Proliferation Resistance and Physical Protection of Generation IV Nuclear Energy Systems [17]

Defines different parameters to evaluate PR, of which safeguards are considered a subset. A particular focus on safeguardability is done, and evaluation parameters are established:

- Detection probability: a detection probability is translated into a qualitative assessment from very low to very high
- Detection resource efficiency measured in inspector time
- Safeguards cost, in terms of capital needed, for which a percentage of the national budget could be an evaluation parameter

Table 21: Summary of the Characteristics for the DP measure [16]

Characteristic	Description
Definition	Cumulative probability and confidence level for detection of a pathway segment
Typical attributes to be considered for estimation	Attributes important to design information verification Transparency of layout Possibility to verify changes in design information during operation Possibility to use 3-d scenario reconstruction models Possibility to have visual access to equipment while operational Comprehensiveness of facility documentation and data Attributes important to nuclear material accounting Uniqueness of material signature Hardness of radiation signature Possibility of applying passive measurement methods Possibility of applying unattended NDA systems and remote data transmission Item/bulk Throughput rate Batch/continuous process Nuclear material heat generation rate Attributes important to containment and surveillance Operational practice Extent of automation Standardization of items in transfer Possibility to apply visual monitoring Possibility to apply surveillance devices and remote monitoring Number of possible transfer routes for items in transit
Example metric	Probability that safeguards will detect diversion or misuse during the execution of a segment /pathway.
Segments-to-pathway aggregation method	Calculate the probability of pathway detection on the basis of the segments involved. (e.g. the probability of pathway detection will be $P(d) = 1 - P(nd)$, where the probability of pathway non-detection, $P(nd) = \Pi(1-P_i(d))$, with $P_i(d)$ being the probability of detection of the i^{th} segment, under the hypothesis of the independence of detection events).

In the technical Addendum to the GIF PR&PP methodology [16], examples on how to evaluate each attribute are given, providing a solid base for reuse.

4.3.1.1. Approach to Prioritizing Safeguardability Evaluation Parameters Using the Delphi Method and Analytic Hierarchy Process, KINAC [73]

The study sorts the safeguardability details for consideration provided by [62] according to the SBD phases and ranks the relative importance of NMA, DIV, and C/S through a Delphi Survey process conducted among a panel of experts. Thanks to an Analytic Hierarchy Process (AHP), the document ranks in a relative (pair-wise comparison), each attribute with respect to its importance to safeguardability. The study proposes the methodologies used to quantitatively convert the qualitative opinions of various experts. Note that this is, given the interpretation used in this thesis, considering the impact on the actual inspection (rather than establishing a safeguards procedure).

The study itself uses INPRO assessment [5], GIF PR&PP evaluation [17], the FSA [80], the IAEA's [62], and the JRC's [81] as literature and combines all the parameters into 35 attributes found below.

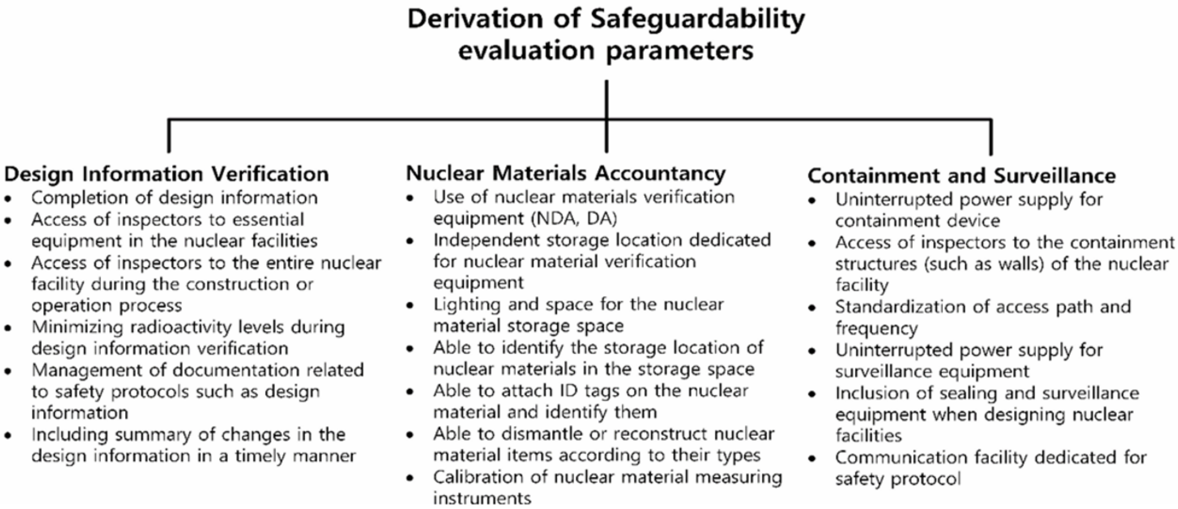


Figure 30: AHP hierarchical structure model for establishing the priorities of safeguardability evaluation parameters [73]

The result of the study is that, when evaluating the relative importance of the safeguards branches, **NMA accounts for 48%, compared to 25% for DIV and 27% for C/S.** It is then sought and expected that this relative importance will be transferred to a higher number of attributes, or a higher weight of these attributes, into the final safeguardability score. See Appendix E for further work on this.

4.3.2.8. Guidelines for the Performance of Nonproliferation Assessments, PNNL [11]

The goal of the project was to advance the process of developing integrated methodologies to address non-proliferation issues, for which none had been previously created. The guidelines build upon earlier work to take the next step towards achieving a hierarchy of methodologies that can be employed and credible to non-proliferation analysts, policy makers, and other stakeholders.

Three general categories of analytical methods, attribute analysis, scenario analysis, and two-sided methods, present potential for use in non-proliferation studies. Attribute analyses evaluate the effectiveness of proliferation to serve a purpose similar to DPA, and the two-sided approaches explore the human role in proliferation.

Multi-Attribute Utility (MAU) has been described as one of the most promising methodologies for assessing a proliferation risk, of which safeguardability can be considered a numerical subset (discussions on the definition of proliferation risk in 5.1).

The study presents a collection of possible metrics and requirements for the quantitative assessment of PR, and safeguards are included as extrinsic barriers to proliferation. The relevant metrics should be considered for inclusion into a subsequent model.

A similar review study is done in [20].

4.3.2.9. Methodology for evaluating proliferation resistance of nuclear systems and its case study: the COMPRE methodology, KINAC [28]

The COMPRE methodology has been developed as an attempt to give a quantitative evaluation of the qualitative parameters (or their qualitative evaluation) considered in a PR assessment. Essentially, the general structure of the INPRO is maintained, and evaluation parameters are given, with an associated score. These evaluation sheets are:

- Evaluation sheet for legal & institutional framework, giving a score based on which legal framework the country adheres to
- Evaluation sheet for material quality and conversion effort
- Evaluation sheet for design information (DIQ process)
- Evaluation sheet for MC&A
- Evaluation sheet for verification (DIV process)
- Evaluation sheet for C/S

The latter three are the most relevant from a safeguardability perspective, on which the study puts emphasis, in alignment with this thesis's goals.

The score is calculated through an additive aggregation method, which means that scores of all measures are added to evaluate the total score for a facility. Since the net contribution from each measure can be identified clearly and intuitively, the additive aggregation method has an advantage over other methods, such as the multiplicative aggregation. The aggregated score allows an effective communication on the safeguardability of a given design. Discerning power is proven even with the very limited quantity of attributes and their simple evaluation parameters. It is implicitly assumed that all measures are independent of each other and can be evaluated separately. If there are nontrivial interactions among measures, a re-evaluation could be necessary.

4.3.2.10. Proliferation Resistance Analysis and Evaluation Tool for Observed Risk (PRAETOR)—Methodology Development [12]

The PRAETOR is a code based on the MAUT and thus employs attributes, which are both intrinsic and extrinsic barriers to proliferation in a NES. The PRAETOR code derives a single metric for PR performance comparison by doing a linear combination of the sixty-three inputs using their respective weights (relative importance). The attributes are also grouped by categories. For each sub-group, MAUA is performed, and each sub-group has a relative weight to the overall system. The highest Overall PR for a realistic assessment obtained using PRAETOR is around 0.80 [12]. The article recognises the limitations of the MAUT in affecting a weight factor that is diversion-path-dependent.

4.3.2.11. Applying System Analysis and Risk-Informed Approach to Support the IAEA Safeguards and Non-Proliferation Regimes [72]

In this PhD dissertation's chapter 5, the Doing-it-Differently methodology stemming from System Thinking is applied to the evaluation of the efficiency of the NMA in any system. This methodology is based on the conversion of each attribute into a process, for which performance indicators have to be evaluated through expert elicitation. The methodology uses mathematical models to propagate the evidence for the success of each process, taking into account the incompleteness of information and uncertainty. The sub-processes and performance indicators can be translated into attributes for an evaluation tool

4.3.3. Summary of the Literature Review

Few models exist for evaluating the safeguardability of NES. Much of the literature focuses on PR/PR&PP. PR is assessed through a barrier evaluation approach or a pathway approach. Safeguards are part of the extrinsic barriers and are applied to counter diversion paths; however, both approaches are not foreseeable in the case of:

- An inexperienced user (the DPA can be effectively performed only by an experienced safeguards expert)
- The information about the design is incomplete, which is surely the case for liquid-fuelled MSRs, or any design in the pre-conceptual phase.

The quantitative evaluation of safeguardability is particularly challenging, but the MAUT seems a promising path forward. [81] recognises the need for providing a safeguardability "score". All the relevant literature can provide attributes, while [73] [12] might provide a basis for setting the weights.

No methodology tailored to MSRs exists, so an integration with attributes from the relevant literature (used in Chapter 3) must be done after the model is developed for a general reactor design.

4.4 The SAT (Deliverable 2)

The Safeguardability Assessment Tool (SAT) is conceived as a comprehensive aggregation of the existing safeguardability evaluation attributes found in the literature.

The persistent goal has been to conceive a tool that, as much as possible, provides a quantitative evaluation of the qualitative attributes that assess the safeguardability of a given design. Having this in mind, it was essential to gather attributes that did not focus on a resistance to a given proliferation pathway but rather capture how a certain design feature would impact the establishment (and eventually operation) of a safeguards approach. Given that **the SAT is not a PR evaluation tool**, its attributes capture features that impact the efficiency or effectiveness of establishing a safeguards approach, i.e. safeguardability. To effectively give an evaluation "score" to a certain design, high-level attributes are not sufficient because their evaluation would heavily depend on the interpretation and ability of the expert performing the assessment. On the other hand, assembling very detailed characteristics of a safeguards system would have yielded a result not dissimilar to the IAEA's "Matrix of safeguards details for consideration" [62], which are to be considered infrastructure guidelines to follow throughout the SBD process [33], without capturing early-design choices' impact on the overall safeguardability. The IAEA checklist is useful as a list of safeguards instrumentation that must be included (even if the safeguards instrumentation and approach are the responsibility of the IAEA, provision for space and auxiliary equipment is the responsibility of the designer [33]).

It is essential to delve into the practical and technical implications of each design choice as designs change safeguards-relevant characteristics based upon non-obvious changes, even in non-actinide materials. As an illustrative example, what are the consequences of choosing a Fluoride-based fuel rather than a Chloride-based fuel? At a higher level from a safeguardability perspective, the choice would seem irrelevant. On a lower level of detail, the designer could ask themselves whether NDA instrumentation is more resistant to corrosion from either salt, or whether the uncertainty of measurement in each salt will impact the safeguards approach, and thus the safeguardability, of each alternative design option. Chlorides require cladding, which constitutes an additional layer impacting measurements. Another instance is the isotope ratios of non-actinides in the feedstock salts, which can indirectly generate a diversion path (see DPA). Due to these implications, the goal of the SAT is not only to establish a safeguardability score at a given moment in time with given safeguards instruments, but **it should be integrated into the SBD approach for grasping the design features that will pose problems** in the future (during safeguards implementation) when opting between design options. So, the user could use this as a list of information that should be known over time. Despite this, the tool has been conceived with the understanding that the designer should be aware of all relevant details about its design when evaluating it (e.g. the neutronics of the core impact the safeguards, but from the point of view of the inspectorate, it is an irrelevant parameter to communicate since it must be known for operating the reactor)

Finally, the SAT could serve as **a decision-making tool to help policymakers (or any decision-maker) by providing an easier-to-comprehend quantitative "safeguardability score"** for a given design, allowing, for instance, the choice of which designer to subsidise or which design is suited for export. For these reasons, the tool must consider that safeguards evolve at the intersection between the hard (technical) layer, which captures deeper-level attributes, and a soft layer that includes, for instance, the legislative and political background of the country in which the design should be implemented. This is the reason why attributes that are related to the broader context of PR are included.

Regarding the desired outcome for the SAT, [11] states that "*Some aggregation of results is necessary in the analysis to make the results understandable. The aggregation of metrics must be done in a manner that minimises the loss of information, minimises interdependencies, is traceable, and provides usable information to policymakers. Detailed results should be documented to enable the policy maker to be able to trace higher-level back to lower-level causes*". These requirements should be fulfilled.

The SAT itself is presented as a list of attributes, first constructed for the general evaluation of nuclear facilities, given the existing methodologies, but later adapted to the specificity of (liquid-fuelled) MSRs. The obtained structure is that of a decision tree. It is presented as a mind map to allow navigation and easier visualisation. Another version is a spreadsheet, in which the attributes are phrased as questions that the user should ask and answer to use the tool. Many attributes had higher or lower levels of detail and were interdependent. Each sub-level should be composed of all the attributes necessary for describing the higher level, and so on. In the construction process, mutual exclusivity had to be avoided to prevent the obtention of a logic tree. Meaning that X = Y1+Y2 and that having Y1 does not exclude the possibility of having Y2 (not either/or structure). This layered version is particularly useful, having **different users who may need different levels of detail**. For instance, a policymaker might want an evaluation on the general NDA applicability, while an inspector needs to delve into the details of NM signatures. This layout was particularly adapted for integrating the attributes coming from [33]. For example, Annexe C.1 "*In this design, is there the potential to create additional diversion paths or alter existing diversion paths?*" was both at a very high level and hard to evaluate. It was kept as a rationale, or father attribute, for the sub-questions 1.1 "*Does this design introduce nuclear material of a type,* *category, or form that may have a different significant quantity or detection time objective than previous designs/processes (e.g., mixed oxide rather than LEU, irradiated vs. unirradiated or bulk rather than items)?"*; 1.2 "*Does this design layout eliminate or modify physical barriers that would prevent the removal of nuclear material from process or material balance areas, e.g., circumventing a KMP?"*; 1.3 "*Does this design obscure process areas or MBA boundaries making containment/surveillance or installation of verification measurement and monitoring equipment more difficult?"*; and 1.4 "*Does this design introduce materials that could be effectively substituted for safeguarded nuclear material to conceal diversion?"*. The approach is satisfying because the higher-level question of the Annexe C is indeed giving a broader context or "rationale" for the lower-level questions.

To make the construction of the tool traceable, relevant references are kept as separate sheets. Usually, these sheets include the attributes in their original form within the source document, and the modified version that suits the general structure of the SAT. Particular work has been done on the IAEA's "Matrix of safeguardability details for consideration" checklist. The SAT would not replace the IAEA's guidelines but rather be complementary to them. While the SAT serves to evaluate design choices and compare between designs, the IAEA's matrix serves as a "check" list of details to consider while designing a facility for more practical operation of the safeguards approach. The list has been reorganised following the three phases of the SBD process and the free branches of safeguards (DIV, NMA, and C/S). Additionally, they have been separated into parameters that either increase the safeguardability through the effectiveness of the safeguards (increase detection probability), increase the ease of performing the inspection (increase detection resource efficiency), or in designing the safeguards approach. **This is useful for providing rationales to a designer using the IAEA safeguardability checklist in an SBD approach**. The adapted matrix can be found in Appendix D.

5 Safeguardability through System Analysis Approaches

5.1 System Analysis or Risk-Informed Methodologies?

With the forecasted strong expansion of civilian applications of nuclear technologies, both in the number of facilities and their geographical spread, the safeguards effort will consequently rise. All these facilities will have to be safeguarded. [5] states that: "*With sufficient cooperation and resources, any (nuclear facility) can be adequately safeguarded, but it is important to bear in mind that the cost of providing safeguards assurances depends, inter alia, on the nature of the nuclear energy system used in a State*". The question is thus not whether the facilities can be safeguarded, since continuous inspector presence can achieve this, but rather how to do so efficiently, given the already limited resources of the inspectorates. To make the safeguards approach more efficient, studies have explored the possibility of applying methodologies widely used in the engineering world to the Safeguards domain. Among those, risk-informed methods are of particular interest. The first reason is that they are extensively applied in nuclear safety engineering, making them familiar to designers. Secondly, it is natural to think about the possibility of designing a safeguards approach through a risk-informed exercise, elaborating on the concept of detection probability. Many risk-analysis[5] methodologies appear to be well-suited. For instance, one could consider proliferation as a success event and construct a success tree with the sequence of successes or failures eventually leading to the proliferation event. Similarly, fault tree analysis could be imagined by taking the proliferation as the top event, whose occurrence depends on the failures of the safeguards verification activities as described in the tree branches[6].

However, a few drawbacks appear. Firstly, **one problem arises from the definition of risk itself**. According to [80], risk is defined as "*the combination of the probability of occurrence of harm and the severity of that harm*". In other words, the product of the

[5] *Risk analysis* is the "systematic use of available information to identify hazards and to estimate the risk". *Risk evaluation* is the "procedure based on the risk analysis to determine whether the tolerable risk has been achieved". *Risk assessment* is the "overall process comprising a risk analysis and a risk evaluation". [20]

[6] when these trees are constructed with the detection probability as the considered metric, there is no doubt they would constitute a safeguards assessment and not a more general proliferation assessment

likelihood of a hazardous event and the consequences of that event**. Can this definition be applied to proliferation?**

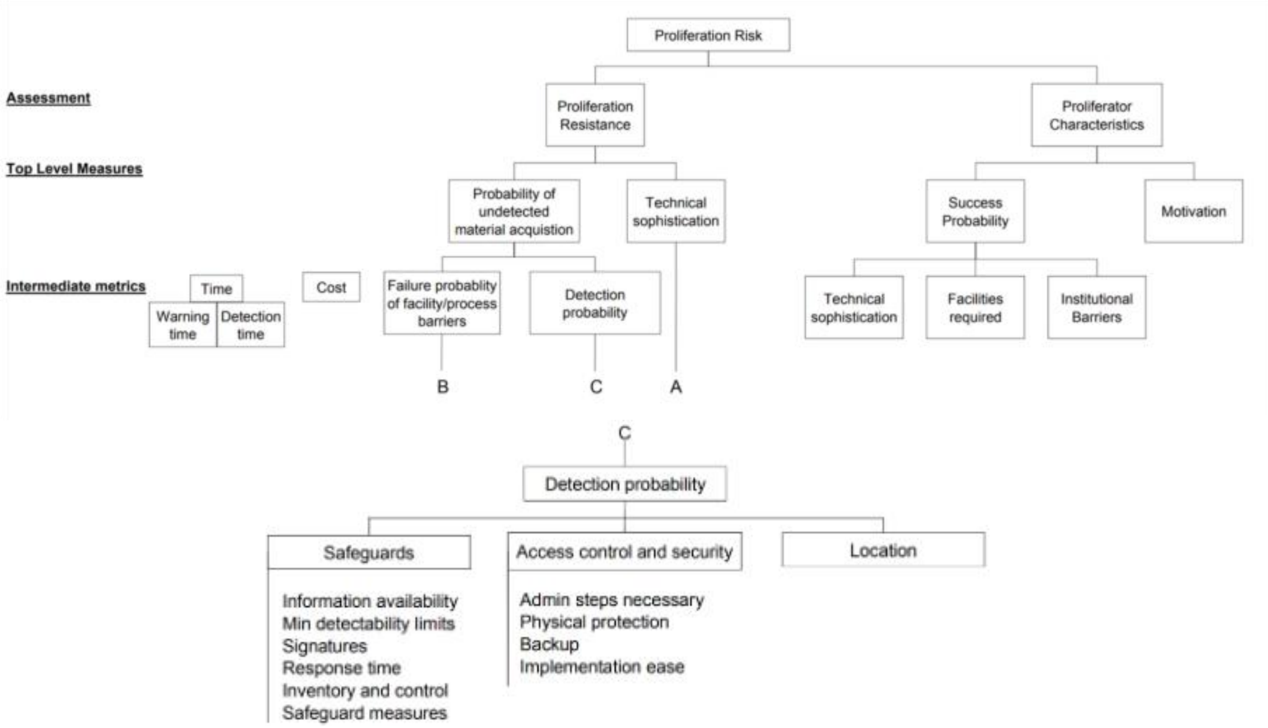


Figure 31: If the definition of Proliferation Risk is accepted, [11] proposes a hierarchy and possible assessment metrics

On the one hand, the likelihood of a diversion event is not easily computed. A diversion path is the combination of diversion and concealment actions, meaning that the process and the signals that a diversion has taken place are actively hidden. Hence, the probability to be assessed would be the combination of the probability of success of a given diversion event and the probability of failure in hiding the generated anomalies. These failures are not only failures in inspections, but also vulnerabilities in the safeguards equipment, combining human and machine failure probabilities to be computed [20]. Moreover, **an intrinsic limitation to the estimation of safeguards-related probabilities of failure is that no database containing such information exists, nor can it exist**. For first-of-a-kind systems, probability distributions cannot be statistically derived due to the lack of operational experience. For already existing systems, a safeguards failure probability database cannot be reliably established due to several factors.

- There are no known, proven, successful proliferation efforts in a safeguarded country.
- Every nuclear facility has a dedicated safeguards approach, with slightly different verification activities differentiating it from other similar facilities.

- With the advent of proper State-Level Safeguards approaches, even the verification activities in the same facility are subject to change, further complicating the creation of a statistically representative database.
- Usually, a single Safeguards equipment failure does not represent a loss of continuity of knowledge, and loss of continuity of knowledge is usually not due only to Safeguards equipment failures.

Additionally, such an approach is limited in that multiple diversion paths can be taken simultaneously, and **the cumulative probability of detection may differ from the sum of the individual probabilities** [12]. Usually, safety-based approaches adopt a closed-world assumption, where the entire universe of discourse is considered to be known. In addition, initiating events are often assumed to be stochastic in nature. In a nuclear proliferation scenario, actions are driven by creative intentionality: it is highly unlikely to be able to completely describe the entire university of discourse, and estimating the likelihood of the occurrence of an intentional event in a given timeframe is not as accurate and reliable as in a typical safety-related event. An open-world approach should be adopted instead to acknowledge that some events are unknown.

On the other hand, when consequences are considered, it emerges that their definition and assessment are not straightforward. Whereas the result of an accidental seal break can be evaluated in terms of economic repercussions for the inspectorate (if a PIV has to be performed, it takes time, and therefore it impacts the resources of the inspectorate), the consequence of successful proliferation by a proliferator is both difficult to define and hard to assess. There is currently no real international agreement on what the consequence to be assessed in a successful proliferation. If the ultimate consequence is the detonation of a nuclear device over a city, **the related risk would be extremely difficult to quantify.** Moreover, the spectrum of potential actors and their potential objectives make the threat identification highly multidimensional [11]. The risk assessment becomes impractical both from the threat (probability of having a certain actor and its objective) and for the severity, given that the target cannot be accurately forecast. Overall, **the definition of risk in the safeguards world is problematic by itself and still an unsolved problem**.

By analysing past approaches, risk-informed methodologies for safeguards assessments (of which a summary and commentary is done in [20], and more generally about non-proliferation in [11]), or the restriction of non-proliferation assessments to safeguards, it is observed that they are based on the evaluation of the detection probability for a given diversion path. This implies **computing all the possible diversion paths, which is a complicated task**. However, even if one were to compute all the possible diversion paths, the ones that do not yet exist or are not known to exist are not considered. **Deterministic approaches that rely on predefined diversion scenarios may overlook proliferation risks unique to novel technologies such as the MSR technology**. This is intrinsic to the fact that experts are biased by their experience with traditional item-counting reactors or batch facilities and lack experience in MSRs. Moreover, with incomplete designs, **not all the necessary information is available to perform a satisfactory DPA**. The diversion paths deriving from already widely accepted features (e.g., online reprocessing unit) can be analysed. Still, ultimately, the exhaustivity of the analysis by the inspector is not guaranteed (also affected by human error, not just bias). Additionally, [32] recognises that the usefulness of the designer performing such an analysis is very limited; thus, **the DPA does not seem to be a suitable instrument for a self-assessment by the designer during an SBD process**.

For these reasons, **this thesis seeks to explore system analysis methodologies rather than risk analysis ones** that could help, in a structured and mathematically rigorous manner, both the designers and the inspectors in working towards the applicability of safeguards to next-gen designs like MSRs, and particularly to assess the applicability of NDA to liquid-fuelled MSRs.

It has been mentioned in the foreword to the SAT that it should be integrated into an SBD approach for grasping the impact of the choices in design features. Building on the foundations of the SAT, the thesis seeks to delve further into the dependability of the safeguardability of a facility on its design choices, what features would positively impact the applicability of NDA, and what R&D should be prioritised for reaching a deployment of safeguards on liquid-fuelled MSRs. A methodology that considers uncertainty and its propagation must be selected, given the gap in expertise, and given that designs are not fully established, nor fully disclosed at times.

How can we reliably assess safeguardability when the reactor designs themselves are still evolving, and many technical parameters remain undefined?

Two methodologies have been selected to be applied to the list of attributes of the SAT.

5.2. The Selected System Analysis Methodologies

5.2.1. Decision Theory and the MAUT

5.2.1.1. Multi-Attribute Utility Theory

The Multi-Attribute Utility Theory (MAUT) is a decision-making method developed to address complex scenarios involving multiple, often conflicting, attributes. This approach originated in decision theory in the mid-20th century, as a systematic framework to quantify and compare preferences when decision-makers face more than one determining factor. It is directly applicable to contexts such as policy analysis, where the growing sophistication of real-world problems defies simple, single-numbered judgments, and in which each contributing factor must be carefully weighed. MAUT is widely used in fields such as engineering, public policy, and project management [86].

The principle behind the MAUT is to synthesise the decision-maker's preferences, which must be captured, into a comprehensive score for each pre-specified alternative. [86]. This is done through the construction of the Multi-Attribute Utility Function (MAUF). The first step is to form a set of n attributes, x_i for $n \in \mathbb{N}$ and $0 \leq i \leq n$, that can best describe the system under consideration. The MAUF can then be written as a function of these attributes, $U(x_1, x_2, \ldots, x_n)$. To each attribute x_i, a utility value is assigned (between 0 and 1), representing the performance of this attribute in the evaluated system, which is then $u_i(x_i)$. To each $u_i(x_i)$, a weighting factor k_i is applied (between 0 and 1 and the sum of which is 1), representing the importance of x_i towards the overall performance of the system. Then, in its additive form:

$$U(x_1, x_2, \ldots, x_n) = \Sigma\, k_i\, u_i\, (x_i) \text{ for } n \in \mathbb{N}\ and\ 0 \leq i \leq n \qquad 11)$$

This is limiting since U is also between 0 and 1, which limits the range and the influence of a single attribute [12].

Applied to the decision theory, it is understandable that the weights correspond to the preferences of the decision maker. A more substantial weight directly reflects the importance that the decision maker attaches to one parameter in making a decision. The weights are only valid for a given decision maker and would radically change when the user changes. Other forms of the MAUF, multiplicative or more complex, exist, reducing this issue but complicating the calculations and tracking.

A complex system composed of many subparts could be the aggregation of several MAUFs, and the total score would exceed 1. An adaptation could be imagined having a relative value of the weights in which their total does not have to add up to 1, but is defined in comparison to the least or most influential factor.

The primary result of this technique is a ranked list of alternatives. The chosen alternative is ideally the one with the greatest overall utility, ensuring that decisions are closely aligned with stakeholders' expressed priorities and risk tolerances. The main criticality is thus the definition of the decision (which can be divided into decisions under certainty and under uncertainty) and the modelling of the decision-maker's preferences. When involving uncertainty, the values of the score of an attribute take intervals/distributions, which complicates calculations. The analytic structure of MAUT provides a transparent, aggregative, trackable, and rational basis for decision-making [11, 12].

5.2.1.2. MAUT and the SAT

The COMPRE methodology [28] was chosen as a starting point for developing the tool because it aligns closely with the objectives set for SAT, even though the desired level of detail was not fully attained. The COMPRE methodology provides a safeguardability score through a MAUF. The MAUT respects the established objectives for the SAT, such as the rigorous and traceable aggregation of metrics. The number of attributes is unlimited, and the weight assigned to each attribute can be tailored to reflect its relative importance (depending on the user) in determining what constitutes a "success" in each attribute towards the safeguardability of the design. The MAUT is conceived specifically for decision-making, making it fit-for-purpose. Lastly, within the SBD process, a higher weight indicates information that should be prioritised when the designer lacks it, and early efforts will result in higher final safeguardability [73]. When constructing the SAT, a decision had to be made regarding the preferred user. Choosing the inspector as the primary user is logical, as it is the inspectorate's duty to safeguard any licensed facility. To help establish the weights for the MAUF, the results of the Delphi survey and AHP [73] have been ranked by score. It should be noted that this marks a substantial departure from the MAUT, since **in the construction of the MAUF, the affected weights should be representative of each decision-maker's preferences**. Here, instead, **the preferences and expertise of the inspectors are captured** and provided to the other users as a baseline for their work. This approach is satisfactory given that only the inspectors have sufficient knowledge about the safeguards system and procedures to provide actionable knowledge. **This methodology can effectively replace the DPA in an early self-assessment by the designer**. Moreover, once the list of attributes is provided, the user can still compose their MAUF.

Finally, the MAUT works for a set of attributes that are on the same level of detail; thus, a version of the SAT including only the final layer has been produced to facilitate the application. If desirable, another MAUF can be established for a higher level of detail. This approach is already partly in place, given that an MAUF is created for each branch of the safeguards: DIV, NMA, and C/S, and could be extended to further sub-categories.

The evaluation parameters, utility functions, and their linear combinations will be obtained by expert elicitation. Dependencies are managed during the expert elicitation because the scores will necessarily be tied. Where attributes are dependent, a high score on one will imply a high score on the other. Otherwise, rationality is compromised.

5.2.2. Evidence Theory and the Italian Flag Methodology

Stemming from the University of Bristol's "Doing-it-Differently" Systems Thinking approach, Italian Flags are a visual representation of the dependability of the success or failure of a process given the available evidence for and against its success. It offers a structured yet flexible alternative for decision-making under uncertainty. Originally developed for engineering design decisions with incomplete information, its three-tiered classification system (green/white/red) provides several advantages for evaluating emerging technologies, such as Safeguards systems for MSRs, which make it worthy of investigation for the application to a safeguardability assessment.

The methodology in this thesis has two key aspects: 1) the structure of the information and the propagation of it, and 2) the way it captures, propagates and visualises uncertainty.

5.2.2.1. The Evidence Theory or Dempster–Shafer Theory (DST)

Systems are characterised by uncertainties, which can be of different natures. **Uncertainty is composed of randomness, fuzziness, and incompleteness of the information**.

- Randomness is the aleatory uncertainty, which is stochastic and can be reduced, but is theoretically and practically ineliminable [87]
- The Epistemic uncertainty results from the **lack of knowledge about the system**, and groups both fuzziness and incompleteness. Fuzziness is about imprecision of definition, and includes, inter alia, modelling uncertainties and approximations. Incompleteness takes into account that complex, real-world systems are seldom completely known and described, and the information about them is not always available. **Epistemic uncertainty can theoretically be eliminated, but not in practice. In a real-world scenario, it can only be reduced**.

Classical Probability theory is particularly apt for addressing randomness, but has limitations in capturing the epistemic uncertainty [87]. Evidence Theory, on the other hand, allows the construction of intervals (that can be probability intervals) representing the interpretation of the available evidence, in an open-world assumption, and naturally combines the different types of uncertainties related to the evidence coming from different sources.

The framework allows for an effective transposition of the qualitative opinion of experts into quantitative success intervals. The evidence is supporting a set of hypotheses, among which one is taken to be true. A proposition about where the truth resides is made and this proposition is evaluated by the expert. The expert associates a so-called **belief mass to each proposition, meaning the expert's interpretation of how likely the truth is to be found within the evaluated set of hypotheses**. This can provide the experts with a tool that could better capture the nuances in their quantifications, especially when information is vague and incomplete.

DST formalism

Conceptually, let Ω be the frame of discernment. This is the whole set of hypotheses ω that fully describe the universe of discourse, so that $\Omega=(\omega_i)$, $i \in \mathbb{N}$. **In the standard DST, the initial set of hypotheses, represented by the frame of discernment Ω, is assumed to be exhaustive and mutually exclusive.** Semantically, Ω is a space of possible truths, and because the description of the universe is taken to be exhaustive, the truth (meaning the true hypothesis) must lie in Ω. Evidence should help to recognise where the truth finally is.

Let A ⊆ Ω be a set of hypotheses. A is called a focal set of Ω if the evidence supports the proposition that "truth lies somewhere in A". Thus, A represents a proposition about where the truth is. If **A is a focal set, the truth lies in A, and no other options exist.** Thus, the standard DST is working under a closed-world assumption. However, when A is a non-singleton subset, it actually represents a partial ignorance about the truth as no support to A's elements individually is done. **The evidence does not provide enough information to distinguish between specific hypotheses**. Larger sets correspond to less informative evidence. If no subset A can be recognised as a focal set with the given evidence, there is a situation of total ignorance, and the only conclusion can be that the truth is within the frame of discernment, which is one of the initial suppositions.

Practically, the model uses basic belief masses (bbm) that represent the expert's interpretation of the evidence defending the different propositions (subsets) about where the truth lies, effectively distributing the belief over the subsets.

A basic belief mass is thus assigned to each subset and represented by m(A); with:

- $m(\emptyset) = 0$ (the belief that no hypotheses are true is zero),
- $\sum$m(A)= 1 for all the subsets A of Ω. This is consistent with the fact that the truth is certainly within Ω.
- When belief mass is assigned to non-singleton sets, there is a situation of partial ignorance.
- Total ignorance is represented by assigning mass to the entire frame of discernment, reflecting imprecision or lack of discriminatory information. Then, m(Ω)>0.

Finally, a function is constructed to quantify the total amount of **belief that the evidence *firmly supports* the proposition** that the true hypothesis lies in A (based only on evidence that does not support any hypothesis outside of A). The belief function is defined as:

$$Bel(A) = \sum_{B \subseteq A} m(B) \qquad 12)$$

The belief function allows for avoiding a yes/no judgment and enables the quantification of the support of a proposition on a set of hypotheses.

Another function is constructed, the **plausibility** function being the sum of all belief masses assigned to subsets C that do not contradict A (the sets have hypotheses in common):

$$Pl(A) = \sum_{C \cap A \neq \emptyset} m(C) \qquad 13)$$

It represents the total extent to which one *fails to doubt* the proposition "the truth lies in A". It includes:

- belief explicitly committed to A,
- belief committed to larger sets that overlap with A and therefore *could* support A.

Unlike belief, plausibility does not require certainty that the truth lies in A; it only requires that the evidence does not rule A out. In classic DST, the two are connected by the following equation:

$$Pl(A) = 1 - Bel(\mathring{A}) \text{ where } \mathring{A} = \Omega \backslash A \qquad 14)$$

The plausibility is then the belief in the hypotheses contradicting A.

The doubt is 1- Pl(A).

Conflict

When two sources of evidence conflict (for example, one source says the answer is A and the other says B, where A and B do not overlap), DST uses **normalisation** to resolve the issue. Normalisation mathematically **discards the conflict** and redistributes that "lost" conflictual mass among the remaining hypotheses so that the total belief always sums to 1. This process forces a conclusion to be drawn, and to do so within the pre-defined set of hypotheses, even if the sources of evidence are in total disagreement. This can be useful when a decision must be made. The resolution of the conflict in DST is essentially avoided, which can be undesirable in some cases. A different frame is proposed by Yager. The resolution of conflict assigns all conflicting evidence to the **whole frame of discernment** (Ω), formally recognising that a conflict of evidence is a source of uncertainty. The underlying principle is that if the source of the conflict is not obvious, then the conflict is intrinsically an uncertainty. It can also be useful in conflicts where it is known that preserving uncertainty is preferable to forced consensus [88]. It increases the gap between belief and plausibility and produces conservative results under disagreement. At the risk of ending up with white across the board, without any actionable knowledge provided, the increased uncertainty will force the resolution of conflict through further investigation.

Other conflict resolution rules:

Dubois–Prade rule: When the expert is sure that the "truth lies within a subset B of A but cannot distinguish between elements of B in terms of accuracy", B is a set of local (partial) ignorance while A is associated with total ignorance. The Dubois-Prade redirects conflict to B rather than entirely to total ignorance, offering a middle ground between Dempster and Yager in conflict management.

PCR rules: Redistribute conflicting mass proportionally to the involved focal sets to better reflect which hypotheses clashed, aiming for intuitive outcomes under high conflict. The result will be an overview of how much red and green are present in the model, and conflict is not really resolved.

Open-world Assumption

In the Safeguards context, it is important to note that a proliferation attempt is actively kept hidden. MSRs being a novel technology, new proliferation strategies might arise that are not known to the decision-maker. For these reasons, it is paramount to be able to include open-world assumptions in the DST model, meaning that the scenario cannot be fully understood, and some information is inevitably missing.

- The concept of exhaustivity of the frame of discernment no longer holds. The truth can now lie outside of it. This is translated by the possibility to assign a mass to the empty set so that $m(\emptyset) > 0$.
- Belief in A, and the belief in Å do not necessarily sum to 1. This is the implication of the fact that, in the open-world assumption, information is missing. This is relevant to an expert elicitation process because the expert could easily assign non-additive values. Mass could be assigned to the empty set, compromising the additivity.
- A new conflict resolution methodology is introduced, Smets TBM, that adapts Yager by actively assigning the conflictual masses to $\emptyset$. Now, the interpretation is that the conflict not only generates uncertainty, but it might hint at the fact that none of the initial hypotheses is correct.

Italian Flag formalism

The methodology is based on the consideration of each attribute as a process, or holon, that is a whole by itself and a part of a greater system. To investigate the success of the overall system, an assessment of the likelihood of the success of the sub-process can be done, and the information is then propagated to the higher level, taking into consideration the dependencies of the sub-processes and their influence on the success of the father. A possible application foresees that these dependencies are always evaluated pairwise to avoid the overstressing of the expert, similarly to the AHP [73]. When reaching the final (desired) level of detail, which allows for a useful description of the system, the processes are then evaluated through enabling factors (EFs) (or key performance indicators – KPIs, when the evaluation is done in time). These EFs are the set of conditions that must be satisfied to consider a success in the last level of attributes and are obtained through expert elicitation (EE). A total dependency implies that the failure in one of the EFs will cause the failure of the others. A total necessity implies the immediate father attribute's failure. However, given the holistic nature of each process, the said attributes can also be evaluated alone. This double approach provides additional information, given the fact that the father process might be more than just the mere aggregation of its sub-processes, and **an evaluation at the higher level could provide additional actionable information compared to the evaluation of the sub-processes and their aggregation**.

Conflicting information can arise in two ways:

- The direct evaluation of an attribute is based on conflicting evidence
- The aggregation of attributes makes evidence conflict

The success of a process can be evaluated through expert elicitation**. The expert evaluates the evidence for/against the hypothesis "the process will be a success", and his opinion is assembled into a unique interval [Bel(A); Pl(A)]**. The Italian flag is constructed so that:

- [0; Bel(A)] is the belief of the success of the evaluated process, represented by the colour green. Conversely, [Pl(A); 1] is the doubt or evidence against the success, represented by red.
- [0; Pl(A)] is the plausibility of the success of the process given the available evidence; thus, [Bel(A), Pl(A)] represents only the subsets where the sources do not provide enough specific information to commit to success or failure. The interval represents the uncertainty of the expert on the success of the process, represented by white.

To provide an example, if the expert's opinion on the NDA applicability to MSRs is [0.2; 0.7], then:

- Given the available evidence, **the expert's belief is a 20% conviction in the success of the applicability of NDA to MSRs**.
- The uncertainty (e.g. applicability depends on the geometry), or conflicting evidence, is 50% of the total. Formally, this means that **the expert finds a success plausible to 70%** (20% for certain success, plus 50% for failure to doubt success).
- Conversely, 30% is the expert assessment of the probability of failure of the given process, based on the available evidence.

Necessities, Sufficiency, and Dependencies towards the success of the father

Sufficiency and Necessity:

Here, the "positive support" (how the sub-process is important for the success of the process) and "negative support" (how the sub-process's failure influences the failure of the process) are considered. From a practical point of view, it is a weight on the green (positive support) and red (negative support): a positive support of 0.6 would weight the green of the sub-process by 0.6. Negative support is related to the necessity of the child to the success of the parent. Positive support is related to the sufficiency (or more accurately the importance) of the child to the success of the parent.

Dependencies:

The Generalised Natural Combination rule is used to manage dependencies that can occur between subprocesses. A success in one of the processes would influence the success of the other. It uses a mapping parameter (r) to represent dependency, where $r = 0$ is independence and $r = 1$ is full dependence.

Takeaways

Finally, the applicability of the methodology to safeguardability assessments has been proven in [72]. The results of this work can serve as further evidence of the utility of this methodology in providing actionable knowledge, but such knowledge is the actual output of the application rather than the proof-of-concept. **The method must provide actionable knowledge, and it is less important to choose a propagation method that is kept throughout the application, rather than choosing or crafting the best-fit propagation method at each instance**.

To make ends meet with the SBD and 3S-by-design frameworks, this holistic approach, which recognises the NES as a hard system operated by a "soft", social layer in a given context, safeguardability and PR are both indices of the quality of the interaction between these layers. PR evaluates the interaction between a NES and proliferation efforts, while Safeguardability evaluates the interaction between the NES and the legal and practical safeguards framework in place [77].

5.2.2.2. The proof-of-concept: Application and the expected results

The methodology has been applied to the holon "NDA", which is a sub-holon of "*Inspector measuring capabilities*" (complementary to the measuring possibilities) in the "*PIV's effectiveness*" branch (which, together with the *PIV's efficiency* branch, constitutes the *NMA* sub-tree). The choice of the branch was made to take advantage of the NDA expertise in the EC Joint Research Centre, and because of the real-world need to identify the gaps in the instrumentation's applicability to liquid-fuelled MSRs. The application of the methodology to this branch will constitute a proof-of-concept to extend the methodology to the whole SAT. The obtained result is a decision tree for which the attributes have corresponding Italian Flags, evaluated both on the bottom-up approach previously described, and as direct evaluation through EE. It is expected to find discrepancies, the reasons for which should be identified. The visual representation of the classification system facilitates communication between the stakeholders. Design engineers, safeguards experts, and regulators can all engage with the colour-coded results. This aspect is crucial for integrating the Italian Flag analysis into an early-stage inclusion of SBD, where dialogue and continuous feedback between stakeholders are essential, making it a very actionable approach. The Italian flag methodology satisfies the previously mentioned condition by providing "usable information to the policymakers" while "allowing them to trace higher-level implications back to lower-level causes" [11].

To summarise, the application of the chosen methodology has several objectives. First, to illustrate and quantify how different design choices influence safeguardability uncertainty levels, providing actionable feedback for reactor designers. Second, to identify which safeguards challenges can be addressed through incremental improvements and those that require a paradigm shift in the safeguards approach (reducing the red). Third, to prioritise R&D efforts where uncertainty is significant and has an impact (reducing the white). Fourth, to stimulate and facilitate dialogue among the various actors.

5.3. The Application of the Methodologies

5.3.1. MAUT Application

The MAUT has been selected as an appropriate technique. Its suitability for a safeguardability assessment has already been proven [5, 12, 28]. However, it has been deemed as less interesting for an expert elicitation process on liquid-fuelled MSRs, given that the direction of safeguards R&D efforts should be the priority, rather than the evaluation. The establishment of the evaluation parameters and the weights is kept as a follow-up action to this thesis.

5.3.2. The Italian Flag Application

The SAT model is tested through the application of the Italian Flag methodology to the evaluation of the applicability of NDA to the MSFR operating on the U-Pu cycle. For applying the Italian Flag, a panel of experts from the EC's Joint Research Centre, with expertise in Euratom inspectors, NDA techniques development, and NDA approaches implementation, has been consulted. All the information provided in the following sections comes from the experts' interpretation of the literature and their personal expertise. For a description of the NDA techniques and measurements, refer to NDA.

The whole decision tree is not reported in the thesis for a readability issue, but portions of it are gradually presented to provide the reader with a visual representation of the father-daughter and dependency relationships between the attributes. **The presentation of the results follows the same bottom-up approach that has been used during the expert elicitation**. A colour code is employed to represent the layer at which the attribute is found in the tree. The EFs could have a layered construction as well.

Figure 31: colour code, the EFs are in light green

The propagation of the flags is done through the expert elicitation process.

5.3.2.1. Evaluation of the attributes relative to the informativeness of the neutron signature

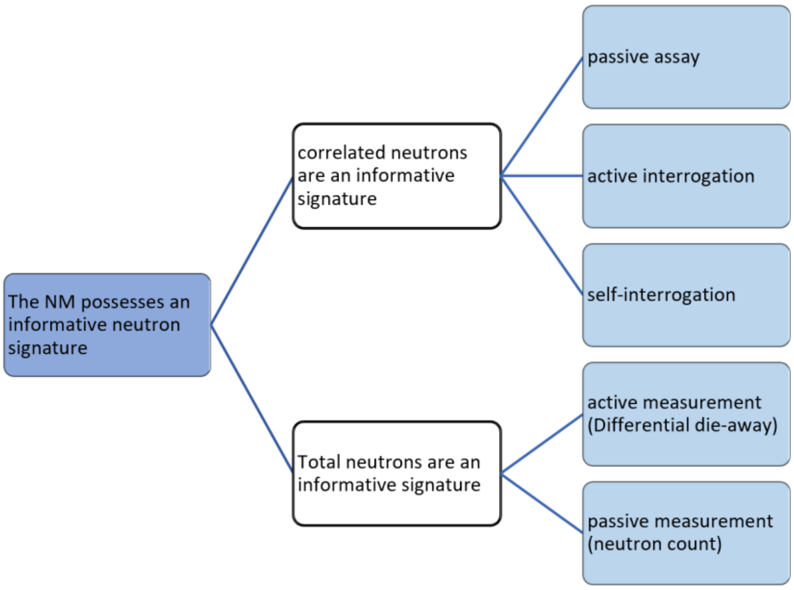


Figure 32: Evaluation of the attributes relative to the informativeness of the neutron signature (EFs excluded for readability)

4.2.2.1.1. Do correlated neutrons constitute an informative signature?

Passive assay

The passive assay attribute is evaluated through 3 EFs.

EF1: need for information about SF-emitting isotopes [1;1]

The isotopic mix is an operational data need and will be necessary at the time of the reactor's operation. It can thereby be considered green.

Further info: Cm is usually found inside SNF. Here, it could be found in fresh fuel if it comes from reprocessed fuel.

EF2: even isotopes of Pu (that have high SF probabilities) are present [1; 1]

Same as EF1

EF3: The ^{242}Cm/Pu ratio is known [0.7; 1]

The operator needs to know the Cm/Pu ratio (Cm tracking), but with today's rules, the inspectorate would need to reproduce the data. Which is expected, with an uncertainty factor.

Further info: EF3 leads to talks about the signatures of diversion. Can the NDA inform me about a Pu removal? The Cm/Pu ratio depends on the operation. Having a low Cm does not imply having a good ratio. We could have a low Pu too, for a U-started core. Moreover, is the homogeneity of the sample guaranteed? Even if it can be considered broadly homogeneous for neutronics, in case we need DA, homogeneity can be crucial

Passive assay techniques will be applicable to the MSFR U-Pu: [0.7; 1]

Further info: For the NDA to become informative, it needs DA sampling (Cm/Pu cannot be trusted through the operator's simulations) and C/S (to make sure Pu is not diverted and, thus, the ratio is correct)

active interrogation

EF1: need for info about fissile isotopes mass/quantities [1; 1]

Operational data needs.

EF2: need for a low neutron environment so that the results of active interrogation are measurable [0.2; 0.7]

The neutron environment does not have to be low per se, but according to the used technique. For instance, with a pulsated source (D-D or D-T or shuffle source), you could work on the temporal correlation of neutrons even in a high background. The 30% stems from the fact that the operator is hardly in favour of additional neutron sources in his facility for safety reasons. The 50% uncertainty derives from the questions: “Can we achieve results? Can we do so efficiently? Can we have field-ready instrumentation?”

EF3: need for a neutron source [0.3; 0.8]

Criticality must be avoided, hinting at the red part (the operator could refuse the source for safety or licensing problems).

EF3.1: Criticality risk is assessed through SBD

Evaluated in EF3.

EF4: The added value of using an active interrogation technique has to be adequate to justify the management of the radio source [0;1]

The added value of active interrogation on the MSFR is unknown.

Active interrogation: [0.2;0.6]

An added value is necessary when opting for this technique due to the criticality risk.

Self-interrogation

EF1: need for information about fissile isotopes other than Pu [1; 1]

EF2: Presence of (alpha, n) reactions that create random neutrons (non-correlated) [0.4; 1]

EF2.1: Presence of alpha-emitting material [1; 1]

Alpha-emitting nuclides (e.g. ^{238}Pu) are present in the isotopic mix

EF2.2: Need for light elements producing (alpha, n) reactions [1; 1]

F, Li or Cl in the salt

EF2.3: The interrogating neutrons must be thermalised and reflected into the item

EF2.3.1: needs suitable geometry and surroundings [0; 1]

EF3: The signature is not hidden by the ^{242}Cm SF [0; 0]

If even an infinitesimal proportion of ^{242}Cm is present, its spontaneous emission is going to be greater than the self-induced fission by orders of magnitude. Which is the expected scenario

Self-interrogation: [0.1; 0.3]

Given the ignorance of the geometry, and most importantly, the absolute necessity of having no obscuration by the ^{242}Cm emissions, self-interrogation techniques are not expected to apply to the MSFR on a U-Pu cycle. They could apply to the MSFR operating the Th-U cycle due to the lower TRU generation. However, even if applicable, it is not expected that they would be applied in the near future.

correlated neutrons are an informative signature

You need to be able to do passive measurements for doing self-interrogation, thus creating a dependency. Passive and active assays are independent. Self-interrogation and active interrogation can be independent because we don't need an active source for doing self-interrogation.

correlated neutrons are an informative signature: [0.8; 1]

The result is due to the syntax of the attribute, which, written as-is, makes the success in a daughter attribute sufficient for the success of the father.

4.2.2.1.2. Do Total Neutrons Constitute an Informative Signature?

Differential Die-Away (active measurement)

EF1: pulsated neutron generator can be used [0.6; 0.9]

A small chance of having a criticality risk is to be considered through the neutronics and geometry.

The MSFR is a fast reactor, the neutrons must be moderated to assay the material. Firstly, the fission cross-section for the interrogated materials are greater for thermal neutrons. The input neutrons have to be moderated. Secondly, for the generated fissions to be observable, they need to interact with the captors. However, ^{3}He captors do not see fast neutrons, and thus, the output neutrons have to be moderated before their interaction with the captor.

The chosen KMP must be adapted, which is hard to achieve with retrofits, and thus must be implemented through SBD.

EF1.1: criticality risk assessed through SBD

EF2: The added value of the use of an active interrogation technique justifies the complications tied to the neutron generator [0; 0.7]

There could not be a need; the added value is unclear.

EF3: fissile (uneven) isotopes presence [1; 1]

Intrinsically true in most areas of the reactor. To test the absence of fissile isotopes in waste streams, other techniques would be implemented.

total neutron counting (passive measurement)

EF1: need for info about the irradiation history [1; 1]

The irradiation history is an operational and NMAC requirement.

Further info: being an integral measurement, different isotopic compositions could lead to the same measurement. Must be tied to C/S and other observables

EF2: even isotopes presence [1; 1]

Present in the isotopic mix.

Total neutrons are an informative signature

The active measurement is dependent on the total neutron counting because it is based on the same measuring instrument.

It is not necessary to have passive to have active measuring possibilities (they depend on different isotopes and are not sub-groups of the same thing).

Both are sufficient, but neither is necessary for the father attribute.

Total neutrons are an informative signature [1; 1]

The NM possesses an informative neutron signature

Correlated neutrons are a part of the total, so there is a total dependency between the two (even if you can have a total count without correlated neutrons).

Both attributes are sufficient conditions for having an informative neutron signature.

The NM possesses an informative neutron signature [1; 1]

5.3.2.2. Does the NM possess an identifiable gamma signature?

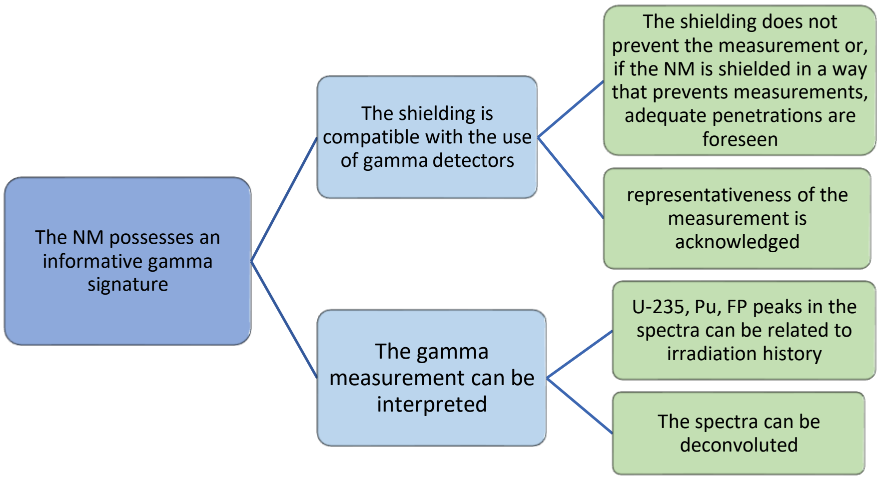


Figure 33: Evaluation of the attributes relative to the informativeness of the gamma signature

The shielding is compatible with the use of gamma detectors

EF1: The shielding does not prevent the measurement or, if the NM is shielded in a way that prevents measurements, adequate penetrations are foreseen [0.5; 1]

It can be implemented through SBD if the problem is acknowledged.

EF2: representativeness of the measurement is acknowledged (penetrability of the signal in the assayed material) [0.8; 1]

There is some degree of uncertainty coming from the fact that the reactors are not in operation yet, but this is a parameter that the inspector will most likely acknowledge. The homogeneity of the NM is likely, but unclear to some extent.

The gamma measurement can be interpreted

EF1: U-235, Pu, FP peaks in the spectra can be related to irradiation history [0;1]

Problems might arise with reprocessing. There is a **crucial need to complete the missing information** that generates the uncertainty.

EF2: The spectra can be deconvoluted [0.7; 1]

Deconvolution means that peaks can be resolved by managing the interferences between them. The instrument must have a sufficient spectral resolution, which is achievable with the existing panel of techniques. In practice, there is a trade-off between the efficiency (in detection) and resolution (measurement effectiveness), because more efficient detectors involve bigger volumes, thus worse resolution. Techniques such as microcalorimetry require heavy accommodations (e.g., cooling) that might be difficult to implement. The deconvolution technique is known and consolidated. However, the presence of FP might mask spectral features from NM nuclides, and an uncertainty factor must be compounded, given that the SNF spectra are not available yet. Thus, we don't know what instrumentation is required.

The gamma measurement can be interpreted: [0.9;1]

The NM possesses an informative gamma signature

Overall, there are no evident showstoppers, but a lot of uncertainty in the data.

It is expected that the NM will emit a measurable and informative gamma signature, but the resolution of the incompleteness of information might lead to the presence of critical showstoppers that need to be identified as early as possible. The amount of green here should not be interpreted too positively

The NM possesses an informative gamma signature: [0.7; 1]

5.3.2.3. Does the NM possess an informative calorimetry signature?

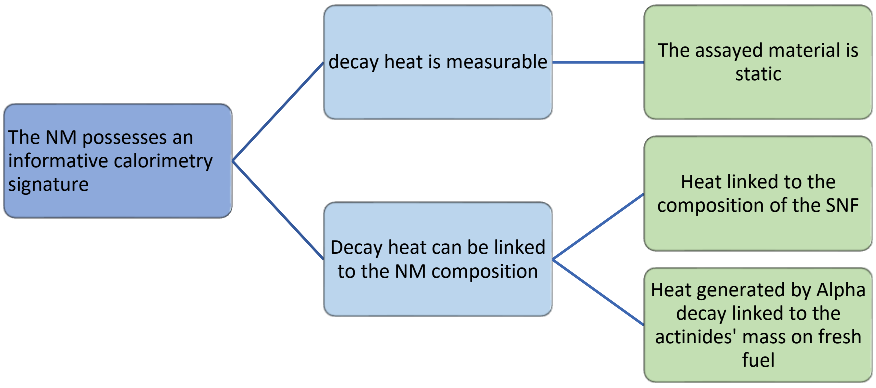


Figure 34: Evaluation of the attributes relative to the informativeness of the calorimetry signature

The decay heat is measurable

EF1: The assayed material is static [0.3; 0.5]

At first impact, only the fresh fuel is static. It produces heat if it is a fresh fuel coming from reprocessed SNF. In the on-site reprocessing, we could have some static inventory (storage). On the SNF batches, it could be possible to measure the decay heat. Inside the reactor, it is not a static inventory, comprising most of the NM on the site.

The decay heat is measurable: [0.3; 0.5]

The measurement of the decay heat is integral, lasting weeks. The other measuring possibility is to observe the temperature gradient in a controlled dynamic environment, lasting a dozen hours. Additionally, the salt is at a high temperature, and to be able to discern and observe the decay heat, instead of the very reactor temperature, you should wait a long time for it to reach equilibrium. The applicability is questioned.

Decay heat can be linked to the NM composition

EF1: Heat linked to the composition of the SNF: [0.7; 1]

How is it possible to track the history of a continuously processed bulk through online reprocessing? We have to take the NM isotopic composition as an input for the measurement to be able to obtain information. Moreover, calorimetry takes as a supposition that the measured item is in a static state.

EF2: Heat generated by Alpha decay linked to the actinides' mass on fresh fuel [1; 1]

If the fresh fuel is reprocessed MOx, calorimetry can be applicable, even if it might not be the best technique. The strong aspect of calorimetry is the precision and accuracy of measurements. In this case, it might not be necessary.

Decay heat can be linked to the NM composition [1; 1]

The technique applies to fresh fuel, and given the semantic construction of the attribute, it is then an overall success. However, the proportion of fresh fuel is limited.

The NM possesses an informative calorimetry signature

Decay heat is necessary and sufficient for having a calorimetry signature (that can be informative or not)

The informativeness of calorimetry is generally limited to the MSFR, but the information can be extracted if the NM composition is obtained. The main question is the measurability of the decay heat. Calorimetry could be a last resort for the inspector, if applicable, given that the accommodations are huge and the obtained information is limited

The NM possesses an informative calorimetry signature [0.5; 0.5]

5.3.2.4. Does the design enable different measuring techniques to complete each other?

There is a dependency between the gamma and neutron signatures, as gamma rays are emitted after a neutron-nucleus interaction. All of the daughter attributes are necessary and sufficient "by pairs" for the success of the father (given the semantics of the attribute).

The design enables different measuring techniques to complete each other (reverification and robustness of the system): [0.5; 1]

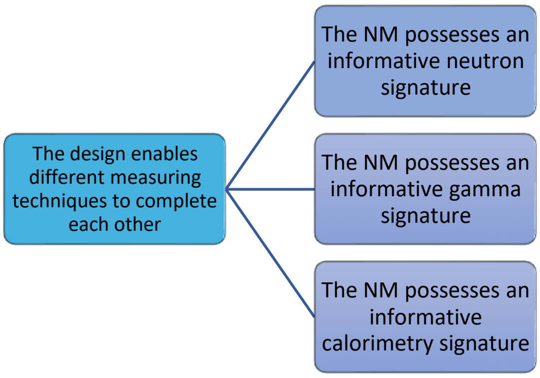


Figure 35: Attributes needed for evaluating the complementarity of measuring techniques

5.3.2.5. Does the design affect the measurement accuracy at KMPs through design characteristics?

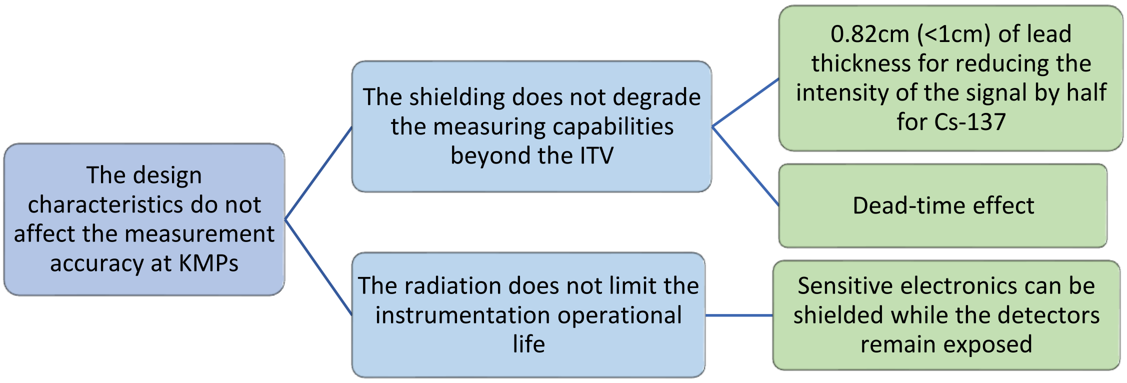


Figure 36: Evaluation of the design characteristics' impact on the measurement accuracy at KMPs

The shielding does not degrade the measuring capabilities beyond the ITV

EF1:0.82cm (<1cm) of lead thickness for reducing the intensity of the signal by half for Cs-137 (of which the signal is stronger by a factor of 10^5)

Directly evaluated at the father attribute

EF2: Dead-time effect

Directly evaluated at the father attribute

The shielding does not degrade the measuring capabilities beyond the ITV (signal is too weak or too strong): [0; 1]

Design characteristics are missing if the design is not complete

Further info: Heavy shielding is likely to be required

The radiation does not limit the instrumentation's operational life

EF1: Sensitive electronics can be shielded while the detectors remain exposed

Directly evaluated at the father attribute

The radiation does not limit the instrumentation operational life (beyond what is reasonable for resource efficiency) [0; 1]

Design characteristics are missing if the design is not complete

Further info: The in-core radiation would not permit the installation of remote in-core measuring capabilities

Does the design not affect the measurement accuracy at KMPs through design characteristics? (e.g. piping thickness tied to the operative pressure, or required shielding degrading the measuring capabilities)

Shielding and operational life are clearly dependent on each other.

Design characteristics are missing if the design is not complete.

The design does not affect the measurement accuracy at KMPs through design characteristics [0; 1]

5.3.2.6. Does the design employ nuclear material types, categories, or forms (physical/chemical/isotopic) that are more difficult to measure accurately at KMP?

The penetrability of the radiation depends on the form of the material (liquid, gas, bubbling...) because of the density change. The mass absorption is the same, but the absolute absorption changes with density. Liquid state often implies a high temperature: the needed measurement techniques do not require materials or detectors that are sensitive to temperature (e.g. polyethylene to thermalise neutrons can melt at high temperature, detectors are sensitive, high temperatures pose a bias in measurements, temperature needs to be constant, implying a need for temperature monitoring included through SBD

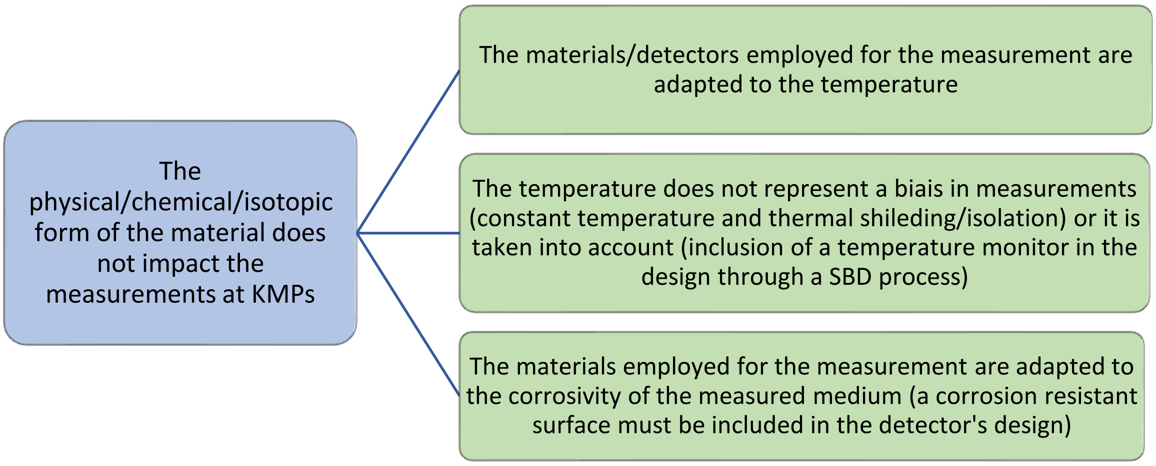


Figure 37: Evaluation of the NM forms' impact on the measurement accuracy at KMPs

EF1: The materials/detectors employed for the measurement are adapted to the temperature [0.8; 1]

Arrangements can be done through SBD

EF2: The temperature does not represent a bias in measurements (constant temperature and thermal shielding/isolation), or it is taken into account (inclusion of a temperature monitor in the design through an SBD process) [0; 1]

Design characteristics are missing if the design is not complete

EF3: The materials employed for the measurement are adapted to the corrosivity of the measured medium (a corrosion-resistant surface must be included in the detector's design) [0; 1]

Design characteristics are missing if the design is not complete.

The form/chemical/isotopic of the material does not impact the measurements at KMPs: [0.5;1]

5.3.2.7. Given a measurement accuracy, is the throughput of NM compatible with the detection goals?

EF1: There is a flow measurement device or a built-in accountancy tank [0; 1]

EF2: If the throughput is too high to reach the detection objectives, separation into multiple streams is done [0; 1]

It is unlikely, given the costs for the operator.

EF3: quantities lower than 1 SQ or less than an item are detectable [0; 0.5]

Given a measurement accuracy, the throughput of NM is compatible with the detection goals: [0; 0.5]

See MBA definition.

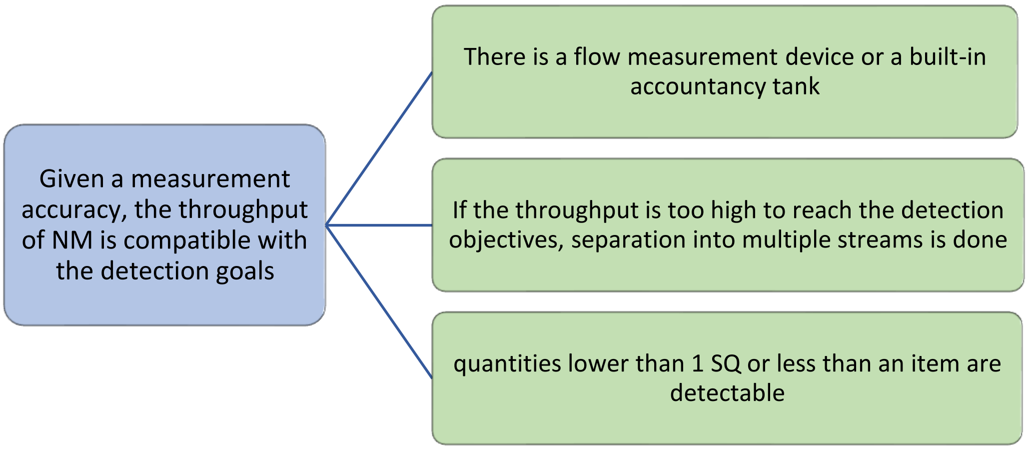


Figure 38: Evaluation of the throughput impact on the detection goals at KMPs

5.3.2.8. Are the uncertainties of detection equipment reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?

If a system can detect partial or even biased defects, it will be able to detect gross defects.

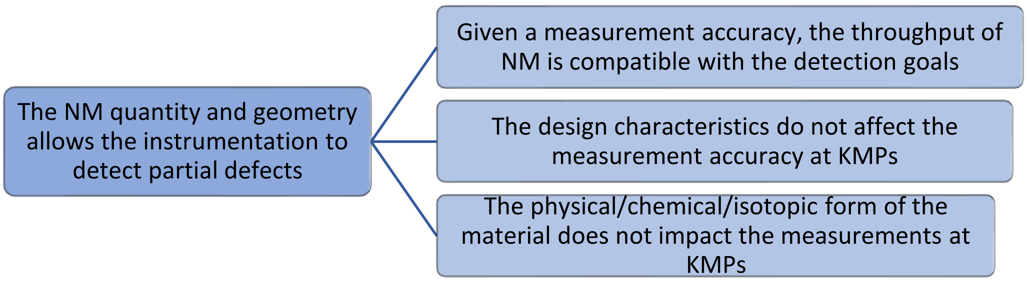


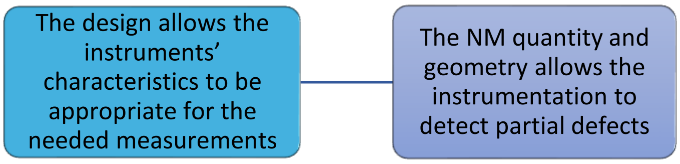
Figure 39: attributes needed for the evaluation of the NM characteristics' impact on the detection goals

1 and 2 are dependent (e.g. coating, plating on pipes); 1 and 3 are dependent; 2 and 3 are dependent (accuracy depends on the NM). All the daughter attributes are necessary for the success of the father; none is sufficient.

The literature and the evaluation of the daughter attributes hint at a concrete difficulty in partial and biased defect detection, especially given the high fissile inventories and high throughput.

The NM quantity and geometry allow the instrumentation to detect partial defects [0.3; 0.7]

5.3.2.9. Does the design allow the instruments' characteristics to be appropriate for the needed measurements?



This refers to the high-level evaluation of the fact that, given a set of instrumentation available, the design does not overly impact the performance of the instrument (which should be the ones that are fit for a similar plant). The design is the one being investigated through the SAT, which explains the wording.

The attribute is evaluated at the daughter attributes, and the flag is directly propagated by the experts, given the total dependency between the two attributes.

The design allows the instruments' characteristics to be appropriate for the needed measurements: [0.3; 0.7]

5.3.2.10. Is the calibration of NM measuring instruments by the inspector possible?

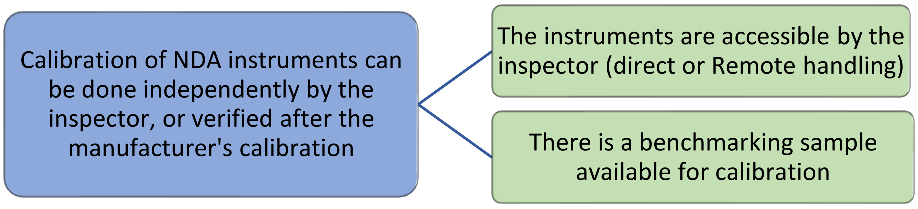


Figure 40: EFs needed for the evaluation of the possibility of NDA instrument calibration

EF1: The instruments are accessible by the inspector (direct or Remote handling) [0;1]

EF2: There is a benchmarking sample available for calibration [0; 1]

For both EFs, which are relative to design specificity, data are unavailable.

Calibration of NDA instruments can be done independently by the inspector, or verified after the manufacturer's calibration: [0; 1]

5.3.2.11. Does the design enable the instruments to be kept in the best working conditions?

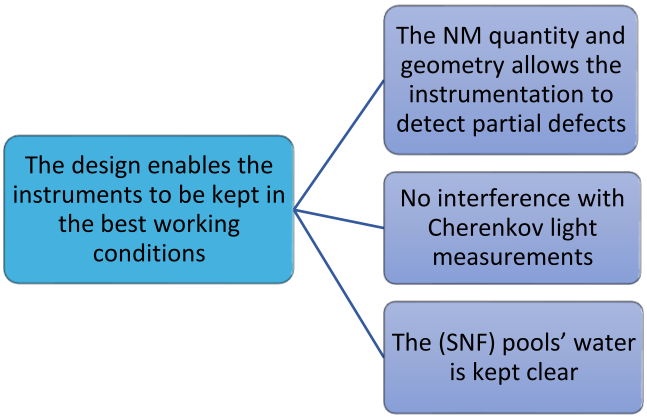


Figure 41: Attributes needed for the evaluation of the design's ability to accommodate the NDA instruments' needs

In LWR facilities, Cherenkov light measurements rely on the collimation of the Cherenkov light inside the channels created by the fuel assemblies in SNF pools. The MSFR will not have fuel assemblies or SNF pools. The two bottom attributes are non-applicable.

Attributes 1 and 2 and attributes 1 and 3 are independent, while attributes 2 and 3 are dependent. Attribute 1 is necessary and sufficient for the success of the father. The other two are necessary, if applicable, but with limited influence.

The design enables the instruments to be kept in the best working conditions: [0; 1]

5.3.2.12. Can the NM be accurately qualified/quantified by NDA?

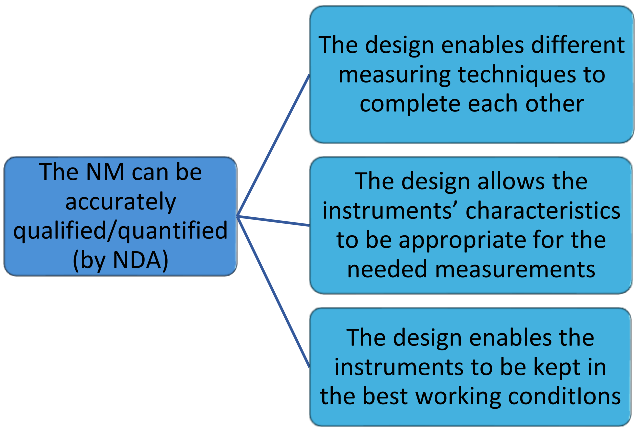


Figure 42: attributes needed to evaluate the possibility of accurately qualifying/quantifying the NM in the evaluated design

1 and 2 are dependent on each other; 2 and 3 are independent; 1 and 3 are independent. The relative importance is 2,3,1 (referring to the order of appearance)

2 is necessary to the success of the father, 3 is highly necessary, and 1 is less so. All attributes are equally sufficient.

The NM can be accurately qualified/quantified (by NDA): [0.3; 0.7] 

5.3.2.13. Is the NM in the facility accurately modelled (by the operator and verifiable by the inspector)?

Much of the modelling needed for safeguards purposes is also operational data needs for the operator, who is required to perform rigorous modelling according to common accuracy targets. It is expected that the modelling accuracy needs will be satisfied. The crucial question is whether the data can be trusted or reproduced by the inspector. This is why the attribute is evaluated at this high level only. Similar to the issues that arise with the measurement accuracy, it is expected that the modelling error margins, combined with the large inventories, will contribute to the inability to detect a bias defect. This hints at red in the flag.

The NM in the facility is accurately modelled (by the operator and verifiable by the inspector): [0.6;0.8] 

5.3.2.14. Is NDA applicable to the evaluated design?

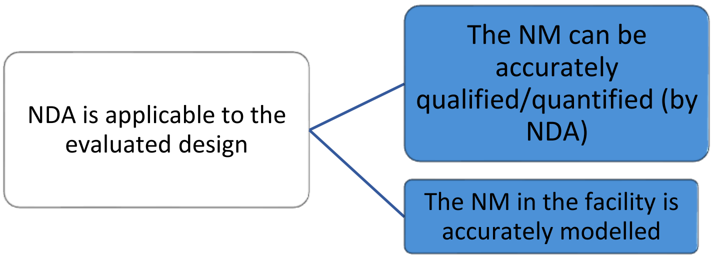


Figure 43: attributes needed for the evaluation of the NDA applicability to the evaluated design

Both attributes are necessary, and neither is sufficient for the success of the father attribute. They are dependent between them, and a success in modelling will help in achieving success in the measurements field.

NDA is applicable to the evaluated design: [0.4; 0.7] 

5.3.3. Interpretation of the Expert Elicitation on the Italian Flag methodology

The first impact left by the final Italian Flag is the strong green portion, seemingly against the literature presented in MSR: Safeguards Challenges and Opportunities. Given the early design phase of liquid-fuelled MSRs and especially given that the MSFR is a paper reactor for which facility peculiarities will not be available, as the reactor will not be constructed, a lot of white would be expected: there is a lot of incompleteness of the information about the applicability of NDA techniques and measurements to the MSFR operating on the U-Pu cycle. However, the semantic construction of the branches intrinsically reduces the white when the aggregation is done. For instance, given the many available measuring techniques, it is expected that at least one will be successful, leading to a successful branch of the tree. But most importantly, a safeguardability assessment makes sense when the safeguards approach is designed, which means that the reactor will be constructed. **Some of the evaluated aspects MUST be resolved to operate the reactor from a safety point of view** (such as what is described as "operational data need"). So, it makes sense to assume that a success would be achieved in these attributes before the safeguards approach would have to be designed, and to consider these EFs as green rather than white. Moreover, any facility can be effectively safeguarded, even with low efficiency (low safeguardability), by guaranteeing continuous inspector presence on-site. This aspect also supports the final green result.

Secondly, **the white part of the flag provides valuable information**. A success in modelling would improve the results in the measurements, and a lot of the uncertainty in the measurements comes from the unavailability of data from the modelling. Thus, the data having become available, the measurements can be interpreted. The uncertainty now comes from the trustworthiness of the models and the data. The negative perspective put forward by the literature can be partially explained by its focus on finding the gaps. **There is relatively "little" white in the final flag, but it is a crucial incompleteness of information that must be resolved to achieve a high safeguardability of the design.** Follow-up actions must be taken.

Finally, the red portion of the flag is also significant, but justified. **The red represents the inapplicability of some approaches, the inability of detecting partial or biased defects with the current ITV accuracy needs, and, substantially, the need for a paradigm shift in how the safeguards approach is conceived** for reactors sharing characteristics with the MSFR.

5.4. Mathematical propagation

The Italian Flags resulting from the mathematical propagation should be compared to the ones coming from the expert elicitation. The reasons for the discrepancies should be identified, this will be a follow-up action.

5.4.1. Pairwise Combination Rule

Pairwise combination is a mathematical framework used to merge uncertain evidence from two distinct sources into a single, unified body of knowledge. The fundamental process of pairwise combination involves creating a **contingency table**. If Source 1

(m_1) provides belief values for a set of hypotheses and Source 2 (m_2) does the same, their combination is determined by the intersection of those hypotheses:

- Grid Construction: The belief masses of Source 1 are placed along the rows, and Source 2 values are placed along the columns.
- Mass Multiplication: Every cell in the grid represents the product of the two corresponding belief masses ($m_1(X) \times m_2(Y)$).
- Assignment to Intersections: The resulting product is assigned to the intersection of the two sets ($X \cap Y$). For example, if Source 1 supports {Peter, Paul} and Source 2 supports {Paul, Mary}, the joint mass produced by their combination is assigned strictly to the overlap: {Paul}.
- When the intersection is the empty set, management of conflict techniques are used, described above.

5.4.2. Propagation through Yager's Combination Rule

Unless not applicable, Yager's propagation method is used, which maximises the uncertainty by reassigning conflict as total ignorance. The same bottom-up approach in the presentation of the three is kept.

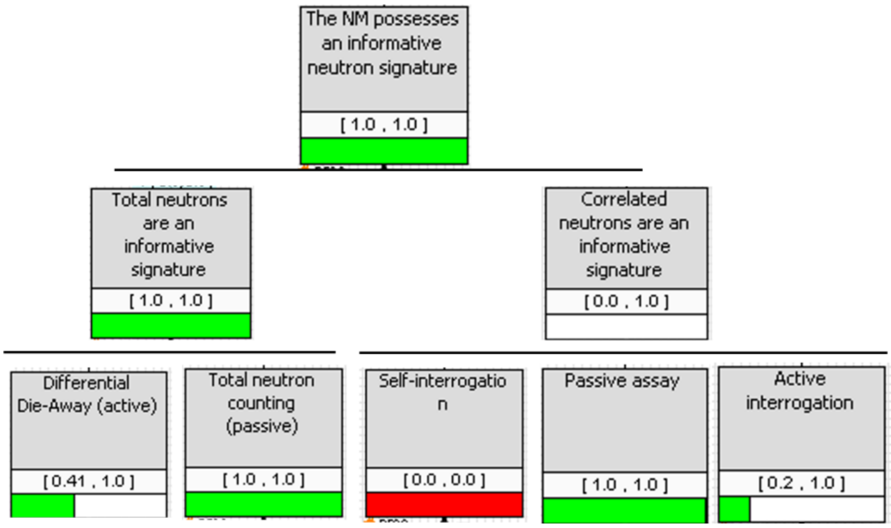


Figure 44: Evaluation of the informativeness of the neutron signature, Yager-propagated

The propagation method translates the conflicting evidence into uncertainty, which explains the flag associated with the informativeness of the correlated neutrons signature. However, this is countered by the semantic of the next father attribute, reducing the overall differences.

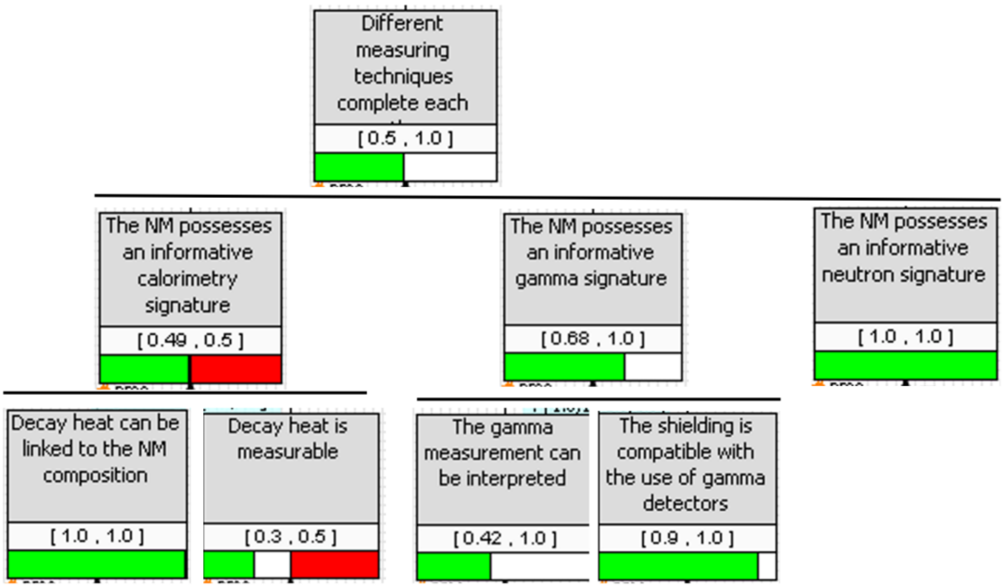


Figure 45: different measuring techniques complete each other, Yager-propagated

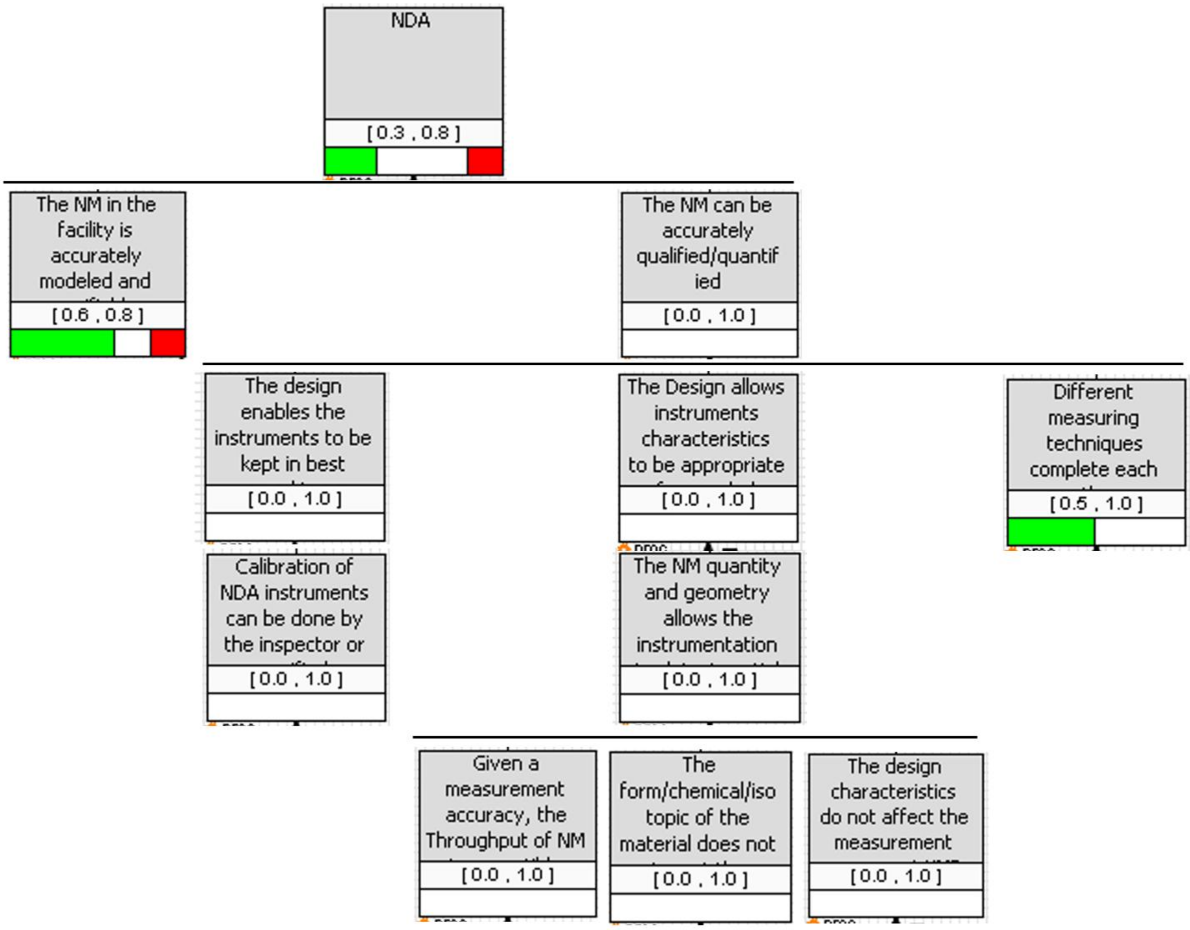


Figure 46: NDA is applicable to the evaluated design, Yager-propagated

5.4.3. Propagation through the Average of Pairs method

In the average of pairs method, the algorithm will average the green, white and red of the pairwise flags.

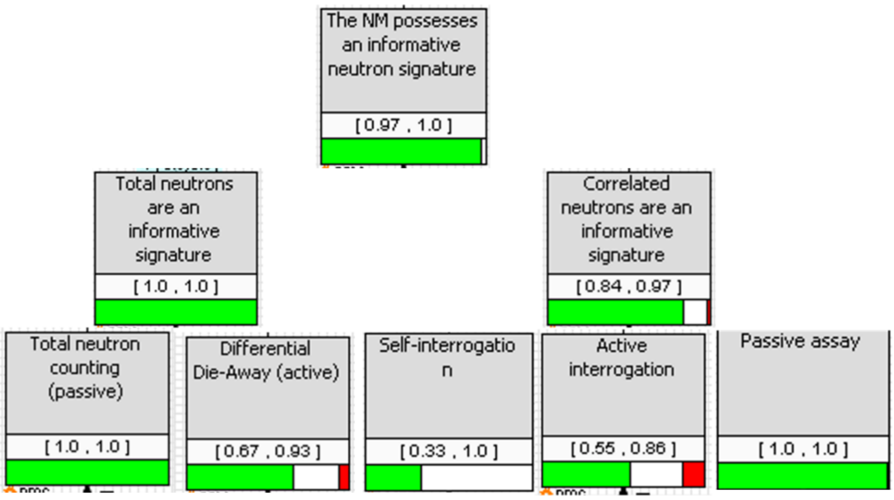


Figure 47: Evaluation of the informativeness of the neutron signature, average of pairs

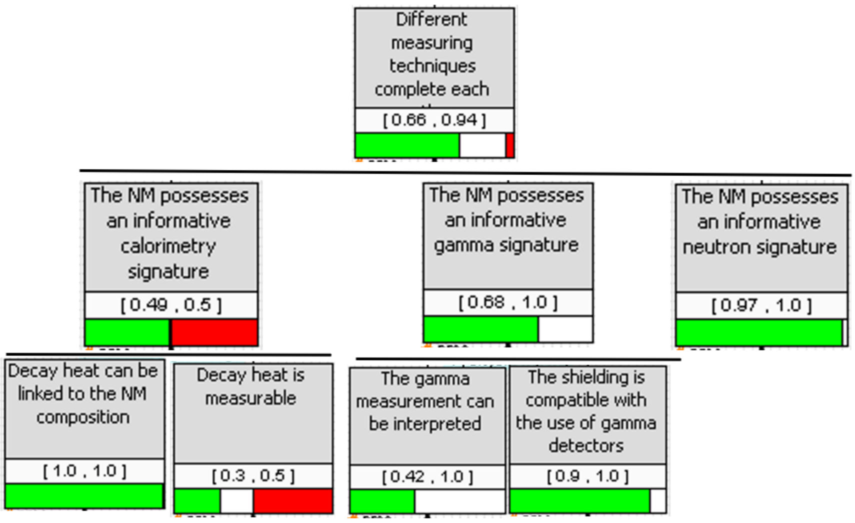


Figure 48: different measuring techniques complete each other, average of pairs

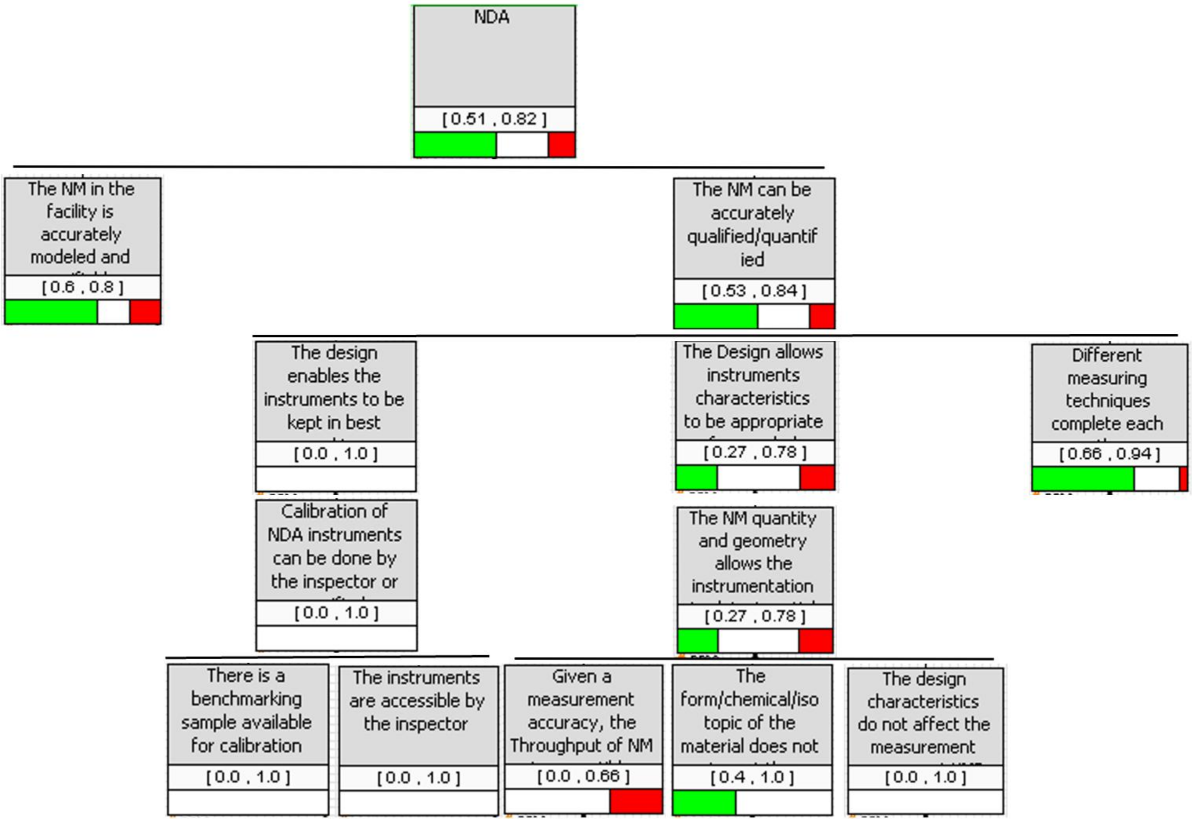


Figure 49: NDA is applicable to the evaluated design, average of pairs

5.5. Chapter's Summary

The chapter's goal has been to quantitatively evaluate the safeguardability of new designs. Reviews on the application of risk-informed methodologies to the PR studies have been done in the literature. For these studies, the safeguards' effectiveness can be regarded as one of the possible metrics. However, the application of risk-informed methodologies holds meaning only if the proliferation risk can be properly defined. It has been demonstrated that this definition is problematic both on the probability of occurrence of an event and the related consequences. System analysis approaches have been used instead, with the selected methodologies coming from the Decision theory or the Evidence theory. The MAUT's purpose is to provide a quantitative score to a given design by capturing the inspector's preferences regarding the design characteristics with respect to safeguards. However, the determination of the weights and evaluation parameters has been kept as a follow-up action. Instead, the application of the Italian Flag Methodology has been deemed more urgent given its usefulness in both refining the SAT's structure and directing R&D efforts. The methodology has been restricted to the evaluation of the applicability of the NDA techniques through the study of the MSFR. The remaining attributes should be evaluated on the same design to refine the structure, find the EFs, and establish a proof-of-concept and comparative basis. The results of the Italian Flag are globally positive and consistent with what was expected from the literature. A green majority represents the expected implementation of a safeguards approach, the red stands for the expected issues in biased defect detection and accuracy issues, while the white represents the urgent need for R&D efforts in the selected fields. There are some discrepancies, as expected, with the mathematically propagated Italian Flags, and the reasons for these differences should be identified in the future.

Conclusion and future developments

A substantial amount of effort has been put into presenting the reader with a comprehensive review of what the Nuclear Safeguards are and how they evolve in the complicated context of political shifts and technological advances. The first two chapters served as a benchmark for establishing how the different design characteristics of MSRs, particularly liquid-fuelled ones, impact the establishment of an effective and efficient Safeguards approach. The development of these nuclear reactors presents both unprecedented opportunities and significant safeguards challenges that fundamentally differ from traditional light water reactor paradigms. A thorough literature review has been performed to highlight the ongoing research on the Safeguards of MSRs. The main issues relate to the questioned applicability of the current NDA techniques, and the expected difficulty, if not impossibility, in achieving the detection and accuracy goals of International Safeguards. The main problems have been found to stem from the large fissile inventories tied to measurement and modelling inaccuracies. Actions must be taken to reduce the inaccuracies, but an operative reactor might be needed to obtain final operational data. Moreover, liquid-fuelled MSRs are suited to the Th-U cycle, for which some NDA techniques are not applicable, and the experience in safeguards is non-existent. On the other hand, an unprecedented opportunity to implement process monitoring and real-time accountancy is presented, for which the existing technologies from bulk-handling facilities should be adapted. The SBD approach has been proven to be the right approach to implement the desired safeguards objectives. To work towards the obtention of design information to assess the designs' safeguardability, a BTC tailored to liquid-fuelled MSRs has been developed and will be distributed to ENDURANCE's designer partners. A proxy for the successful implementation of SBD is the quantification of the design's safeguardability. This thesis has contributed to addressing the Safeguards challenges by establishing a systematic framework for assessing safeguardability that can be integrated into the design process from inception. Moreover, Safeguards-by-Design principles respond to the critical timelines of inspectorate instrumentation development, which can span up to a decade.

The Safeguardability Assessment Tool developed herein represents a consolidated effort to regroup fragmented safeguards knowledge from international literature, particularly from the International Atomic Energy Agency, Generation IV International Forum, and US National Laboratories, and to adapt these approaches specifically for the unique operational characteristics of the bulk-handling MSRs. Unlike historical safeguards practice, which was applied retrospectively upon design completion, the SAT enables prospective assessment during conceptual design phases when modifications can be incorporated without significant retrofit costs. The tool's layered architecture provides flexible assessment at multiple levels of detail, enabling both policy-level evaluations and granular inspector-focused assessments, thereby serving the diverse stakeholder needs within the safeguards ecosystem. This multiscale design philosophy reflects the thesis's recognition that safeguardability is fundamentally an emergent property arising from the interaction between hard technical systems and the soft institutional frameworks within which they operate.

The thesis took care of commenting on the definition of a proliferation risk and thus applied System Analysis approaches instead of Risk Analysis, specifically Multi-Attribute Utility Theory and Dempster-Shafer Theory through the Italian Flag methodology. The MAUT framework allows decision-makers to capture and quantify expert preferences (instead of their own) regarding which attributes most significantly contribute to effective safeguards implementation, thereby translating subjective expertise into measurable utility functions that respect both the traceability and aggregative logic essential for decision-making. This will help the designers in a safeguardability self-assessment. The Italian Flag methodology provides a structured means to quantify inherently qualitative expert judgments about safeguardability.

As a follow-up action, the MAUF must be defined. To help in this direction, work related to this thesis has been to find possible evaluation parameters and weighting factors. This is why the results of the studied Delphi survey have been ranked by score (Appendix E). Concurrently, the DST-based Italian Flag methodology addresses the persistent challenge of incomplete information and expert uncertainty that characterises the pre-conceptual design phase by explicitly distinguishing between confirmed evidence of success, uncertainty arising from incompleteness or conflicting information, and evidence of failure. The proof-of-concept application to Non-Destructive Assay instrumentation applicability for the Molten Salt Fast Reactor demonstrates that this methodology successfully propagates expert judgement through hierarchical dependencies and generates actionable knowledge.

However, current limitations and future research directions must be acknowledged. The SAT is currently evaluated primarily through expert elicitation, requiring further validation through application to concrete MSR designs at various development stages. The application of the Italian Flag methodology must be extended to the whole SAT, as the Expert Elicitation process will contribute to the validation and finalisation of the model. The propagation of uncertainty through complex dependencies remains sensitive to the choice of combination rules within Evidence Theory. Future work should identify the reasons for the discrepancies between the mathematical propagation and the expert elicitation propagation. The final Italian Flag identifies the need for R&D in NDA and further design characterisation, while also hinting at a need for a paradigm shift, as expected. Additionally, while the thesis successfully identifies that NDA applicability is contingent on design choices regarding fuel chemistry, geometry, and operational parameters, the trade-offs between safeguards implementation costs and design retrofits must be clearly quantified. Q**uantitative models linking specific design variables to measurable safeguards parameters are**

needed. Then, the EFs could be refined to obtain an implemented matrix of safeguards details for consideration**.** This is why, as a deliverable for the thesis, the steps in this direction have been taken to help include the matrix in the SBD process and safeguardability mindset (Appendix D). **Future work will systematically test the SAT against the ENDURANCE designs**.

The framework established in this Thesis provides a systematic **foundation for implementing safeguards-by-design principles in the emerging generation of advanced reactor technologies**, supporting both the IAEA and regional inspectorates in managing the expanding and diversifying nuclear fuel cycle while maintaining the credible assurance of non-diversion that underpins international non-proliferation commitments.

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A Appendix

If you need to include an appendix to support the research in your thesis, you can place it at the end of the manuscript. An appendix contains supplementary material (figures, tables, data, codes, mathematical proofs, surveys, …) which supplement the main results contained in the previous chapters.

A.1. Uranium Mining

Uranium deposits are mined to obtain uranium ore, historically from underground or open-pit mines. An increasing proportion of the world's uranium now comes from in situ leach mining, where oxygenated groundwater is circulated through a very porous orebody to dissolve the uranium oxide and bring it to the surface. This process extracts the uranium that coats the sand and gravel particles of groundwater reservoirs. The sand and gravel particles are exposed to a solution with a pH that has been slightly elevated by the addition of oxygen, carbon dioxide, or caustic soda. The uranium dissolves into the groundwater, which is pumped out of the reservoir and processed at a uranium mill [89]

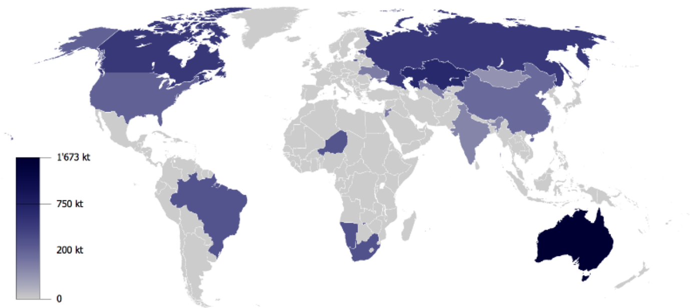
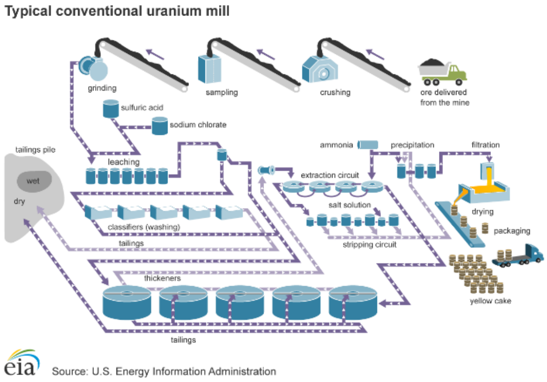


Figure 50: U reservoirs worldwide [89]

A.2. Uranium Milling

This is the process of extracting the U from the ore. The final product is a concentrate of U_3O_8 called yellowcake. This is typically done on the site of extraction or at a central mill when multiple mines are located near each other. The logistic advantage is clear. Mined uranium ore yields one to four pounds of U_3O_8 per ton of ore, and the concentration can go up to 80% from an initial concentration of U that can be as low as 0.01%. In a mill, the ore is ground and then leached in sulfuric acid (or an alkaline solution) to allow the separation of uranium from the waste rock. It is then recovered as a precipitate of U_3O_8. After drying and usually heating, it is packed in 200-litre drums, which is the so-called yellowcake. The remainder of the ore, containing most of the radioactivity and nearly all the rock material, becomes tailings. Tailings must be isolated from the environment due to their radioactivity and potential toxicity. However, the radioactivity is less and shorter-lived than in the original ore.



A.3. Uranium Conversion

The conversion of yellowcake to uranium hexafluoride (UF_6) is a critical stage in the nuclear fuel cycle, bridging the gap between uranium milling and enrichment. This process transforms raw uranium concentrate into a volatile compound suitable for isotopic separation. Initially, yellowcake is dissolved in nitric acid to form uranyl nitrate $UO_2(NO_3)_2$, which is then purified. Subsequent calcination yields UO_3, which is then reduced to uranium dioxide (UO_2) using hydrogen. The pivotal chemical transition occurs through hydrofluorination, where UO_2 reacts with hydrogen fluoride (HF) to produce uranium tetrafluoride (UF_4), commonly referred to as "green salt."

Finally, UF_4 is fluorinated with fluorine gas (F2) to form UF_6, a compound that sublimes directly to gas at a low temperature of 56.5 °C.

$UF_4 + F_2 \rightarrow UF_6$

The strategic importance of UF_6 conversion also underscores its role as a supply chain chokepoint, with only a few nations operating large-scale facilities, thereby placing it at the heart of global nuclear governance and security.

A.4. Uranium Enrichment

Uranium enrichment is the most sensible step from a proliferation perspective, particularly at the front end of the fuel cycle, and possibly the entire cycle.

Uranium found in nature consists largely of two isotopes, ^{235}U and ^{238}U. However, ^{238}U is not a fissile isotope, meaning it does not undergo fission and cannot be directly used towards the production of energy in nuclear reactors. The ^{235}U isotope, instead, is fissile, and it is this isotope that is taken advantage of for nuclear reactions. Natural uranium contains 0.7% of the ^{235}U isotope. The remaining 99.3% is mostly U-238. Isotope separation is a physical process to concentrate (ergo enrich) one isotope relative to others. The 1.27% difference in mass between U-235 and U-238, deriving from the three additional neutrons in the latter, allows the isotopes to be separated and makes it possible to increase or "enrich" the percentage of U-235. All present and historic enrichment processes, directly or indirectly, make use of this small mass difference.

There are two main technologies for uranium enrichment. Historically, diffusion and, more recently, centrifuges.

The diffusion method is a process that relies on Graham's law of effusion, which states that the rate of effusion of a gas is inversely proportional to the square root of its molecular mass. Since $^{235}UF_6$ is slightly lighter than $^{238}UF_6$, molecules containing ^{235}U pass through a porous barrier at a marginally higher rate. The key component is the diffusion barrier, a thin membrane with sub-micron pores of the order of magnitude of the free path of a UF_6 molecule. When UF_6 gas is pressurised on one side of the barrier, the lighter $^{235}UF_6$ molecules diffuse through the pores slightly more frequently, resulting in a stream that is minimally enriched in ^{235}U on the low-pressure side. The gas that does not pass through becomes depleted. A single barrier provides only a minimal separation effect. Therefore, the process is repeated across thousands of stages arranged in a cascade. The slightly enriched output from one stage is fed into the next, while the depleted stream is often recycled to previous stages. Despite its historical significance (it was used in the Manhattan Project and for early nuclear weapons), gas diffusion is an immensely inefficient process compared to modern gas centrifuge technology. It is extremely energy-intensive, requiring large plants and vast amounts of electrical power to compress the gas and overcome the pressure drop across the barriers. Due to this high operational cost and the ascendancy of centrifuge technology, gas diffusion enrichment is now largely obsolete.

The centrifuge is a vertically oriented, rotating cylinder that spins at an extremely high speed, ranging between 50,000 and 100,000 revolutions per minute. The rotation creates a centrifugal force that acts differently on the two isotopes, due to their mass difference. The heavier $^{238}UF_6$ molecules are pushed more strongly toward the outer wall of the rotor, while the lighter $^{235}UF_6$ molecules tend to concentrate closer to the centre.

A counter-current flow is established within the rotor to amplify this separation. As the gas spins, thermal gradients cause it to circulate vertically. The gas moving upward along the periphery, which is slightly richer in heavy molecules, and the gas flowing downward near the centre, which is slightly richer in light molecules, interact continuously. This counter-current flow acts like a cascade within a single rotor, significantly enhancing the separation effect. A single centrifuge can achieve a separation effect thousands of times greater than a single gas diffusion stage.

However, the output of one centrifuge is still only minimally enriched. Centrifuges are connected in series and parallel to form a cascade. The slightly enriched output from one centrifuge is fed as input to the next stage, while the depleted "tails" are recycled back to a previous stage. The number of stages may be limited to 10 to 20, rather than thousands or more, for diffusion. Centrifuges are designed to run continuously for approximately 25 years and cannot be slowed or shut down and restarted according to demand. Western cascades are intended for a 0.18 to 0.22% tails assay, while Russian ones are designed for a 0.10% tails assay. The sizing of the centrifuges can vary significantly, ranging from 12 meters for the American models to under a meter for the Russian ones. The gas centrifuge process is the dominant enrichment technology today due to its superior energy efficiency: it requires only about 5% of the energy consumed by the older gas diffusion method, requiring only about 40-50 kWh per SWU. This high efficiency, coupled with a smaller physical footprint, makes centrifuge plants more economical and difficult to detect, which also underscores their significant proliferation sensitivity.

The enrichment process is not a linear effort and is measured in separative work units (SWU). This is also the measure of the capacity of an enrichment plant. The measure is intricate but is generally affected by two factors: the desired enrichment and the concentration of U-235 in the tails.

Producing a kilogram of 90% U-235 requires vastly more SWU than for a kg of 4% U-235. The effort increases exponentially because, as the feedstock becomes richer, it becomes progressively harder to separate the remaining, less abundant U-238 atoms from the desired U-235 atoms. The process becomes less efficient in its later stages.

Having a lower tail assay means you are recovering more U-235 from the feed, making the process more efficient in its use of natural uranium. However, it increases the separative work required because it is extremely difficult to remove the last few U-235 atoms from the waste stream since the relative number of atoms of interest is very limited. An optimisation is usually done between the two. However, given that a weapon can be produced with a relatively small amount of material, the capacity of a plant producing feedstock for fuel fabrication (usually enriched between 3% and 5%) has similar SWU to a weapon-grade enrichment facility. (see image below).

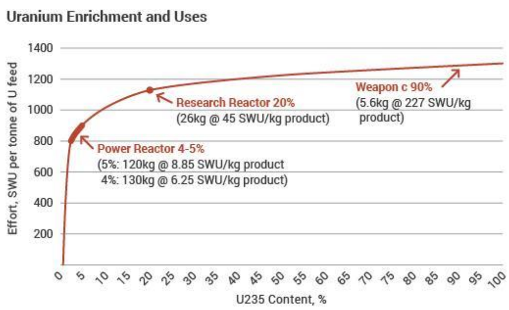


Figure 51: [90]

The graph illustrates how one tonne of natural uranium feed might be utilised: as 120-130 kg of uranium for power reactor fuel, as 26 kg of typical research reactor fuel, or potentially as 5.6 kg of weapons-grade material. In a somewhat counterintuitive way from the previous statements, the curve flattens out because the mass of material being enriched progressively diminishes to these amounts, from the original one tonne, so it requires less effort relative to what has already been applied to progress a lot further in percentage enrichment. The relatively small increment of effort necessary to achieve the increase from normal levels is the reason why enrichment plants are considered a sensitive technology in relation to preventing weapons proliferation and are tightly supervised under international agreements.

A.5. Fuel Fabrication

Since the vast majority of reactors are light water reactors (LWR) that use fuel pellets enclosed in fuel rods and grouped in fuel assemblies, the fuel fabrication paragraph is referring to this type of fuel for illustrative purposes.

Nuclear fuel fabrication is the industrial process that transforms enriched uranium UF_6 into solid ceramic fuel pellets, which are then loaded into metal tubes and assembled into robust fuel assemblies ready for irradiation in a nuclear reactor. The process begins with the conversion of UF_6 gas, typically delivered in cylinders, into uranium dioxide (UO_2) powder, often through an ammonia-based precipitation process. This fine, ceramic-grade UO_2 powder is then pressed into small, cylindrical pellets using high-pressure mechanical or hydraulic presses. These "green" pellets are subsequently sintered in a high-temperature furnace (approximately 1750°C in a hydrogen atmosphere) to densify them, resulting in a hard, stable ceramic with a high uranium density and a defined microstructure. This also reduces the UO2 to a stoichiometric compound (from UO_{2-x} to UO_2). The sintered pellets are then precision-ground to ensure uniform dimensions, which are necessary for consistent heat transfer.

The pellets are stacked into long, slender tubes made of a zirconium alloy, which are known as fuel rods or fuel pins. These tubes are filled with an inert gas (like helium) to improve heat conduction and to prevent internal corrosion, before being sealed with welded end plugs. The final step involves arranging hundreds of these fuel rods into a precisely spaced lattice structure, held together by top and bottom nozzles and intermediate grid spacers, to form a complete fuel assembly.

A.6. Reactors

There are hundreds of reactors operating worldwide, with dozens of variants in technology and design, as well as varying enrichment requirements. For instance, Canadian designs use natural uranium to avoid enrichment. This is the step in the uranium fuel cycle where U serves as a fission fuel. U undergoes inevitable transformations. Fission results in fission products (FP) that are lighter elements. An interaction between a neutron and a “fertile” isotope of U-238 will “breed” it into ^{239}Pu, which itself is a fissile isotope. Other interactions can yield other isotopes. The product of a nuclear reactor is spent (nuclear) fuel (SNF), which is a mix of isotopes of U, Pu, other minor actinides, and FP, which is highly radioactive.

The following figures may be regarded as typical for the annual operation of a 1000 MWe nuclear power reactor typical of many operating today, using 4.5% enriched fuel and with 45 GWd/t burn-up:[9]

Mining	Anything from 20,000 to 400,000 tonnes of uranium ore
Milling	249 tonnes of uranium oxide concentrate (which contains 211 tonnes of uranium)
Conversion	312 tonnes of uranium hexafluoride, UF_6 (with 211 tU)
Enrichment	35.9 tonnes of enriched UF_6 (containing 24.3 t enriched U @ 4.5%) – balance is 'tails' @ 0.22%
Fuel fabrication	27.6 tonnes UO_2 (with 24.3 t enriched U)
Reactor operation	8760 million kWh (8.76 TWh) of electricity at 100% output, hence 24 tonnes of natural U per TWh
Used fuel	27.6 tonnes containing 280 kg transuranics (mainly plutonium), 26 t uranium oxide (<1.0% U-235), 1 tonne fission products.

Figure 52: [90]

From the provided table, we can extract information on the quantity of fuel needed. A 1GWe (around 3GWth) will “create” 280kg of transuranic.

A key property of nuclear fuel is its burn-up, which quantifies the amount of energy extracted from a unit of fuel and is expressed in gigawatt-days per metric tonne (of heavy metal) (GWd/t). The enrichment level of the fuel directly influences this potential. Historically, the achievable burn-up was constrained by the mechanical durability of fuel assemblies, limiting values to approximately 40 GWd/t at enrichments near 4%. However, advances in materials and assembly design widely enable a burn-up of 55 GWd/t using 5% enriched uranium, with prospects of reaching 70 GWd/t using 6% enrichment. These higher burn-ups enable extended operational cycles and reduce the volume of spent fuel generated by approximately one-third. As a result, fuel cycle costs are projected to decrease by about 20%. A modern EPR with 5% enriched fuel and a burnup of 65 GWd/t would reduce the mass of spent fuel to 19.1t from 27.6t (table’s value)

A.7. Spent Nuclear Fuel (SNF)

Upon removal from the reactor, the SNF emits significant radiation (primarily from fission products) and substantial heat. For conventional LWR, it is immediately transferred to a water-filled storage pool located near the reactor. The water serves both to shield against radiation and to cool the fuel. Heat is removed via water circulation through external heat exchangers. Fuel is typically stored in these pools for several months to years before possibly being moved to on-site, naturally ventilated dry storage facilities.

Depending on national policies, some spent fuel may later be transferred to centralised storage sites. While interim storage is common and practical, the spent fuel issue must be addressed: either through reprocessing, where usable materials are recovered and recycled, or via direct long-term storage and eventual permanent disposal. Over time, radioactive decay reduces the fuel's radioactivity, making it easier to handle and lowering the associated hazards.

A.8. Reprocessing

The processing of used nuclear fuel, commonly referred to as reprocessing, is designed to separate valuable materials for recycling from the radioactive waste products generated during fission in a reactor. The primary goal is to recover over 95% of the original uranium and plutonium, which can then be reused to fabricate new fuel, notably Mixed Oxide (MOX) fuel, thereby significantly enhancing the efficiency of uranium utilisation and reducing the volume and long-term radiotoxicity of the final waste stream requiring geological disposal. The dominant industrial method employed is the PUREX (Plutonium Uranium Extraction) process, which involves dissolving the used fuel rods in nitric acid and then using a solvent extraction technique to selectively separate uranium and plutonium from the highly radioactive fission products and minor actinides. Following this separation, the recovered uranium can be re-enriched, and the plutonium is directly used in MOX fuel fabrication. At the same time, the remaining fission products are vitrified (immobilised in a stable glass matrix) for safe long-term storage. While reprocessing is undertaken in several countries, such as France, the UK, and Russia, as a cornerstone of their closed fuel cycle strategy, it is less sensitive in NWS. In the USA, there are no reprocessing facilities, and all the SNF is stored. In NNWS like Japan, it represents a significant economic and technical undertaking. It necessitates robust international safeguards to ensure the plutonium is used solely for peaceful purposes, making it a pivotal yet complex component of advanced nuclear fuel management. A proposed solution is the coextraction of Pu with U in oxides in the form of (U, Pu)O_2, of which relative isotopic concentrations can vary, instead of pure Pu.

Unlike conventional aqueous methods, pyroprocessing uses high-temperature electrochemical reactions in molten salts, such as chlorides or fluorides, to collectively recover uranium, plutonium, and minor actinides from spent nuclear fuel, leaving behind only fission products for conditioning into a stable waste form. This integrated recovery of actinides offers a notable non-proliferation advantage, as the resulting fuel mixture remains highly radioactive and self-protecting, and no Pu is isolated. On the other hand, pyroprocessing is easier to misuse as the separation power depends on the electrical potential, which can be easily adjusted, raising proliferation concerns. The process is compact, suitable for on-site application at reactors, and particularly well-suited for advanced systems such as fast reactors and molten salt reactors, where it supports the efficient recycling of all actinides without requiring precise isotopic separation. Although pyroprocessing can handle high-burnup fuel with short cooling times and operates within a simplified apparatus (combining reduction, electrorefining, and product consolidation), it remains at a relatively early stage of industrial development compared to established hydrometallurgical processes, such as PUREX.

A.9. Storage and Disposal

The management of radioactive waste from the nuclear fuel cycle is structured according to the waste's level of radioactivity and half-life. Waste is categorised as low, intermediate, or high-level, with each category requiring distinct handling, storage, and final disposal strategies. Low-level waste (LLW), which constitutes the bulk of the volume but contains only minor amounts of radioactivity, is typically compacted and safely buried in shallow landfill sites. Intermediate-level waste (ILW), often originating from reactor components and reprocessing sludges, requires more robust containment and is commonly immobilised in cement or bitumen within stainless steel containers, stored in engineered facilities at or near the surface while awaiting final disposal.

The most significant challenge lies with high-level waste (HLW), which contains the highly radioactive fission products and long-lived transuranic elements from a reactor's core. This category includes both the used fuel itself (if designated as waste) and the concentrated waste streams from reprocessing. For reprocessed waste, the current best practice is vitrification: the liquid HLW is calcined (heated and ground) into a powder and then incorporated into a molten borosilicate glass matrix. This glass is poured into durable, corrosion-resistant stainless-steel canisters. A year's waste from a 1000 MWe reactor is contained in approximately five tonnes of such glass, or around 12 canisters, each measuring 1.3 metres in height and 0.4 metres in diameter. These can be readily transported and stored with appropriate shielding. This process effectively immobilises the radioactive elements within a stable, solid form that is highly resistant to degradation and corrosion, making it suitable for long-term isolation.

Regardless of its final form, whether it be vitrified canisters or intact used fuel assemblies, all HLW generates substantial heat and radiation, necessitating initial management through interim storage. This is primarily achieved in two ways: wet storage in specially designed pools located adjacent to reactors, where water provides both cooling and shielding, and dry storage, where fuel that has cooled for several years is sealed in ventilated casks made of steel and concrete. These casks are stored in secure, monitored facilities on-site or at centralised locations. This interim stage allows for the radioactive decay of shorter-lived isotopes, significantly reducing the heat and radioactivity of the waste over several decades, which simplifies its eventual permanent disposal.

The universally accepted scientific and engineering solution for the final disposal of HLW is deep geological disposal. This method involves isolating the conditioned waste in a multi-barrier system located hundreds of meters underground within a stable geologic formation, such as granite, clay, or salt. This system is designed to contain the radioactivity until it has decayed to safe levels, potentially indefinitely. The barriers include the stable solid waste form (the glass matrix or the fuel itself), the robust sealed canisters, backfill materials like bentonite clay to limit water movement, and the surrounding host rock itself. Pioneering projects like Finland's Onkalo repository demonstrate that this technology is mature and capable of providing a safe, secure, and permanent solution for managing the most radioactive materials, thereby closing the nuclear fuel cycle in an environmentally responsible manner.

B

The brief description of Endurance's design is presented here. It is relevant to the thesis as some attributes are tailored to the Endurance's design. This is the case, for instance, for the attributes including the "cartridges" from Thorizon. The testing of the models has been done based on the MSFR.

Molten Salt Fast Reactor (MSFR)

The MSFR is a fast-spectrum breeder reactor with a large negative power coefficient and does not employ any solid moderator (no graphite lifespan issues). The initial version of the MSFR is based on a fluoride salt, achieving temperatures of up to 750°C for a volume of 18 m^3, and can be operated in a Thorium (Th) fuel cycle. The fuel salt is composed of LiF for 77.5 mol%, and a mixture of AcF3 and AcF4 for 22.5 mol% (Ac indicates actinides). The fission chain reaction requires a fissile driver. The fissile driver that would permit reaching isotopic equilibrium the fastest is U-233. This isotope is unavailable as of today. A more practical solution would be the use of a Pu driver, as it would employ the current PUREX reprocessing technique and facilities to recover an already available fissile resource. The MSFR would then initially operate as a Pu burner. The entire TRU vector with 5-year cooled TRUs from LWR used fuel could be used as a driver. Finally, U-235 is to be considered (A chloride-based version, named MSFR-Cl, can be operated in the Uranium fuel cycle, with a total volume of 45 m^3. Other versions for actinide management are being developed). The MSFR include features such as homogeneous fuel irradiation and the possibility of fuel reload and processing online or in batch mode, without requiring reactor shutdown. The external core structures and the fuel heat exchangers are protected by thick reflectors made of nickel-based alloys, which are designed to absorb more than 99% of the escaping neutron flux. These reflectors are themselves surrounded by a 20cm thick layer of B_4C, which protects from the remaining neutrons. The radial reflector includes a 50 cm-thick fertile blanket with a volume of 7.3 m^3. The employed passive gas bubbling has two objectives. First, removing metallic particles by capillarity and extracting gaseous fission products before their decay in the salt. The second part of the processing is a semi-continuous salt reprocessing at a rate of some tens (10-40) of litres per day, to limit the lanthanide and Zr concentration in the fuel salt. The salt sample is returned to the reactor after purification and after addition of ^{233}U and Th as needed to adjust the fuel composition and the redox potential of the salt by controlling the U4+ to U3+ ratio. Because this design will be used as a reference for the application of methodologies in the Thesis, design specifications will be included here.

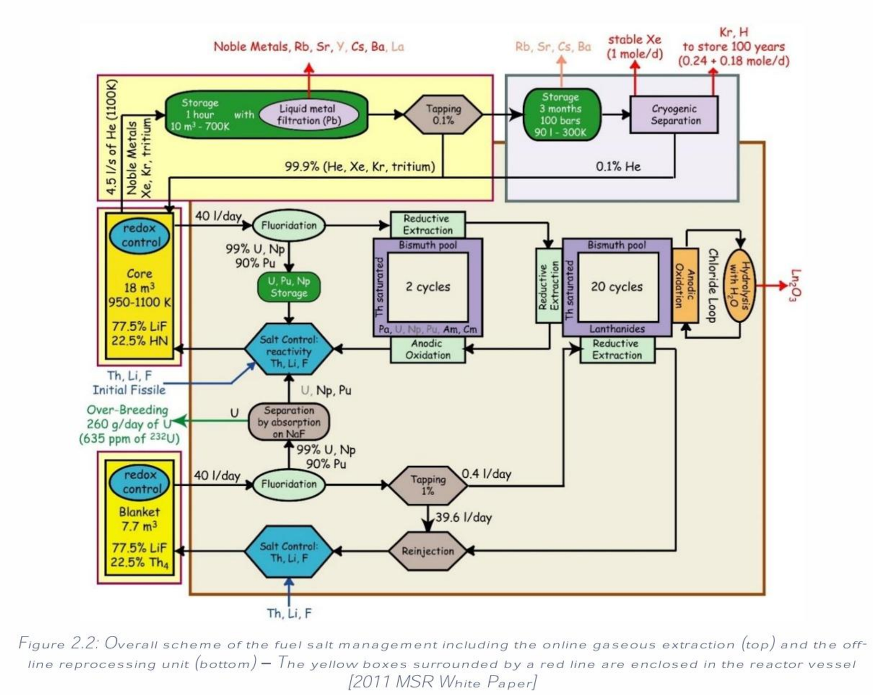


Figure 2.2: Overall scheme of the fuel salt management including the online gaseous extraction (top) and the off-line reprocessing unit (bottom) – The yellow boxes surrounded by a red line are enclosed in the reactor vessel [2011 MSR White Paper]

The salt flow schematics [49]

In the separate batch reprocessing unit, 99% of U (including ^{233}U) and Np and 90% of Pu are extracted by fluorination and directly reintroduced into the core. The remaining actinides are then extracted together with Pa and sent back to the core. In the last step, a second reductive extraction is performed by pyroprocessing to separate the thorium from the lanthanides, which are then sent to waste disposal. More than one MSFR could be co-located on the same facility, and reprocessing capabilities could be shared. [49] The isotopic composition, i.e. the isotopes that will have to be identified through safeguards measures, is listed below (after 1 year of operation).

Table 22: Main Isotopic Inventories for a 3GWth MSFR [49]

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

Isotope	^{233}U -started after 1 year	^{enr}U+TRU - started after 1 year	Fuel salt steady state 200 years	Fertile salt
^{232}U	3.5	142 g	13	34 g
^{233}U	4976	514	4658	58.5
^{234}U	143.9	12.8	1769	0
^{235}U	4.9	2506	510	0
^{236}U	0	149.5	562	0
^{238}U	0	16300	1	0
$^{232}U/U$	700 ppm	50 ppm	1700 ppm	600 ppm
$^{233}U/U$	97%	2.7%	62%	99%
^{238}Pu	0	239	161	0
^{239}Pu	0	3265	66	0
^{240}Pu	0	1617	57	0
^{241}Pu	0	641	48	0
^{242}Pu	0	491	10	0
$^{239}Pu/Pu$	0	52 %	19 %	0
		16		
^{232}Pa	3.9 g	0	15 g	15.4 g
^{233}Pa	124	45.6	108	13

An interesting PR feature of the MSFR is that when operated on a Th-U fuel cycle, it cannot be used to make plutonium usable for nuclear weapons MSFRs operated on a thorium fuel cycle cannot be used to make plutonium readily usable for nuclear weapons, only possible when Pu started if the initial Pu has less than 3 to 5% Pu-238, and not possible at all when ^{233}U-started (given the small amounts of Pu produced, which has a high ratio of Pu-238 [49]. However, international safeguards do not make such a distinction on Pu qualities.

ARAMIS

The ARAMIS concept is a 300 MWth fast-spectrum molten salt reactor. Its purpose is to burn actinides for inventory management. Its derived and more recent version, ARAMIS-A, aims at converting americium (Am) and Pu. A chloride salt was selected thanks to a higher solubility of plutonium than in fluoride salts, a harder neutron spectrum and the solubility of chloride salts in water, which would make the concept compatible with hydrometallurgical treatment and separation processes already used in France. The chlorine fuel salt is enriched in ^{37}Cl. The Pu recharge is 250kg/year or 31.25 SQs. The depleted salt requires treatment that aims to recover actinides, recycle chlorine and synthesise a new salt free of fission products that is re-injected into the reactor. [50]

The Seaborg Compact Molten Salt Reactor (CMSR)

The Seaborg CMSR is a small modular molten salt reactor characterised by the use of liquid fluoride molten salt fuel in direct contact with graphite moderator. The CMSR reactor core consists of several hundred graphite blocks arranged into columns in a central active core region and a periphery reflector region. The CMSR has a thermal power capacity of 250 MWth/100 MWe. It operates on standard LEU with enrichment below 5% U-235, utilising high-frequency online refuelling to achieve its 12-year operating lifetime while maintaining low excess reactivity and minimising control rod movement.

The peculiarity of the CMSR is that it is installed on a potentially mobile barge.

Copenhagen Atomics Waste Burner

The main objective of **Copenhagen Atomics** is to provide proof of the possibility to burn many of the long-lived actinides out of SNF and reduce the storage time to 300 years, while producing enough energy to pay for the process and decommissioning. It is heavy-water-moderated and designed for full automation, without requiring any human operator.

Stellarium

The Stellarium reactor is a Sodium chloride-based fast breeder MSR. The reactor is designed to provide both electricity and high-temperature steam, with applications to energy-intensive industries. The reactor utilises natural convection in the fuel vessel to carry heat from the core to the primary heat exchanger, reducing mechanical complexity and enhancing reliability. The core employs iso-reactivity, maintaining a stable reactivity balance over time due to in-core breeding and burning processes. This allows for long operational cycles of up to 20 years without refuelling. Iso-reactivity also reduces the need for reactivity control equipment. The reactor relies on inherent neutronic stability due to its strong negative temperature feedback. Any temperature increase reduces reactivity, leading to rapid, self-regulating behaviour. This feedback mechanism eliminates the need for emergency control rods. Instead, the reactor includes slow-acting control rods, primarily used to maintain subcriticality during shutdowns. The fuel salt contains a mixture of U, Pu, Th and minor actinides (MA), supporting a closed fuel cycle that minimises waste. The fuel has the following composition: NaCl at 55.0 mol%; $ThCl_4$ at 5.0 mol%; UCl_3 at 32.0 mol%; $PuCl_3$ at 8.0 mol%. Chlorine is enriched to 99% in 37Cl, U is Nat-U, and Pu is derived from spent MOX fuel. The SF Storage Reservoirs are designed for a total decay period of 12 months (6 months of filling and 6 months of decay).

Thorizon One MSR

The design of the fast-spectrum Thorizon One MSR is based on a modular core of multiple individually contained replaceable modules, named cartridges. Each cartridge consists of a double-layer vessel where the chloride fuel salt circulates, and the thermal power generated is exchanged with a secondary coolant salt. The ensemble of the cartridges makes up the core. When the cartridges are removed, the reactor has no primary circuit and no primary circuit components. The technology offers fuel cycle flexibility, with the first goal being to burn the SNF's Pu and commercialise, which can be followed by advanced versions on the same technology basis, in which U-breeding can be included, and fuel cycles can be closed.

Each of the 19 cartridges contains a specific amount of fuel salt, a mixture of NaCl with U, Th, and Pu chlorides. A Helium gas pocket is included to accommodate pressure build-up during operation. Power is generated in the fuel salt when it is circulated inside each cartridge. Since the gas pocket is relocated to a specifically designed plenum region in the middle of the cartridge when the pumps are operated, and the cartridge is designed, through the arrangement of the radial and upper reflector structures in the reactor cavity, and the neutron reflecting and absorbing material inside each cartridges, so that the criticality can be achieved in the upper part of the core. As soon as the pumps are stopped, the gas pocket rises at the top, and the core becomes subcritical. The cartridges making up the reactor core are closed systems and therefore, no purification, bubbling, or off-gas system is present during reactor operations.

XAMR

The XAMR (eXtra small Advanced Modular Reactor) is a small modular, molten chloride salt-fueled and -cooled fast neutron spectrum reactor with a reference nominal power of 80 MWth. The XAMR aims to burn the excess plutonium and minor actinides from the existing UOX and MOX cycles.

The fuel salt used is 395L of NaCl-(U, Pu)Cl_3 at the actinide trichloride fraction of the NaCl-PuCl3 eutectic, i.e. 36% mol. The XAMR is being designed for a variety of plutonium isotopic vectors, so the UCl3 fraction is adjusted to compensate for the reactivity of more favourable Pu and Cl isotopic vectors. A minimum fraction of 1 mol% UCl_3 is maintained for reasons of redox potential control in the fuel salt. The base salt would be recycled in the processing scheme to avoid losses of enriched chlorine in the waste stream. Fission products are extracted by pyroprocessing. The salt would be refuelled with plutonium actinides and shipped back to the reactor.

C Deliverable 1

The proposition for the adapted BTC tailored to liquid-fuelled MSRs can be found below.

	Safeguards relevant Information	Detail	Why is the information relevant for safeguards
IDENTIFICATION OF THE INSTALLATION	**facility name, location, address, owner, operator, status, purpose, etc.**	General information including the facility's purpose (research, power production, actinides management...) and status (TRL, conceptual, licensed, in construction, in operation ...). The owner and operator are the legally responsible bodies or individuals	Who is going to be the site representative (interface responsible for the facility's operation)? The status of the project provides an indicative schedule to define the safeguards approach. "Purpose" is to have an idea of the general operations performed in the plant.
	site layout	Consider the whole site overview with description of where the different installations, and potential interfering facilities (H2 storage...) are. Site map showing the installation, boundaries, buildings, roads, rivers, railways, etc.	To consider the broader context of the installation for a 3S approach.
	installation description, including general flow diagrams	Describe the flow of material within the facility. Use nodes and diagrams without specific locations	To define the Material Balance Area (MBA) and key measurement points. To consider the flow of NM to inspect/study diversion paths, 3S considerations
	Layout of the installation (incoming and outgoing material and storage, laboratories, waste disposal)	Detailed description of the installation (structural containment, fences, and access routes; incoming material storage, reactor building), including auxiliary facilities (test and experiment area, laboratories). A map representing the exact locations of the previously mentioned nodes of the flow diagram	to consider the flow of NM to inspect/study diversion paths, 3S considerations
	rated thermal output	MWth	to picture the plant's size, and for burnup calculations and irradiation simulations

	electrical output	MWe	to have an idea of the efficiency of conversion (BOP) in the system
	number of units (reactors)		Important to know how many units are hosted in the plant/facility, and thus how many units are to be put under safeguards
	reactor type		
	moderator		To know the range of neutron spectra, and to know about the accessibility of NM for verifications. The moderator could be opaque.
	coolant		To know about the accessibility of NM for verifications. The coolant could be opaque or needs to be operated in an inert atmosphere.
	Blanket, reflector, control rods	Provide details about the blanket and reflectors' location and structure	to know whether tampering/clandestine production is possible within non-fuel components. To have info for neutronics calculations and irradiation.
	type of refuelling	off-load, on-load, item vs. quasi-bulk	To define the C/S features (particularly to grasp the sealability of the vessel) and the inventory change features that affect the accountancy method. The goal is to define the inspection approach.
description of nuclear material	**the reactor core's U enrichment range**		The main point is to know whether the reactor operates with LEU (<20%) or HEU, understand the proliferation risk of the whole fuel cycle. The subsequent set of safeguards measures and strategies, particularly on timeliness of inspections, is determined by the proliferation risk.
	Pu concentration (initial)		

	types of fresh fuel	high-level description of the fuel physical form and content (bulk salt, MOX? Enriched? flexible?)	
	fresh fuel enrichment (235u) and/or Pu content	Provide a description of the various types of fresh fuel that can be used, particularly if the fissile% can vary	info about the flexibility of the reactor, indicating the possibility of misuse and possible clandestine production
	nominal weight of fuel in elements or assemblies	mass of the fuel salt	to characterise the fuel element in bulk
	Physical and chemical forms of fresh fuel	description of incoming fuel "packaging". How is the salt in the form it is received? And the physical form in which it is introduced in the reactor? What is its chemical composition?	
	reactor structural functioning details (eg, Cartridge or bulk fuel as coolant)	namely, bulk salt flowing or cartridge	
	description of fresh fuel elements: e.g. chemical form, dimensions, number of pellets, cladding, bonding (when applicable)	How is the fuel received in the facility? Description of incoming fuel "packaging" and physical/chemical attributes	
	Basic accounting units (e.g. fuel elements, assemblies)	method for accountancy for the NM? in kgHM	
	means of nuclear material identification	Applicable only to cartridges.	
	means for checking the composition of the fuel salt	Most important for facilities with online salt reprocessing. Provide information on the measuring point specific to the salt composition check and reprocessing. How much salt is extracted and for how long? How is it reintroduced?	It is a crucial point for the safeguardability of the system. Information is needed to study the potential for seal applications, online NDA investigations, or DA.
flow of nuclear material	**Schematic flow sheet identifying measurement points, storage/inventory locations**		Of the previously identified NM, it is important to know its location and intended movements. We have to know any access points to the salt and NM.
	expected inventory/capacity (including storage)		What and how much NM is present inside and outside the reactor, and why, where. It is to be inspected.
	other nuclear material in the facility	consider fresh and spent fuel, and any other NM	to have an inventory of all the NM present within

			the installation, outside the core
	reactor input/output flow rate or batch size (cartridge) and frequency	salt inflows and outflows	
	reactor load factor		for burnup calculations and temperature/operating parameters monitoring
	reactor vessel details, core diagram, flow diagrams (drawings)		crucial to know how the core is organised, and the NM within it. Any access points, etc. With this information, the inspector can have an idea of what is enclosed and not accessible and can thus be sealed for safeguards purposes.
	initial reactor core loading	NM initial inventory, enrichment, minor actinides	neutronics and simulations. Identification of possible misuse.
	average neutron flux in core (thermal and fast)		for irradiation simulation for spent fuel
	burnup (average, maximum)		to estimate the isotopic composition of the irradiated fuel for NDA purposes. Residual heat in the 3S approach
	refuelling details (quantity of nuclear material, time interval, throughput)	How often, how much, and with what is the reactor refuelled? Describe the refuelling operation	Is it possible to implement an item counting/NM quantification on the refuelling and defueling?
	description of the salt recycling/discard stage	Describe the fuel back-end. With what methodology is the salt reprocessed (pyroprocessing...) or discarded? stored? shipped?	
Handling of nuclear material	**nuclear material handling details**	handling, transfer, and loading equipment and machinery for fresh and spent fuel.	relevant for knowing what machines are available to the inspector, what fuel is thus accessible for verification, and what operations could be performed on the fuel onsite.
	Instrumentation for measuring neutron and gamma flux	What NDA techniques, if any, are used for online monitoring of the operations?	is there a possibility for shared information, or a need to implement independent

			complementary measures?
	Irradiated fuel details	chemical, physical, and isotopic composition. Quantity.	safeguards and safety measures are to be taken on SF
	whether the irradiated fuel is to be processed or stored	It can be a mix of the two	To know the intended operations on the SF, especially if there is an actinide isolation
	nuclear material testing areas, including equipment available, and shipping containers	What operations can be performed on the SF directly on site? Can fuel elements be opened and accessed? can fissile material be isolated?	how to maintain continuity of knowledge in the facility? what operations can be performed on the NM? Can fissile material be isolated?
	Basic measures for physical protection	security considerations	3S approach
	Basic measures for radiation safety (engineering aspect), including rules for inspector compliance (radiation protection)	e.g., radiation in the environment? rules of compliance with the operator's rules	Personnel safety
	Description of nuclear material accountancy and control	How is the accountancy of NM performed by the operator? are there any controls in place by the operator? This is intended as a bureaucratic accountancy and control	To verify compliance with international requirements and methodology
	Optional information that the operator considers relevant to safeguards	Any information that can help the inspector to apply safeguards: adapt NDA and organise inspections	Facilitate the safeguard process by providing additional input as the designer/operator of the reactor that you consider relevant.

D

The IAEA's matrix of safeguards details for consideration adapted to the SBD phases can be found below [62]. Legend: attributes for increasing the effectiveness of the safeguards approach; attributes for increasing the efficiency of the safeguards approach; attributes for increasing the ease of designing the safeguards approach.

Phase 1: Basic Design Features (Intrinsic to Process)		
Design Information Verification (DIV)	Nuclear Material Accountancy (NMA)	Containment & Surveillance (C/S)
In-core fuel verification capability	Adequate space, support, and illumination to handle, identify, and conduct measurements on fuel	Pond layout for complete monitoring/surveillance
Supportive facilities (office space, changing rooms)	Maintain water quality & clarity in pools	Select equipment locations that protect equipment
The system allows for unambiguous Design Information Verification (DIV) throughout life cycle [49]	NDA verification in SFP for all types of fuel present	Select equipment locations to minimize impact on operations
The system provides for the installation of measurement instruments, surveillance equipment and supporting infrastructure likely to be needed for verification [49]	Minimize nuclear material storage times at reactor, if possible	Clearly label safeguards-relevant equipment
No locations in or near the core of a reactor where undeclared target materials could be irradiated [49]	Avoid UV light interference with Cherenkov imagers	Physical protection of seals from accidental damage
The system makes the use of operation and safety/related sensors and measurement systems for verification possible, taking in to account the need for data authentication [49]	Water clarity maintained to facilitate identification and measurements of assemblies	Ability to seal fuel transport paths/equipment
Protect sensitive or proprietary information	Unique identifiers or labels for nuclear material items	Redundant/battery-backed systems
	Easy to read identifiers or labels for nuclear material items (and adequate illumination)	Support for unattended and remote monitoring
	Tamper-proof identifiers or labels for nuclear material items	Sealing nuclear material in core (e.g., the missile shield or reactor slab)

	Authenticated identifiers or labels for nuclear material items	Easy C/S to Used Fuel Storage & Shipping Areas
	Inspector use of automated readout of location and assembly being moved	Minimize access points in barriers around nuclear material
	Storage racks in a single layer permits viewing of top of each assembly/readout of identifier	Adequate illumination and adequate viewing angles
	Layout of storage of fuel to allow progressive verification and sealing of groups of assemblies without affecting nuclear material already under seal	Provision for transfer cask/flask monitoring equipment
	Arrange fuel in store to minimize need to move fuel to access specific assemblies	Provision for monitoring spent fuel transfers
	Provision for fresh MOX segregation	Infrastructure at dry storage for containment, monitoring, surveillance, including layout & design of casks/silos
	MOX fresh fuel containers sealing arrangement	Provision to seal lower layers in fuel store, if multiple layers
	Fresh MOX fuel loading monitoring capability (surveillance from delivery till start-up reactor)	Authentication of data (sensors)
		The core is not accessible during reactor operation [49]
Phase 2: Safeguards Equipment Installation		
Design Information Verification (DIV)	Nuclear Material Accountancy	Containment & Surveillance (C/S)
Fuel assembly reconstitution verification		Provision of structures for maintaining sensors (ladder, platform)
In-core fuel verification capability		Suitable mounting for permanent and temporary sensors (camera)
Core opening monitoring (e.g., verify assembly positions during refueling)		
Facilitate inspector activities		
Provide access for inspectors to verify barriers (e.g., containment integrity)		
Provision of supportive facilities and services (changing rooms, inspector's office space, etc.)		
Minimize need to revisit site to resolve questions		

Phase 3: Safeguards Efficiency and Effectiveness (during operation)		
Design Information Verification (DIV)	Nuclear Material Accountancy	Containment & Surveillance (C/S)
	nuclear material accountancy systems will be able to verify that any MUF is within the range allowable by the IAEA [47]	
	The facility does not require overly-frequent inspections (material type, refuelling frequency…) [16]	
	The cost for the IAEA is standard for this type of facility (reactor, reprocessing..) [16]	
	Ability to collect data on site, but analyse off site	
	Facilitate transmission of safeguards information off site (separate network, inspector)	
	Nuclear materials remain accessible for verification to the greatest practical extent	

E

The safeguardability parameters were assigned an importance score through a Delphi survey among a panel of experts in [73], and have been ranked to provide an initial path for the weighting of the SAT attributes during the MAUF composition. The order of appearance of the branches in the table is random, but note that when evaluating the relative importance of the safeguards branches, NMA accounts for 48%, compared to 25% for DIV and 27% for C/S.

C/S	27. Is it possible to install a containment device? (including seals to protect the device against damage and power supply)	4,77
C/S	35. Can surveillance equipment be installed? (including surveillance equipment to protect the device against damage, visibility obstruction, power supply, and lighting)	4,77
C/S	29. Can inspectors access the containment and surveillance equipment for installation and verification? (access path of inspectors for attachment or detachment of seals and radioactivity levels)	4,68
C/S	30. Is the containment structure (such as walls) designed to have no path (hole) through which nuclear materials can pass?	4,64
C/S	33. Does the containment structure (such as walls) is not designed to be drilled after construction?	4,5
C/S	34. Are the nuclear material movement paths and frequencies standardized to facilitate containment and surveillance?	4,36
C/S	37. Is surveillance equipment installation, such as power or communication cables and lighting, considered in the design?	4,18
C/S	36. Is there an independent or auxiliary power supply, such as an uninterruptible power supply (UPS) dedicated to surveillance equipment?	4,09
C/S	39. Can an independent network be established to transmit safeguard information?	4,05
C/S	31. Can inspectors verify the integrity of the containment structure (such as walls) of the nuclear material storage facility?	4
C/S	28. Is there an independent or auxiliary power supply, such as an uninterruptible power supply (UPS), for the containment device?	3,95

C/S	32. Can the installation of sealing cables be facilitated in the containment structures (such as walls) of the nuclear material storage facility?	3,95
C/S	38. Can containment and surveillance equipment be connected online for verification in real time?	3,91
DIV	1. Is the design information completed in accordance with the IAEA DIQ format?	4,86
DIV	9. Are the initial, final DIQ, and major changes in the design information submitted in a timely manner?	4,73
DIV	8. Are documents such as layout and drawings of a nuclear facility managed accurately and systematically?	4,36
DIV	4. Can inspectors access the nuclear facility to confirm the change in design information during the life of the facility?	4,32
DIV	3. Can inspectors access the nuclear facility during the construction process for visual verification?	4,09
DIV	6. Is there any equipment or information that restricts access to inspectors due to security reasons? a) Is there any equipment or information that restricts access to inspectors for safety or security reasons? b)	4
DIV	2. Can inspectors access essential equipment during the operation of the nuclear facility for visual verification?	3,95
DIV	5. Are the radioactivity levels at the access point and the route of the inspector minimized during design information verification?	3,91
DIV	7. Can inspectors use 3D scanners for design information verification at the nuclear facility? a) Can inspectors use the latest technology such as 3D scanners for design information verification at the nuclear facility? b)	3,23
NMA	22. Does the nuclear facility periodically calibrate nuclear material measuring instruments and manage the uncertainty of measuring instruments?	4,73
NMA	10. Can non-destructive analysis (NDA) equipment from the IAEA be installed in a nuclear material storage facility for verification? (space for installing, power cable, and evaluation of the safety impact of the nuclear facility)	4,68
NMA	18. Is there a tag or label attached to the nuclear material for item identification?	4,68

NMA	20. Are the tags or labels for identifying nuclear materials designed to prevent easy removal or alteration and to maintain readability over a long storage period?	4,64
NMA	11. Can inspectors access nuclear materials to confirm NDA verification? (appropriate space, lighting, access path, and radioactivity levels)	4,55
NMA	16. Does the nuclear material storage facility have sufficient space and lighting for inspectors to count items?	4,45
NMA	12. Is it possible to install sampling equipment for destructive analysis (DA) in the nuclear material process or storage at the nuclear facility? (sampling port, sampling equipment space for installation, power supply, and evaluation of the safety impact of the nuclear facility)	4,41
NMA	19. Is there an index attached to the nuclear material storage facility to easily locate the items?	4,36
NMA	17. If nuclear materials are stored in two or more layers, can inspectors apply safeguard measures to verify the bottom layer or install seals?	4,23
NMA	21. Can assembled nuclear material items be dismantled or reconstructed in the nuclear facility?	4,23
NMA	15. Is there sufficient space to store NDA and DA safeguards equipment at the facility? (possibly for installing seals or monitoring equipment)	4,18
NMA	14. Is the accessibility of inspectors secured for sampling verification? (appropriate space, lighting, access path, and radioactivity levels)	4,18
NMA	13. Is the independence of the sampling equipment for safeguard measures achieved in the nuclear material process? (exclusive sampling port for inspection, system for transferring inspection samples, and storage facilities for inspection samples)	4,09
NMA	23. What is the annual throughput of nuclear material in a nuclear facility?	3,91
NMA	26. Can real-time nuclear material measurement and accounting systems be established?	3,82
NMA	25. What is the radiation dose rate of the nuclear materials?	3,18
NMA	24. What is the heat generation rate of the nuclear materials?	2,82

F

The SAT, as it stood at the time of completing the thesis, is presented below. The SAT is comprised of 5 Evaluation Sheets (given its adaptation from a MAUF usage deriving from [28]). The Evaluation Sheet for NMA is presented first, given its more advanced state, presenting a layered structure and the enabling factors for part of it. Further work is needed but the proof-of-concept is consolidated. Note that the SAT reported here only presents three columns, namely the attribute and the dependencies with the other attributes, to reconstruct the decision tree, and the rationale for the attribute. The references to the literature for the attribute and the rationale are not reported in this thesis.

EVALUATION SHEET FOR NMA		
Rationale	Dependencies (or Comment)	Attribute
Does the design allow for multiple verification methods for the inspector? Some techniques are faster (NDA passive that is always operable vs. NDA active that requires interrogation of the sample vs. DA that requires sampling, perhaps shipping …). NDA is also much cheaper, impacting DE for the inspectorate.		What equipment can the inspector use for achieving detection objectives?
	What equipment can the inspector use for achieving detection objectives?	Can the NM be accurately qualified/quantified by NDA?
	Can the NM be accurately qualified/quantified by NDA?	Does the design enable different measuring techniques to complete each other (reverification and robustness of the system)?
Does the design allow for multiple verification methods that complement each other? it must be related to a simulation of expected heat from a declaration of irradiation history. But even if the measurement is consistent, it must be complemented with at least CoK, since it is an integral measure and it does not inform on the emitting isotope (a clever replacement of a higher emitting isotope with a large quantity of	Does the design enable different measuring techniques to complete each other (reverification and robustness of the system)?	Does the NM possess an informative neutron signature?

low-emission isotopes, e.g. U235 replaced by Cm)		
Does the design allow for multiple verification methods that complement each other? An intrinsic material characteristic that contributes to the difficulty of concealing the nuclear material radiation signature for identification purposes. It makes the recognition easier. Reducing the proportion of fissile Pu additionally increases PR as a whole by reducing material attractiveness.	Does the NM possess an informative neutron signature?	Do the correlated neutrons from spontaneous fission (SF) constitute a recognisable and informative signature, through passive assay?
The emissions of Cm are orders of magnitude greater than other Pu isotopes, which hides the signature	Does the correlated neutrons from spontaneous fission (SF) constitute a recognisable and informative signature, through passive assay?	The Cm/Pu ratio is known
There is a need for modeling of the isotopic mix	Does the correlated neutrons from spontaneous fission (SF) constitute a recognisable and informative signature, through passive assay?	Need info about the SF-emitting isotopes
	Does the correlated neutrons from spontaneous fission (SF) constitute a recognisable and informative signature, through passive assay?	Even isotopes of Pu are present
Does the design allow for multiple verification methods that complement each other?	Does the NM possess an informative neutron signature?	Do the correlated neutrons from the induced fission of an active assay constitute a recognisable signature?
	Do the correlated neutrons from the induced fission of an active assay constitute a recognisable signature?	need for info about fissile isotopes mass/quantities
	Do the correlated neutrons from the induced fission of an active assay constitute a recognisable signature?	need for a low neutron environment so that the results of active interrogation are measurable

	Do the correlated neutrons from the induced fission of an active assay constitute a recognisable signature?	need for a neutron source (criticality risk assessed through SBD)
The introduction of a radiosource is always problematic from a radioprotection point of view. Moreover, the criticality risk must be assessed.	Do the correlated neutrons from the induced fission of an active assay constitute a recognisable signature?	The added-value of using an active interrogation technique has to be adequate (much more precise, time-efficient, or practical) to justify the management of the radiosource
Does the design allow for multiple verification methods that complement each other?	Does the NM possess an informative neutron signature?	Do the correlated neutrons from a self-interrogation of the item constitute a recognisable signature?
	Do the correlated neutrons from a self-interrogation of the item constitute a recognisable signature?	need for information about fissile isotopes other than Pu
	Do the correlated neutrons from a self-interrogation of the item constitute a recognisable signature?	Need for (alpha, n) reactions that create free neutrons so alpha-emitting material and light-elements in the chemical composition are needed
	Do the correlated neutrons from a self-interrogation of the item constitute a recognisable signature?	Because the interrogant neutrons have to be thermalised and refelcted into the item, there is a need for a suitable geometry and surroundings
	Do the correlated neutrons from a self-interrogation of the item constitute a recognisable signature?	The signature is not hidden by the 242Cm emission
Can inform about irradiation history (thus isotopic composition related to the general amount of undergone fissions), measurement of neutron-emitting actinides (burnup is usually measured through Cs, a FP. Cm is short-lived and its activity can directly inform about the irradiation history and allow to distinguish between a same level of burnup obtained through different irradiations)	Does the NM possess an informative neutron signature?	Does the total neutrons count obtained through passive measurement constitute a recognisable signature?
irradiation history (burnup+cooling periods; which can inform us about the supposed composition of the item)	Do the total neutrons count obtained through passive measurement constitute a recognisable signature?	need for info about the irradiation history
	Do the total neutrons count obtained through passive measurement constitute a recognisable signature?	need even isotopes presence

Can inform about irradiation history (thus isotopic composition related to the general amount of undergone fissions), measurement of neutron-emitting actinides (burnup is usually measured through Cs, a FP. Cm is short-lived, and its activity can directly inform about the irradiation history and allow for distinguishing between the same level of burnup obtained through different irradiations).	Does the NM possess an informative neutron signature?	Does the total neutron count obtained through an active measurement technique (such as the differential die-away) constitute a recognisable signature?
(different from active interrogation because there the source can be any neutron source, and you have only one source, no dependency)	Does the total neutron count obtained through an active measurement technique constitute a recognisable signature?	pulsated neutron generator needed
	Does the total neutron count obtained through an active measurement technique constitute a recognisable signature?	The added value of using an active interrogation technique has to be adequate (much more precise, time-efficient, or practical) to justify the management of the neutron generator
	Does the total neutron count obtained through an active measurement technique constitute a recognisable signature?	fissile (uneven) isotopes presence
EP3.1.2: Inspectors measurement capability: does the facility allow for multiple verification methods for the inspector? Signature = The intrinsic material characteristics that contribute to the easiness of identifying/recognizing the nuclear material type and composition.	Does the design enable different measuring techniques to complete each other (reverification and robustness of the system)?	does the NM (fresh, spent, operational) possesses an informative gamma spectrometry signature?
	does the NM possess an informative gamma spectrometry signature?	Is the shielding compatible with the use of gamma detectors
Can a measurement be made?	Is the shielding compatible with the use of gamma detectors	The shielding does not prevent the measurement or, if the NM is shielded in a way that prevents measurements, adequate penetrations are foreseen
	Is the shielding compatible with the use of gamma detectors	representativeness of the measurement is acknowledged (penetrability of the signal in the assayed material)
	does the NM possess an informative gamma spectrometry signature?	Can the gamma measurement be interpreted?
	Can the gamma measurement be interpreted?	U-235, Pu, FP peaks in the spectra can be related to irradiation history

No covering by other FP (that cannot be related to the irradiation history). Peaks can be resolved by managing the interferences between them. Higher resolution instruments or better software could be needed.	Can the gamma measurement be interpreted?	The spectra can be deconvoluted
The NM can be accurately qualified/quantified by NDA. Does the design allow for multiple verification methods that complement each other? Calorimetric analysis is an NDA technique commonly used for verification purposes. This attribute relates to the possibility of applying such a technique inside the Nuclear Energy System.	Does the design enable different measuring techniques to complete each other (reverification and robustness of the system)?	Does the NM possess an informative calorimetry signature?
Calorimetry applicability is necessary but not sufficient; it must be related to a simulation of expected heat from a declaration of irradiation history. But even if the measurement is consistent, it must be complemented with at least CoK, since it is an integral measure and it does not inform on the emitting isotope (a clever replacement of a higher emitting isotope with a large quantity of low-emission isotopes, e.g. U235 replaced by Cm)	Does the NM possess an informative calorimetry signature?	Is the decay heat measurable so that it constitutes a calorimetry signature?
	Is the decay heat measurable so that it constitutes a calorimetry signature?	The assayed material is static
	Does the NM possess an informative calorimetry signature?	Can the decay heat be linked to the NM composition?
The NM can be accurately qualified/quantified by NDA. Does the facility allow for multiple verification methods for the inspector? Calorimetric analysis is an NDA technique commonly used for verification purposes. This attribute relates to the possibility of applying such a technique inside the Nuclear Energy System.	Is the decay heat measurable so that it constitutes a calorimetry signature?	Heat is linked to the composition of the SNF

The NM can be accurately qualified/quantified by NDA. Does the facility allow for multiple verification methods for the inspector? Calorimetric analysis is a NDA technique commonly used for verification purposes. This attribute relates to the possibility of applying such technique inside the Nuclear Energy System.	Is the decay heat measurable so that it constitutes a calorimetry signature?	Heat generated by Alpha decay linked to the actinides' mass on fresh fuel
This refers to the high-level evaluation of the fact that, given a set of instrumentation available, the design does not overly impact the performance of the instrument (which should be the ones that are fit for a similar plant). The design is the one being investigated through the SAT, which explains the wording	Can the NM be accurately qualified/quantified by NDA?	Does the design allow the instruments' characteristics to be appropriate for the needed measurements?
here it is not the design that's evaluated but the instrumentation, which is useful for the inspectorate's side of the SBD process	Does the design allow the instruments' characteristics to be appropriate for the needed measurements?	are the uncertainties of detection equipment reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?
Needs to be evaluated through the evaluation of the daughter attributes NMAC 18, 19, 21	are the uncertainties of detection equipment reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?	quantities lower than 1 SQ or less than an item are detectable
The signal can be too weak or too strong and the shielding has to be adequate. Similarly, the system characteristics (piping and the related plating) can impact the measurement accuracy, intensity of the signal, representativity of the measured layer, which should be taken into account	are the uncertainties of detection equipment reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?	3.1.1 Does the design affect the measurement accuracy at Key Measurement Points (KMPs) through design characteristics? (e.g. piping thickness tied to the operative pressure, or required shielding degrading the measuring capabilities)
	3.1.1 Does the design affect the measurement accuracy at KMPs through design characteristics?	0.82cm (<1cm) of lead thickness for reducing the intensity of the signal by half for Cs-137 (of which the signal is stronger by a factor of 10^5)
	3.1.1 Does the design affect the measurement accuracy at KMPs through design characteristics?	Dead-time effect

3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of physical inventory taking (PIT) by the operator or the effectiveness of physical inventory verification (PIV) by the IAEA? See MBA definition challenges	are the uncertainties of detection equipement reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?	3.1.1 Does the design affect the ability to reach the detection goals, given a fixed accuracy at KMPs through geometry/throughput? (e.g. the reprocessing unit flow is too high which impacts the ability to reach the detection goals given a certain uncertainty)
	3.1.1 Does the design affect the ability to reach the detection goals, given a fixed accuracy at KMPs through geometry/throughput?	There is a flow measurement device or a built-in accountancy tank
	3.1.1 Does the design affect the ability to reach the detection goals, given a fixed accuracy at KMPs through geometry/throughput?	quantities lower than 1 SQ or less than an item are detectable
	3.1.1 Does the design affect the ability to reach the detection goals, given a fixed accuracy at KMPs through geometry/throughput?	If the throughput is too high to reach the detection objectives, separation into multiple streams is done
	Does the design affect the ability to reach the detection goals, given a fixed accuracy at KMPs through geometry/throughput etc.?	Does the radiation exposure limit the instrumentation operational life (beyond what is reasonable for resource efficiency)
	Does the radiation exposure limit the instrumentation operational life (beyond what is reasonable for resource efficiency)	The sensitive electronics can be shielded when the detectors have to remain exposed
The penetrability of the radiation depends on the form of the material (liquid, gas, bubbling...) because of the density change. The mass absorption is the same but the absolute absorption changes with density. Liquid state often implies a high temperature: the needed measurement techniques does not require materials or detectors that are sensitive to temperature (e.g. polyethylene to thermalise neutrons can melt at high temperature, detectors are sensitive, high temperatures pose a biais in measurements, temperature needs to be constant,	are the uncertainties of detection equipment reasonable to detect partial defects, given the NM quantity and its geometry, in a timely manner according to IAEA detection goals?	3.1.4. Does the design employ nuclear material types, categories, or forms (physical/chemical/isotopic) that are more difficult to measure accurately at KMP?

implying a need for temperature monitoring included through SBD		
	3.1.4. Does the design employ nuclear material types, categories, or forms that are more difficult to measure accurately at KMP?	The materials/detectors employed for the measurement are adapted to the temperature
	3.1.4. Does the design employ nuclear material types, categories, or forms that are more difficult to measure accurately at KMP?	The temperature does not represent a biais in measurements (constant temperature and thermal shileding/isolation) or it is taken into account (inclusion of a temperature monitor in the design through a SBD process)
	3.1.4. Does the design employ nuclear material types, categories, or forms that are more difficult to measure accurately at KMP?	The materials employed for the measurement are adapted to the corrosivity of the measured medium (a corrosion resistant surface must be included in the detector's design)
	Can the NM be accurately qualified/quantified by NDA?	Does the design enable the instruments to be kept in the best working conditions?
	Does the design enable the instruments to be kept in the best working conditions?	Is the calibration of nuclear material measuring instruments by the inspector possible? (e.g.: hot cells accessibility)
	Is the calibration of nuclear material measuring instruments by the inspector possible? (e.g.: hot cells accessibility)	The instruments are accessible by the inspector (direct or Remote handling)
	Is the calibration of nuclear material measuring instruments by the inspector possible? (e.g.: hot cells accessibility)	There is a benchmarking (sealed) sample available for calibration

Lighting frequency and intensity, including underwater lighting, should meet the minimum specifications of selected C/S equipment for lighting all areas subject to surveillance or inspection. Sources of illumination within areas subject to surveillance should be designed to avoid impeding surveillance by equipment or IAEA inspectors.	Does the design enable the instruments to be kept in the best working conditions?	is UV light interference with the collimation of the Cherenkov light in the assemblies channels?
Water clarity in the spent fuel storage area should be sufficient to permit operation of Cerenkov radiation monitoring/ viewing instrumentation.	Does the design enable the instruments to be kept in the best working conditions?	Can water quality & clarity in pools be maintained to facilitate identification and measurements of assemblies/SNF?
MODELING		
	What equipment can the inspector use for achieving detection objectives?	Is the NM in the facility is accurately modeled (by the operator and verifiable by the inspector)?
	Is the NM in the facility is accurately modeled (by the operator and verifiable by the inspector)?	Has the depletion and activation/burnup been modeled and independently benchmarked? (both in the reactor and in storage)
	Has the depletion and activation/burnup been modeled and independently benchmarked? (both in the reactor and in storage)	identifiable peaks directly linked to burnup
	identifiable peaks directly linked to burnup	Cs-137, Eu-154, U-235 ... peaks
Each process has a physical amount of nuclear material that is not accessible during inspection activities, because,e.g., the material may be inside process pipelines. A system optimised to minimise this amount of material would facilitate the closing of the material balance by the inspectorate. Having a MUF less than a SQ significantly improves the detection goals and objectives	Is the NM in the facility is accurately modeled (by the operator and verifiable by the inspector)?	What is the amount of material unaccounted for (MUF) for the facility and is it duly taken into account?
	What is the MUF for the facility and is it facility duly taken into account?	Are the MUF uncertainties are accurately estimated?
Expected measurement uncertainty δE (relative standard deviation) associated with closing a material balance is mandated by the IAEA.	Are the MUF uncertainties are accurately estimated?	Are the uncertainties within the allowable range?

	Are the uncertainties within the allowable range?	EP3.1.1: Pu or U Sigma(MUF)/SQ
The inspector does not necessarily have access to all the instrumentation available to the designer. The UMI/MUF have to be quantifiable even with limited instrumentation in an independent manner	Are the MUF uncertainties are accurately estimated?	is it possible for the inspector to reproduce the MUF estimation in an independent manner with the available instruments?
To reduce uncertainty on nuclear material that cannot be presented to inspectors for verification (assessment of the 'hidden inventory'). UMI will necessarily be part of MUF and has to be quantified accurately by means that are not measurements (modelling...).	What is the MUF for the facility and is it facility duly taken into account?	3.2.3 Does the design create the potential for Un-Measureable Inventory (UMI) (or hidden inventory) in locations such as pipes, pumps, or evaporators?
	What is the MUF for the facility and is it facility duly taken into account?	Can the Hold-up inventory be modelled at any point in time (between cleaning actions)?
	Is the NM in the facility is accurately modelled (by the operator and verifiable by the inspector)?	Are the FPs accurately modelled?
	Are the FPs accurately modelled?	Are the design's FPs removals accurately modelled?
	Are the design's FPs removals accurately modelled?	Has the gaseous removal of FP been modeled with independently benchmarked models?
	Has the gaseous removal of FP been modelled with independently benchmarked models?	The gaseous extraction system does not separate NM
	Are the design's FPs removals accurately modeled?	Is the noble metals removal been modeled and independently benchmarked?
	Are the design's FPs removals accurately modeled?	Is the rare earths removal been modeled and independently benchmarked?
similar to UMI but not for NM		Are the FPs accumulations and buildups accurately modeled
	Are the FPs accumulations and buildups accurately modeled	Is the noble metal plating and buildup in elements such as filters and pipes been modeled and independently benchmarked?
	Are the FPs accumulations and buildups accurately modeled	Is the rare earth plating and buildup in elements such as filters and pipes been modeled and independently benchmarked?

	Is the NM in the facility is accurately modeled (by the operator and verifiable by the inspector)?	Is the fissile material evolution modeled accurately?
	Is the fissile material evolution modeled accurately?	Is the radioactive decay modelled at all locations of the NM?
The NM is temporarily out of the neutron flux	Is the radioactive decay modelled at all locations of the NM?	was the decay of NM in the reprocessing units modelled with independently benchmarked models?
The NM is temporarily out of the neutron flux	Is the radioactive decay modelled at all locations of the NM?	was the decay of NM in the storage after reprocessing modelled with independently benchmarked models?
	Is the fissile material evolution modeled accurately?	Is the removal rate of fissile isotopes in the online chemical separation unit modelled?
Material balance evaluations require quantification of the nuclear material added to the system during refuelling. The method and frequency with which salt is added will determine what technologies are practical for the IAEA to use to monitor fuel additions.	Is the fissile material evolution modelled accurately?	Is the addition of fissile fresh fuel through on-load fresh fuel top-ups modeled?
	Is the fissile material evolution modeled accurately?	Is the addition rate of bred fertile from the blanket to the core modeled?
	What equipment can the inspector use for achieving detection objectives?	Can the NM be accurately qualified/quantified by DA?
3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA? In the source, it is limited to "at the time of PIV" due to the cognitive bias in which the PIV is always performed during downtimes at bulk handling facilities. This could not be the case in newer facilities, especially if real-time accountancy is implemented	Can the NM be accurately qualified/quantified by DA?	3.1.5. Does the design preclude or limit the ability of the IAEA to analyse (at the site) independent samples for verification?= Samples can be collected for DA measurement at any time

	Samples can be collected for DA measurement at any time	Is radiation managed at any time?
MEASURING POSSIBILITIES		
		3.1.3. Does the design preclude PIV measurements on some inventory? If so, does the design include features to permit CoK to be maintained from a previous measurement and verification? (= Is all the NM accessible for verification?)
3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA? Meaning accessibility of all NM for measurements and verification	Is all the NM accessible for verification?	Is the fresh fuel accessible for verification? (e.g. minor actinides radiation/MOX)
	Is all the NM accessible for verification?	Is the Irradiated fuel accessible for verification?
	Is the Irradiated fuel accessible for verification?	3.1.2 Is the SNF accessible for verification?
accessibility of all NM for measurements and verification. Irradiated fuel is accessible. 3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA?	Is the Irradiated fuel accessible for verification?	is there a provision for in-core fuel verification capability?
accessibility of all NM for measurements and verification. Irradiated fuel is accessible.	Is all the NM accessible for verification?	Is the MUF reduced as much as possible?
	Is the MUF reduced as much as possible?	can the hidden inventory (UMI) be quantified accurately with measurements? Is it stable over time?
UMI cannot be reduced by cleaning operations	Is the MUF reduced as much as possible?	Can the Hold-up inventory be measured at any point in time (between cleaning actions)?

	Samples can be collected for DA measurement	3.1.5. Does the design preclude or limit the ability of the IAEA to take independent samples at any time for DA verification?
3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA? In the source, it is limited to "at the time of PIV" due to the cognitive bias in which the PIV is always performed during downtimes at bulk handling facilities. This could not be the case in newer facilities, especially if real-time accountancy is implemented	3.1.5. Does the design preclude or limit the ability of the IAEA to take independent samples at any time for DA verification?	Is the sampled material homogeneous?
if a sample is extracted from the bulk, is that sample representative of the whole salt, or multiple samples have to be taken? What means are there to guarantee homogeneity	3.1.5. Does the design preclude or limit the ability of the IAEA to take independent samples at any time for DA verification?	Is radiation managed at any time?
DESIGN IMPACT ON THE PIV'S EFFICIENCY		
Item-based accountancy avoids the problems of the MUF and should always be implemented, when possible, even if this means breaking the consistency of the accounting method throughout the facility. This could refer to the cartridges' encapsulation		Is item always preferred to bulk when possible?
		Is the accountancy system defined throughout the facility?
	Is the accountancy system defined throughout the facility?	Is the number of MBAs kept to a minimum?
	Is the number of MBAs kept to a minimum?	Are the boundaries between the MBAs clearly defined? Exist in physical form?
	Is the number of MBAs kept to a minimum?	Does the accounting method take into account the transition between MBAs?
	Does the accounting method take into account the transition between MBAs?	Is the CoK maintained between MBAs

	Is the number of MBAs kept to a minimum?	3.1.1 Does the design impede the use of Key Measurement Points (KMP)? If so, are there other well-defined locations that could be considered by the inspectorate as KMPs?
Even if the safeguards instrumentation and approach is responsibility of the IAEA, provision for space and auxiliary equipment is responsibility of the designer	3.1.1 Does the design impede the use of Key Measurement Points (KMP)? If so, are there other well-defined locations that could be considered by the inspectorate as KMPs?	Is there a provision to install all the necessary measuring equipment for previously mentioned NDA techniques?
THE ACCOUNTANCY SYSTEM IS DEFINED THROUGHOUT THE FACILITY	Is the accountancy system defined throughout the facility?	Is the sampling/measuring scheme is defined and fit-for-purpose?
THE ACCOUNTANCY SYSTEM IS DEFINED THROUGHOUT THE FACILITY. This would lead to different measurement methods, easier and less expensive saefguards approach, and additional verification methods	Is the accountancy system defined throughout the facility?	Is there a possibility to use shared equipment for NMAC and MC&A
shared equipment and joint effort	Is there a possibility to use shared equipment for NMAC and MC&A	Is there a provision for data sharing and verification of data by the operator?
shared equipment and joint effort	Is there a possibility to use shared equipment for NMAC and MC&A	Is the periodic calibration of nuclear material measuring instruments planned by the operator? Can the calibration be done or verified independently?
For some processes the closure of material inventory for timeliness purposes is particularly challenging. Having the possibility to implement near-realtime accounting would greatly help the inspectorate in closing the material balance frequently and without interrupting theprocess, in order to achieve timeliness objectives while not intruding in the system's operation	Is the accountancy system defined throughout the facility?	is it possible to establish a Near realtime accountancy system
	is it possible to establish a Near realtime accountancy system	Is process monitoring implemented?
	Is process monitoring implemented?	Can the reactor operation temperature be monitored to detect diversion through a decreased reactivity?
	Is process monitoring implemented?	Is the fuel salt composition monitoring implemented?

The key point here is that the NM has to be directly accessible when it is accessible 3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA?		3.1.2.NM is directly accessible for verification (no need for moving the NM for measurement due to stacking, for instance)
The radiation levels are compatible with efficient NMA (ease of access, ease of completing tasks under shielding…)	3.1.2.NM is directly accessible for verification (no need for moving the NM for measurement due to stacking, for instance)	Inspectors are adequately protected from radiation throughout the inspection
3. Are there design specifics that make it difficult to verify that diversion has not taken place by reducing the efficiency or effectiveness of PIT by the operator or the effectiveness of PIV by the IAEA? lower relevance because some movement cannot be avoided if the time between the two events is long		3.1.6. Does the design prevent inventory change/movement between the time of the PIT and the PIV? If so, does the design include measures to maintain CoK of the changed or moved inventory between the time of the PIT and the PIV?
		Do the PIVs, when absolutely necessary, match as much as possible already planned downtimes?
	Do the PIVs, when absolutely necessary, match as much as possible already planned downtimes?	the frequency at which components are replaced is defined
	The frequency at which components are replaced is defined	The replacements are scheduled

EVALUATION SHEET FOR THE LEGAL FRAMEWORK		
Rationale	Dependencies (or Comment)	Attribute
see Volume 5 of the Final Report of Phase 1 of the INPRO vol5 annexe A.3.1. "Indicator IN1.1 States' commitments, obligations and policies regarding non-proliferation to fulfil international standards " and the subsequent evaluation parameters		International Agreements
See previous rationale		Legal Framework
See previous rationale		Competent Authority
See previous rationale		Import/Export Control
See previous rationale		Regional Cooperation

Recidivism risk can be considered. It is PR, but it can help the decision maker with the export of designs. See Volume 5 of the Final Report of Phase 1 of the INPRO EP1.1.10		Recorded violation of non-proliferation commitments.
this is a direct measure of the cost/effort efficiency for the implementation of a safeguard system		Is the Detection resource efficiency (DE) comparable to the DE of similar-sized plants?

EVALUATION SHEET FOR MATERIAL CHARACTERISTICS		
Rationale	Comment	Attribute
The material quality impacts the timeliness of detection and detection goals		Material Quality
The more SQ are present in the facility, the less detectable a diversion would be. Eg, for a core containing exactly one SQ, the whole core would have to be diverted to obtain one SQ. However, this is a highly detectable event.		Material Quantity

EVALUATION SHEET FOR DESIGN INFORMATION		
rationale	Comment	Attributes
DESIGN INFORMATION COLLECTION		
		On what fuel cycle does the design operate?
Higher enrichment allows for more precise measurements (intensity of the signal)		What's the U enrichment? Does the design use HALEU?
		does the design allow for fuel cycle switching? If yes, are the timeliness objectives consistent when switching cycles?
The physical and chemical form of the SNM will affect the material attractiveness, which will affect the difficulty of various diversion scenarios of SNM via theft. Physical form identifies whether the SNM is in item form (e.g., in containers, fuel tubes) that can be individually counted or in bulk form. The physical and chemical form of the SNM will also influence the measurement techniques and equipment that can be used to quantify the SNM.	UO_2, UF_4; liquid, solid; ...	What's the chemical form and state of the fuel?

In the design, is there the potential to alter existing diversion paths?		1.1. Does this design introduce nuclear material of a type, category, or form that may have a different significant quantity or detection time objective than previous designs/processes (e.g., mixed oxide rather than low-enriched uranium, irradiated vs. unirradiated or bulk rather than items)?
The rationale here is that the more diluted the fissile fuel is, the more salt will have to be diverted for obtaining an SQ, which complicates the task and makes detection of diversion easier. This is the equivalent of "How many assemblies/elements for an SQ" in traditional item-counting facilities.		Mass of bulk material for SQ (kg)
Radiological hazard that has an impact on accessibility to areas where nuclear material is kept, but can make containment and surveillance easier.		If the MSR is operating a Th-U cycle, is it designed to breed 233U through decay of Pa outside the reactor core?
Fuel cycle characterisation	Check dependencies on the NMA tree	Does the design do breeding through a fertile blanket?
The fuel cycle facilities offer diversion paths and concealment possibilities		Are different steps of the fuel cycle co-located on the facility?
Are different steps of the fuel cycle co-located on the facility? Also consider breeding and fertile blankets. Fertile material types, quantities, and locations must be declared to, and can be independently verified by, the IAEA. Having a blanket adds a different type of nuclear material that is safeguarded in a separate location in the core.	Does this design create new or alter existing opportunities for facility misuse?	4.1. Does this design include new equipment or process steps that could change the nuclear material being processed to a type, category, or form with a lower significant quantity or timeliness objectives? (eg, reprocessing capabilities)
	Are different steps of the fuel cycle co-located on the facility? check dependencies on C/S for online reprocessing	Does the design employ online reprocessing of fuel salts (eg, in which a fraction of the salt is periodically withdrawn while the reactor is operating)?
on one hand, the removal of FP implies that part of the salt will be removed for reprocessing, potentially allowing new proliferation paths. On the other hand, the modelling of the evolution of the fuel composition has to take into account the removal of FP (accurate cross sections, etc.)	Are different steps of the fuel cycle co-located on the facility?	does the design employ the removal of fission products? Can the system's parameters be modified for misuse?
	Are different steps of the fuel cycle co-located on the facility?	does the design employ the removal of rare earth elements?

	Are different steps of the fuel cycle co-located on the facility?	Does the design employ the removal of noble metals?
	Are different steps of the fuel cycle co-located on the facility?	Can the actinides be removed by gaseous flotation via the gas FP removal system?
DESIGN INFORMATION VERIFICATION		
The design can be easily and readily verifiable by allowing the inspectors' accessibility		Can inspectors access the nuclear facility during the construction process for visual verification?
The design can be easily and readily verifiable by allowing the inspectors' accessibility		Can inspectors access the nuclear facility during operations for visual verification?
The design can be easily and readily verifiable by allowing the inspectors' accessibility		Can inspectors access the nuclear facility during decommissioning for visual verification?
When basic tools are adequate, they are also sufficient, and their use would impact the DE positively		Are adequate tools available for supporting verification activities?
The design can be easily and readily verifiable. A feature that facilitates DIV to ensure the design and the facility are equal. Modern techniques, such as the ones based on 3D laser range finders, allow performing a 3D mapping of the area to be verified, thus allowing computer-assisted verification of the equipment and facility layout.	Are adequate tools available for supporting verification activities?	Can state-of-the-art technology for the verification of design changes be applied?
The level of difficulty for modifying a fuel cycle facility depends on the complexity of the modification, the cost for the facility modification, the safety implications of such a modification, the time required to perform the relevant modification, and the ease with which inspectors can detect such modifications. Such modification might be detected by Design Information Verification (DIV) measures		Can the facility be easily modified?
A feature that facilitates DIV to ensure the design and the facility are equal. The extent of automation for a process influences its ease of modification. The more a process is automated, the stronger its proliferation resistance, since the degrees of freedom of movement are reduced.	Can the facility be easily modified?	does the design employ automation?
		If a safeguards approach based on false alarms is implemented, are the false alarms reduced to a minimum while maintaining a high DP?

THE DESIGN'S IMPACT ON THE DIVERSION PATHS		
		Does the design increase the difficulty of the inspectors' activities?
Are there design specifics that increase the difficulty of design information verification (DIV) by IAEA inspectors?	Does the design increase the difficulty of the inspectors' activities?	2.1. Does the design incorporate new or modified technology? If so, does the IAEA have experience with the new or modified technology?
Are there design specifics that increase the difficulty of design information verification (DIV) by IAEA inspectors?	Does the design increase the difficulty of the inspectors' activities?	Are there new design features with commercial or security sensitivities (not modifiable) that would affect inspector access to equipment or information? (e.g. the centrifuges are considered a secret and are not seen by the inspector, only the connection between them is inspected)
Are there design specifics that increase the difficulty of design information verification (DIV) by IAEA inspectors?		2.3. Do aspects of the design limit or preclude inspector access to, or the continuous availability of, Essential Equipment for verification or testing? (Provide Adequate space, support, and illumination to handle, identify, and conduct measurements on fuel)
Are there design specifics that increase the difficulty of design information verification (DIV) by IAEA inspectors?		2.4. Are there aspects of the design that would preclude or limit the IAEA's verification activities during the life of the facility?
Does this design create new or alter existing opportunities for facility misuse?		Are there locations in or near the core of a reactor where undeclared target materials could be irradiated?
Does this design create new or alter existing opportunities for facility misuse? Eg, reflectors substituted for blankets		Does this design introduce materials that could be effectively substituted for safeguarded nuclear material to conceal diversion?
Is there the potential to create additional diversion paths or alter existing diversion paths? This is particularly relevant for research reactors in which the layout changes. The same goes for fast reactors that can be burners, breeders, …		1.2. Does this design layout eliminate or modify physical barriers that would prevent the removal of nuclear material from process or material balance areas, e.g., circumventing a key measurement point (KMP)?

Does this design make detection of misuse more difficult?		4.2. Would this design preclude/affect the effectiveness of short-notice inspections, measurements, or process parameter confirmations?
Does this design make detection of misuse more difficult?		4.3. Do the design and operating procedures modify the transparency of plant operations (e.g., availability of operating records and reports or source data for inspector examination or limited inspector access to plant areas and equipment)?

EVALUATION SHEET FOR C/S		
rationale	Comment	Attributes
	CONTAINMENT	
The integrity of containment structures is paramount to the efficient application of C/S measures. A containment that can be altered poses risks.		Is the containment structure (such as walls) designed not to be drilled after construction?
This would impact the ease of practically designing the safeguards system		is the need for C/S equipment taken into account? (need for mounting spaces, wiring, reliable current supply, transmission of data)
In the design, is there the potential to create additional diversion paths or alter existing diversion paths?		1.3. Does this design obscure process areas or material balance area (MBA) boundaries, making containment/surveillance or installation of monitoring equipment more difficult?
		is the online chemical reprocessing unit (salt reprocessing) within structural containment?
Radiological hazard that has an impact on accessibility to areas where nuclear material is kept, but can make containment and surveillance easier		Does the radiation field from 232U contamination for 233U (ppm) pose the need for remote handling?
		In the case of breeder reactors in which fissile material is removed from the fuel salt circuit, is there a need for an in-containment bred fuel salt storage location?
		Are all the seals accessible for verification?

If seals are unprotected and undergo accidental breakage, it would be treated as a potential attempt at diversion and force the inspector to perform an inventory taking that is time-consuming and resource-intensive. Overall, this could easily impact DE and then improve safeguardability		Is there a provision for protecting seals from accidental damage?
Difficulty of inspector access to nuclear material might be used as part of a diversion strategy, but might also be compensated by strong C/S measures.		Is all the NM accessible for sealing of groups of assemblies without affecting nuclear material already under seal?
	SURVEILLANCE	
Few movements allow for easier application of surveillance, reconstruction of events, and thus maintaining CoK		is the need to move the NM, especially that under seal, minimised?
Having standardised items (e.g., flasks) in transfer would facilitate the application of C/S measures in several ways. Examples are: easier interpretation of recorded images in the review phase, easier surveillance of standard items in transit as they would probably result in moving at the same speed...		Is the transfer of nuclear material organised so that an item is transferred (or batch)? Are the items standardised?
Having only one or very limited possible transfer routes for items in transit would greatly improve the ease of performing surveillance and interpreting surveillance records during the review phase. This attribute is also particularly important for facilitating the application of containment measures, given that operational rules allow their use (e.g., no transfers for long periods of time).		34. Are the nuclear material movement paths and frequencies standardised (to facilitate containment and surveillance)?
		is the online chemical reprocessing unit (salt reprocessing) fully automated?
Some legal provisions might impede the use of optical surveillance (e.g. historically France). In current practice, surveillance relies heavily on the use of images recorded by surveillance cameras. To maximise the benefits that this technique embeds, the system can be designed keeping this in mind.		Possibility of applying optical surveillance and RM (remote monitoring)?

Remote surveillance helps to achieve timeliness and saves on-site inspection efforts		
Goes together with surveillance		The illumination system is reliable (redundancy, intensity and homogeneity)
		Automation of C/S equipment

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