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Mapping of the different status and alternatives for each ESNII concept

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Summary

This deliverable proposes a mapping of the different status and alternatives for each ESNII concept waste treatment process. The study undertakes a comparative assessment of the different recycling needs and options considered and identifies the facilities that could integrate the reprocessing options at EU level.

Approval

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GENIORS

GEN IV integrated oxide fuels recycling strategies

D8.3 Mapping of the different status and alternatives for each ESNII concept

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TABLE OF ABBREVIATIONS

Abbreviation	Meaning
1c-SANEX	One cycle Selective ActiNide Extraction
ALFRED	Advanced Lead-cooled Fast Reactor European Demonstrator
Am	Americium
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
Bi	Bismuth
CFV	Coeur à faible vidange
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
Cm	Curium
DIAMEX	Diamide EXtraction
ENEA	Energy and the Environment
ESNII	European Sustainable Nuclear Industrial Initiative
EU	European Union
EXAM	Extraction of AMericium
FP	Fission Product
FR	Fast Reactor
GANEX	Grouped ActiNide EXtraction
GENIORS	Generation IV Integrated Oxide fuels Recycling Strategies
GFR	Gas-cooled Fast Reactor
HLW	High-Level Waste
IRSN	Institut de Radioprotection et de Sûreté Nucléaire
i-SANEX	Integrated Selective ActiNide EXtraction
LBE	Lead-Bismuth Eutectic
LFR	Lead-cooled Fast Reactor
LWR	Light-Water Reactor
MA	Minor Actinide
MOX	Mixed OXides
MYRRHA	Multi-purpose hYbrid Research Reactor for High-tech Applications
NEA	Nuclear Energy Agency
Np	Neptunium
P&T	Partitioning & Transmutation

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Pb	Lead
Pu	Plutonium
PUREX	Plutonium and Uranium Refining by EXtraction
SANEX	Selective ActiNide Extraction
SCK•CEN	Studiecentrum voor Kernenergie • Centre d'Étude de l'énergie Nucléaire
SDG	Sustainable Development Goal
SFR	Sodium-cooled Fast Reactor
SNETP	Sustainable Nuclear Energy Technology Platorm
TBP	TriButyl Phosphate
TODGA	N,N,N',N'-Tetraoctyl Diglycolamide
U	Uranium
UN	United Nations
UOX	Uranium OXides
WP	Work Package

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EXECUTIVE SUMMARY

The European Commission launched in 2010 the European Sustainable Nuclear Industry Initiative (ESNII), with the aim of promoting the development of new designs of nuclear reactors – described as Generation IV. One major aim is to improve the management of nuclear wastes and thus to increase the global efficiency of the nuclear fuel cycle, by recovering a higher proportion of the energy contained in the natural uranium.

In line with the agenda for the deployment of ESNII, **GENIORS** (Generation IV Integrated Oxide fuels Recycling Strategies) addresses research and innovation in the fuel cycle chemistry and physics for the optimisation of fuel designs. It focuses on reprocessing and manufacture of Mixed Oxides fuels (MOX), potentially bearing minor actinides (MA). Within GENIORS, **work package (WP) 8** 'System Studies' explores the holistic impacts of changes in the partitioning and transmutation (P&T) processes on the nuclear fuel cycle.

This deliverable **D8.3** corresponds to the **task 8.3 'Process Mapping Studies'**, that aims at exploring the diverse fuel reprocessing options and strategies for some of the ESNII concepts: ASTRID, ALFRED and MYRRHA, in order to assess the requirements of treatment facilities that could integrate these reprocessing options at the EU level. The idea is to evaluate different recycling scenarios that ESNII concepts can adopt.

The report stats off by presenting the **developments of ASTRID[1], ALFRED and MYRRHA** including concept, objectives, status of the project, technical advances and challenges. Then, based on three reactor's concepts (ASTRID, ALFRED and LWR) a total of **six fuel cycles** have been described, bearing in mind that the exact parameters of these cycles will result from further engineering studies. **ASTRID and ALFRED consider the same two fuel cycles: MOX fuels with actinides recycled in the reactor and MOX fuels with actinides treated in MYRRHA**. In addition, **a fuel cycle of a light-water reactor and a fuel cycle coupling to MYRRHA** have also been included in the report so as to point out the differences between fast reactors and LWRs.

After the description of the reactor's fuel cycles, various **recycling processes** for the spent fuel were mapped, with a focus on EU processes. They are classified into two categories: **partitioning processes** (sequential and grouped separation) and **transmutation processes** (which aim at destroying some elements with a fission reaction).

From this analysis, **three recycling scenarios** were built, depending on the extent of recycling and recovered components (U, Pu, fission products, minor actinides, etc), which are studied in detail for each ESNII concept and its associated fuel.

- **Scenario 1. Multi-recycling of U and Pu**
- **Scenario 2. Multi-recycling of U, Pu and Am**
- **Scenario 3. Multi-recycling of U, Pu and all minor actinides**

For each scenario, the quantities of used fuel, flows of matter, evolution of radiotoxicity, treatment facilities and footprint of the high-level waste were described. This exercise makes it possible to compare the results with the existing capacities of the reprocessing plants in Europe and conclude on the possibility of increasing EU nuclear capabilities. The following schema depicts the proposed three scenarios, possible reprocessing processes and the associated fuel cycle option.

[1] This document does not include the current political uncertainties around ASTRID.

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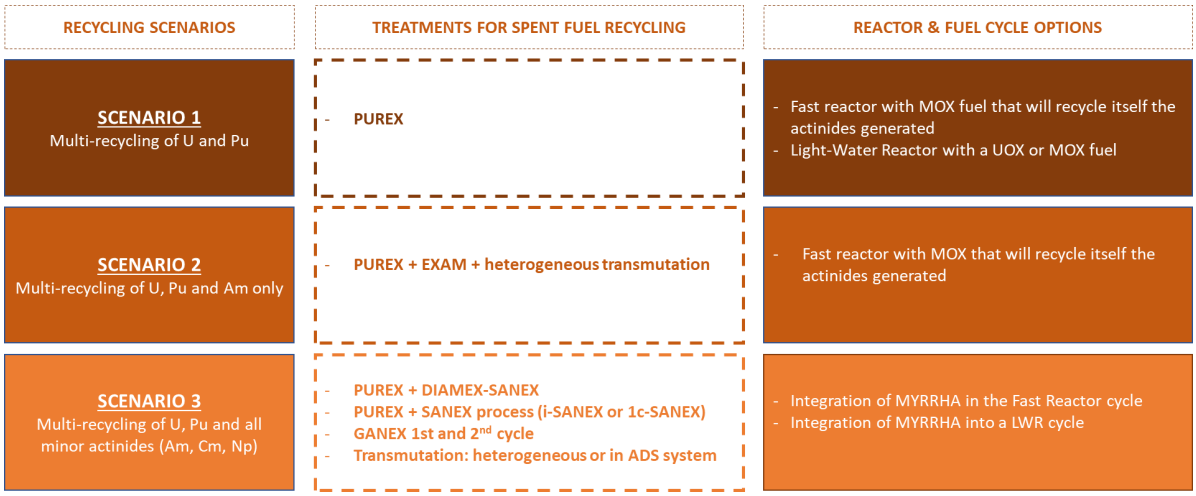

Finally, it is important to mention that this task does not include detailed calculations of the flows of matters in the case of the 'closed' fuel cycle. The holistic impacts of the implementation of a closed fuel cycle will be investigated in the following task 8.5 of the WP8. This task will build on existing work, including the present study, to push further the analysis to the whole nuclear fuel cycle.

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INTRODUCTION

'Ensure access to affordable, reliable, sustainable and modern energy for all' is one of the 17 Sustainable Development Goals (SDG) promoted by the United Nations (UN) (United Nations Sustainable Development Knowledge Platform 2018, 7). Similarly, the European Union (EU) is committed to reducing greenhouse gas emissions to 80-95% below 1990 levels by 2050, with the power sector being one of the highest contributors to this effort (European Commission 2012). With an interest to reduce significantly carbon dioxide emissions from their power supply infrastructure, many countries consider nuclear energy as an option in their energy mix. As of 2015, nuclear energy represents 10.6% of the world's total electricity production, with about 450 reactors (World Nuclear Association 2018).

However, several challenges need to be overcome to improve the sustainability of nuclear energy. First, almost all currently operating powerplants rely on uranium-235, the natural resources of which are limited. In effect, the efficiency of exploration is decreasing and if no new significant deposit is discovered, proven resources would last around 100 years at current production rate (Monnet 2016). Moreover, current nuclear reactors use only a tiny fraction of the energy contained within their natural fuel: 0.6 to 0.8% of the total potential for energy production in reactors (CEA 2012b, 1). A more sustainable nuclear technology should make use of a higher proportion of the energy available in the natural uranium. Finally, there have always been concerns regarding the waste produced by the nuclear industry, in particular the long-lived, high-level radioactive waste (HLW): the minor actinides (isotopes of neptunium, americium and curium). These wastes represent a small fraction of total waste volumes but account for most of their radioactivity: a safe and sustainable solution is needed to handle them.

As of today, there are different waste management and treatment strategies in the world. The first can be described as an open cycle or 'once through'. In this case, the spent fuel is at first stored in interim storage for several years of decades, then packaged in dedicated containers to be disposed of in underground storages or in repositories at the surface. This results in wasting large amounts of energy that could be recovered through recycling (Horvath and Rachlew 2016). Another strategy is the 'twice-through' fuel cycle: following interim storage, the uranium and plutonium contained in the spent fuel are recycled to manufacture a Mixed Oxide fuel (MOX), which contains a certain percentage of Pu. The smaller volume of residual waste is then stored underground (Horvath and Rachlew 2016). However, so far there are no industrially mature processes to effectively remove the minor actinides during reprocessing. Yet, these minor actinides represent most of the decay heat of spent nuclear fuels. Effectively separating them from the rest of the spent fuel could therefore improve the fuel cycle efficiency and sustainability (provided that an industrial solution for the handling of minor actinides exists).

A comparison of once-through and twice-through nuclear fuel cycles is depicted in the following scheme, which illustrates the once-through and twice-through cycles for the French case (Poinssot et al. 2014):

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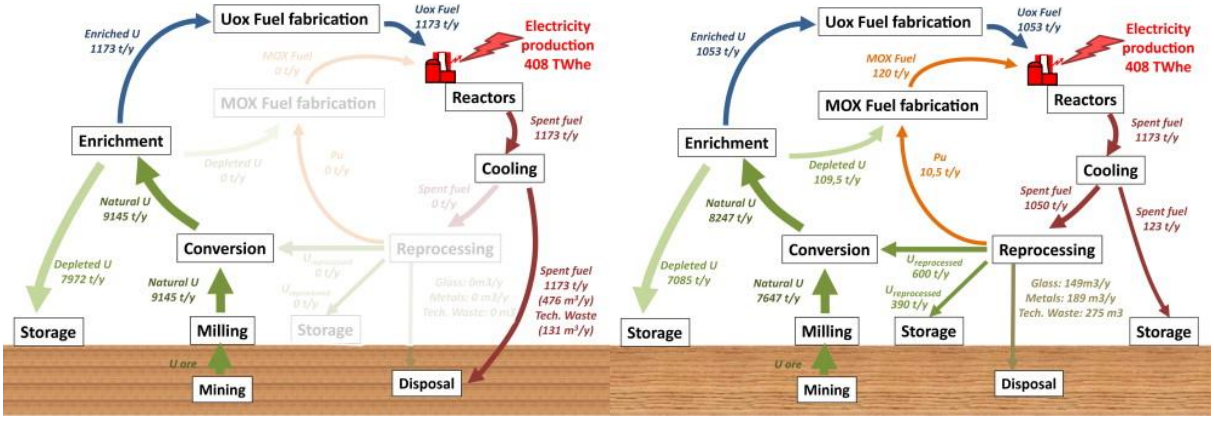

Figure 1: Comparison of a once-through (left) and a twice-through (right) cycle in the French case (Poinssot et al. 2014)

To tackle the challenges of a more sustainable nuclear energy, especially waste management, new concepts of reactors are needed. The Generation IV International Forum (GIF) has identified six innovative concepts which it aims at promoting: The Sodium-cooled Fast Reactor (SFR), the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Molten Salt Reactor (MSR), the Very High Temperature Reactor (VHTR) and the Super Critical Water Reactor (SCWR). The GIF identifies several objectives for Generation IV reactors, including waste minimisation (Generation IV International Forum 2018). Following this initiative, the European Commission launched in 2010 the European Sustainable Nuclear Industrial Initiative (ESNII) to support Generation IV fast reactors projects. More specifically, ESNII supports the SFR demonstrator project ASTRID, the LFR demonstrator ALFRED, the GFR demonstrator ALLEGRO and the multi-purpose Accelerator-Driven System (ADS) MYRRHA.

In parallel, there has been a tremendous R&D effort to come up with new reprocessing strategies in order to close efficiently the fuel cycle. The aim is to multi-recycle the uranium and plutonium of the spent fuel on one hand, and to partition and transmute the minor actinides on the other hand.

The ASTRID project is a 600 MW sodium-cooled fast reactor design to demonstrate the capability to implement the sodium technology with particular attention on safety-related issues incorporated in the design. The main objectives of ASTRID are to implement safely the multi-recycling of plutonium and the minor actinide transmutation, in order to meet the challenges in terms of waste management, resource utilisation and safety level (SNETP 2013b; IRSN 2015).

A shorter-term alternative technology is the lead-bismuth fast reactor. The LFR demonstrator, advanced Lead-cooled Fast Reactor European demonstrator (**ALFRED)**, is mainly focused on lead coolant activities to address specific characteristics that differ from lead-bismuth. ALFRED is an industrial scale lead-cooled fast reactor (LFR) of 300 MW_{th} that will generate 100-125 MW_e (SNETP 2013b). The project will be the first LFR plant connected to the grid and fulfilling the Generation IV goals.

In parallel to this project is **MYRRHA**. MYRRHA stands for Multi-purpose hYbrid Research Reactor for High-tech Applications. Currently under development by the Belgian Nuclear Research Centre (SCK•CEN), it is a multipurpose research facility based on the Accelerator-Driven System (ADS) concept. The ADS is a system which combines a subcritical 65-100 MW_{th} reactor core (i.e. there is not enough fissile material to spontaneously maintain the fission) fed by a 600 MeV proton accelerator via a spallation target (Baeten et al. 2014). With MYRRHA, ALFRED will reach the sufficient level of maturity needed to start the licensing and construction phase (IRSN 2015). An important remark is that MYRRHA is not designed for electricity production.

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OBJECTIVES & STRUCTURE OF THE DELIVERABLE

The aim of the deliverable 8.3 is to explore the diverse fuel reprocessing options and strategies for some of the ESNII concepts: ASTRID, ALFRED and MYRRHA, in order to assess the needs and evolution of the facilities that could integrate these reprocessing options at the EU level. The GFR demonstrator ALLEGRO has not been included in this study as it is a less mature project with fewer data available, making impossible to compare it with the other demonstrators.

The first section presents **the latest developments of the ESNII concepts** included in this study (provisional timetable, expected technical specification, etc). For ASTRID, ALFRED and MYRRHA, describe the concept, provide updates on the latest timeline and highlight the technical advancements and remaining challenges.

Then, in the **second section**, **fuel cycle options** foreseen for ASTRID, ALFRED and a general LWR were described (bearing into mind that the exact parameters of these cycles will result from further engineering studies). MOX fuels are considered for both ASTRID and ALFRED, and two options are explored for MYRRHA (associate it with a fast reactor of a light-water reactor (LWR)).

After describing the reactor concepts, the **third section** consists of a **mapping the various treatment processes** for the used fuel, with a focus on EU processes. They are classified into two categories: partitioning processes (sequential and grouped treatments) and transmutation processes (which aim at destroying some elements with a fission reaction). For all partitioning and transmutation options, a simplified scheme is given to illustrate the topic and some comparisons are made between the processes.

This mapping enables us to build **three recycling scenarios**, which are detailed in the **last section**. These scenarios differ on the extent of recycling and the recovered components (U, Pu, fission products, minor actinides…). They are studied in detail for each ESNII concept (and its associated fuel). For each scenario and each ESNII reactor, the quantities of used fuel and compare the scenarios in terms of flows of matter, evolution of radiotoxicity and footprint of the high-level waste were estimated. This exercise makes it possible to conclude on the capacities of the reprocessing plants in Europe and their potential future evolutions.

The objectives of the study are described in the following flow chart (Figure 2) which links the structure of the deliverable and its main aims.

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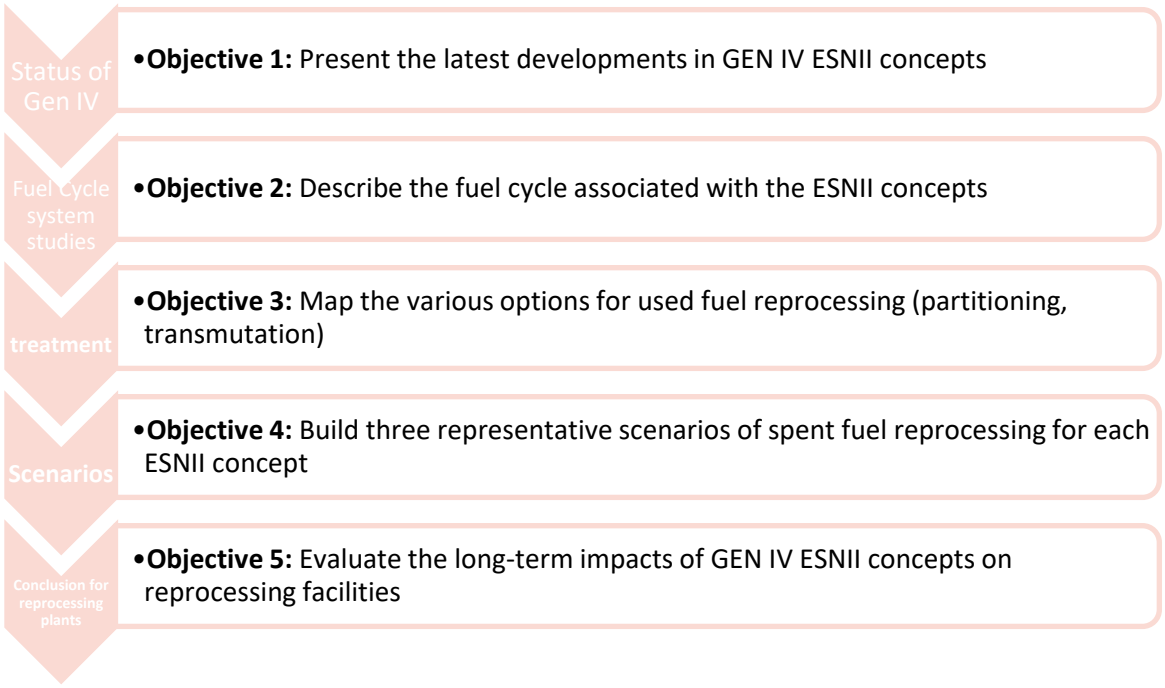

Figure 2: Flow chart of the deliverable showing its high-level objectives

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METHODOLOGY

The methodology undertaken for this report is laid out on Figure 3. The work has been divided into three steps (desktop research, interviews and selection and evaluation of scenarios) explained in terms of objectives, methods used and outcomes.

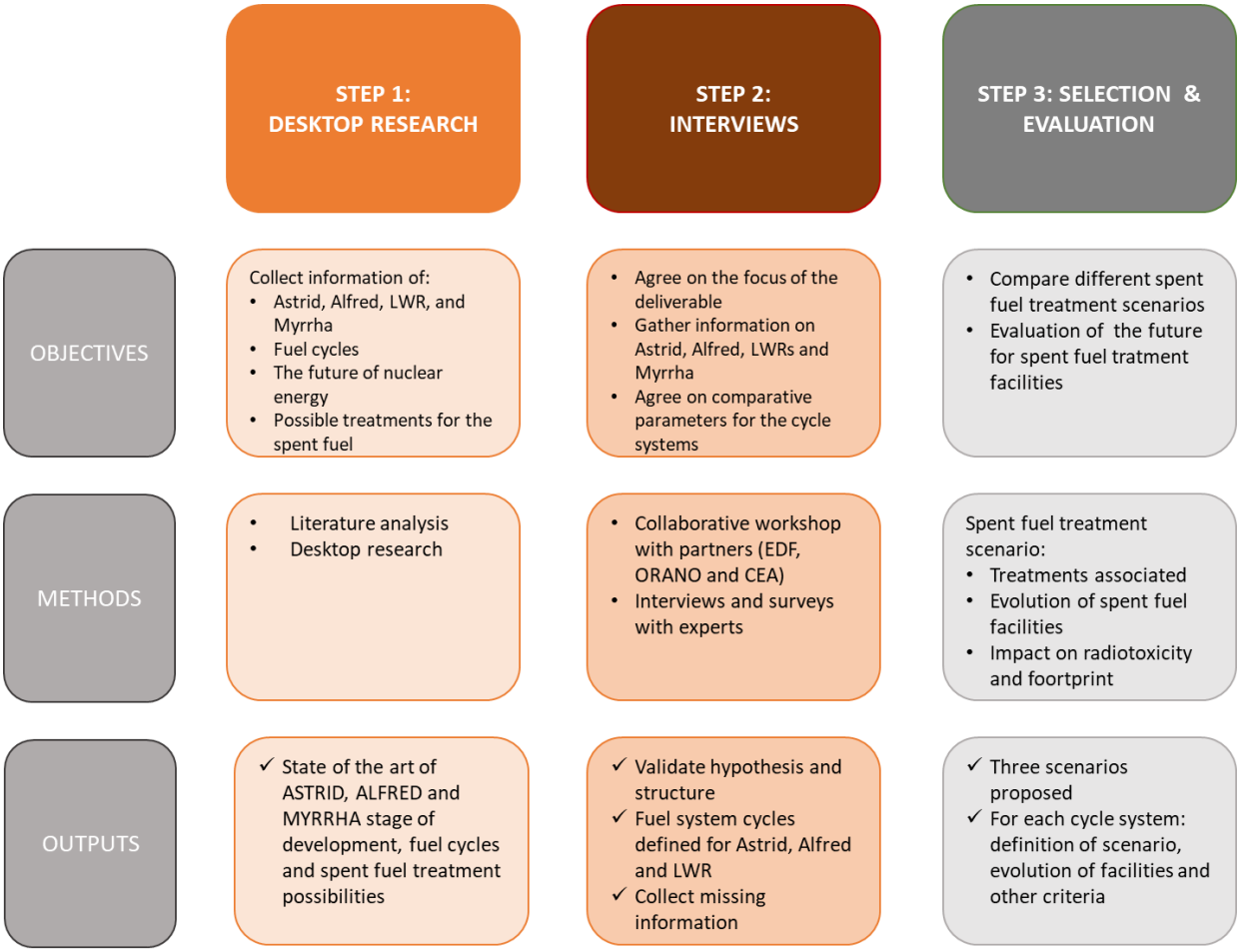

Figure 3 :Methodology for D8.3 (LGI, 2018)

Step 1. Desktop research

Objectives

The first step regards literature analysis and desktop research. It aimed at collecting relevant information about ASTRID, ALFRED, MYRRHA and a typical 3rd generation LWR to have a complete picture of the concepts, stages of development, technical advancements, challenges and the possibilities of fuel cycles. Also, information on the future of nuclear energy strategy, partitioning and transmutation possibilities for spent fuel treatment and facilities were studied.

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Methods

To achieve these objectives a deep literature and desktop research was carried out. The literature reviewed mainly comes from the following sources:

- Commissariat à l'énergie atomique et aux énergies alternatives (CEA)
- L'Institut de Radioprotection et de Sûreté Nucléaire (IRSN)
- the Nuclear Energy Agency (NEA)
- European Sustainable Nuclear Industrial Initiative (ESNII)
- Sustainable Nuclear Energy Technology Platform (SNETP)

Outputs

The main outputs of this first step were the following:

- Better understanding of the future for a sustainable nuclear energy in terms of development of IV generation concepts, spent fuel treatment needs and possibilities.
- State of the art of ASTRID, ALFRED and MYRRHA. Information was collected regarding the concept and objectives of each project, the status and timeline, main technical advancements and challenges that block the development of the technology.
- Selection of fuel cycle systems. ASTRID, ALFRED and a general 3rd generation LWR were compared in this study. MYRRHA as a standalone concept cannot be directly compared to other fast reactor since its aim is not to produce electricity but to treat minor actinides. Nevertheless, the coupling of fuel cycles to MYRRHA is relevant and was included in the study. A general third generation LWR was included in the study to highlight the differences between the 3rd and 4th generation concepts.
- Benchmark of the most important reprocessing processes used for partitioning and transmutation of Plutonium (Pu), Uranium (U) and minor actinides.
    - Sequential separation treatments
        - PUREX
        - DIAMEX-SANEX
        - i-SANEX
        - 1c-SANEX
        - EXAM
    - Grouped separation treatments
        - GANEX 1st cycle
        - GANEX 2nd cycle (EURO-GANEX, CEA-GANEX and CHALMERS)
    - Transmutation
        - Transmutation in fast reactors
        - Transmutation in accelerator-driven systems (ADS)

Step 2. Interviews

Objectives

The second step of the methodology aimed at completing the desktop research to have a precise idea of the stage of development and viable paths of study. This step also looked for agreement on the focus and structure

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of the deliverable with partners, especially the three recycling scenarios to analyse, gathering relevant information restricted to partner's sources.

Methods

Several interviews and workshops were carried out with the GENIORS partners and external experts of the nuclear energy sector. More specifically, partners from ORANO, Electricité de France (EDF) and CEA have strongly participated in the definition of the deliverable and the focus of it. A collaborative workshop was held at LGI's headquarters in Paris on the 13rd June. The partners who attended this workshop were:

- Frédéric Laugier (EDF) – Safety engineer
- Benjamin Fleury (EDF) – R&D Business Manager
- Gerald Senentz (ORANO) – Manager
- Amar Ait Abderrahim (LGI) – Trend Watcher
- Soraya Molinero Pérez (LGI) – Trend Watcher

A dedicated questionnaire was created for this workshop in order to raise questions and ease the discussion between partners.

In addition to the collaborative work with the GENIORS partners, several experts were interviewed to broaden ideas and complete the existing information found. LGI has contacted:

- Christine Chabert (CEA) – Department of Reactor Studies. Expert on ASTRID.
- Michele Frignani (ANSALDO) – General Manager. Information about ALFRED.
- Giacomo Grasso (ENEA) – Nuclear reactor core designer. Information about ALFRED.
- Hamid Aït-Abderrahim (SCK•CEN) – PhD in Reactor Physics. Information about MYRRHA.

All questionnaires created for the interviews can be found in Annexes.

Outputs

The early stage of development of many 4th generation concepts and the current lack of public information made these exchanges crucial to validate hypothesis and collect missing information. The workshops and interviews with the GENIORS partners and experts served to:

- Validate initial hypothesis and possible recycling scenarios.
- Propose several fuel cycle options for each type of reactor:
    - ASTRID:
        - SFR with MOX fuel recycling the actinides inside the reactor
        - Integration of MYRRHA in the SFR fuel cycle
    - ALFRED:
        - LFR with MOX fuel recycling the actinides inside the reactor
        - Integration of MYRRHA in the LFR fuel cycle
    - LWR:
        - LWR with a mix of UOX/MOX fuel
        - Integration of MYRRHA in the LWR fuel cycle

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Step 3. Selection and evaluation

Objectives

The last step gathered the outcomes and information collected during step 1 and 2 to select and evaluate three recycling scenarios proposed for the 4th generation reactors. It also includes the possibilities of the future spent fuel treatment facilities and waste management needs.

Methods

Three recycling scenarios were proposed within this last step. They have been classified by the relevant elements recycled.

Scenario 1. Multi-recycling of Pu and U

Scenario 2. Multi-recycling of Pu, U and Am only

Scenario 3. Multi-recycling of Pu, U and all minor actinides (Am, Cu and Np)

For each scenario, the most suitable treatments were described. For instance, PUREX treatment could fit well in Scenario 1, and so on. Afterwards, each fuel-cycle option of ASTRID, ALFRED and general LWR were related to the most suitable recycling scenario. As an example, ASTRID treating minor actinides inside the reactor could fit Scenarios 1 and 2. Other criteria were included such as the evolution of reprocessing facilities, the impact on radiotoxicity's evolution of minor actinides and final waste surface disposal when information was available.

Outputs

The main outcomes of the last step were:

- Definition of three recycling scenarios including the possible reprocessing processes associated.
- For each fuel cycle system, selection of a specific scenario.
- The indicators used for the comparison of scenarios are presented as follows:

Criterion	Applicable in
Evolution and needs of reprocessing facilities	All options
Impact on minor actinides' radiotoxicity	All options
Impact on final waste surface disposal	When information was available

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1. STATUS OF THE IV GENERATION REACTORS

This section aims at presenting the status of different 4th generation concepts: SFR (ASTRID project), LFR (ALFRED project) and MYRHHA (a research reactor for transmutation purposes). Fast reactor technology is fundamental to close the fuel cycle and therefore, achieving a more sustainable implementation of nuclear energy. This section highlights the stage at which these projects are, presents some technical advantages, barriers and potential deployment timeline. For each reactor, the following information has been structured in:

- **Concept and objectives:** presentation of the project and main objectives pursued.
- **Status of the project and timeline[2]:** situation of the reactor, its developments and potential timeline for deployment.
- **Technical advancements:** over traditional nuclear reactor and other fast reactors.
- **Challenges**: main blockage points and barriers that each technology has to deal with.

1.1.ASTRID: SODIUM-COOLED FAST REACTOR (SFR)

CONCEPT & OBJECTIVES

In 2007, the ASTRID programme was launched by the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) gathering European and international partnerships. ASTRID will be a pool type, sodium cooled fast reactor of 1500 MWth that will generate about 600 MWe (SNETP 2013b).

The main objective of the programme is to **ensure the demonstration of the Gen IV Sodium-cooled Fast Reactor (SFR) at industrial scale**. The project looks for the realisation of sodium technological loops, the validation of components as well as the construction of a fuel manufacturing facility (AFC). In addition, **ASTRID pretends to demonstrate the multi-recycling of plutonium**, in order to **preserve natural uranium resources, minor actinide transmutation, and so reducing nuclear waste**.

The sole coolant used is **liquid metallic sodium**. Although this liquid metal coolant has low specific heat, a great advantage is that sodium allows significant absorption of heat during liquid phase. However, it is highly chemical reactive so, special safety measures are needed when it is in contact with water or air.

There is significant industrial experience in this type of reactor. For instance, the French reactor Phoenix has been operating for 35 years and was shut down in 2009. Significant knowledge has been gathered to further improve this concept. The following table presents some features of the SFR technology to be highlighted (CEA 2012a).

Table 1 : Features of the Sodium-cooled Fast Reactor of the ASTRID project (CEA, 2012)

CORE AND FUEL	
Fuel material	MOX, potentially with high minor actinides content
Cladding material	15-15 Ti work hardened austenitic steel AIM1
Fuel assembly	Steel pin which contains the fuel in the form of annular pellets
Core and subassemblies	CFV core (core with low void effect)
COOLANT	

[2] This document does not include the current political uncertainties around ASTRID and is based on the information published by the CEA in 2012.

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Type	Sodium (Na)
Melting Point (°C)	98
Boiling Point (°C)	883
Chemical reactivity (w/air and water)	Highly reactive

STATUS OF THE PROJECT AND TIMELINE

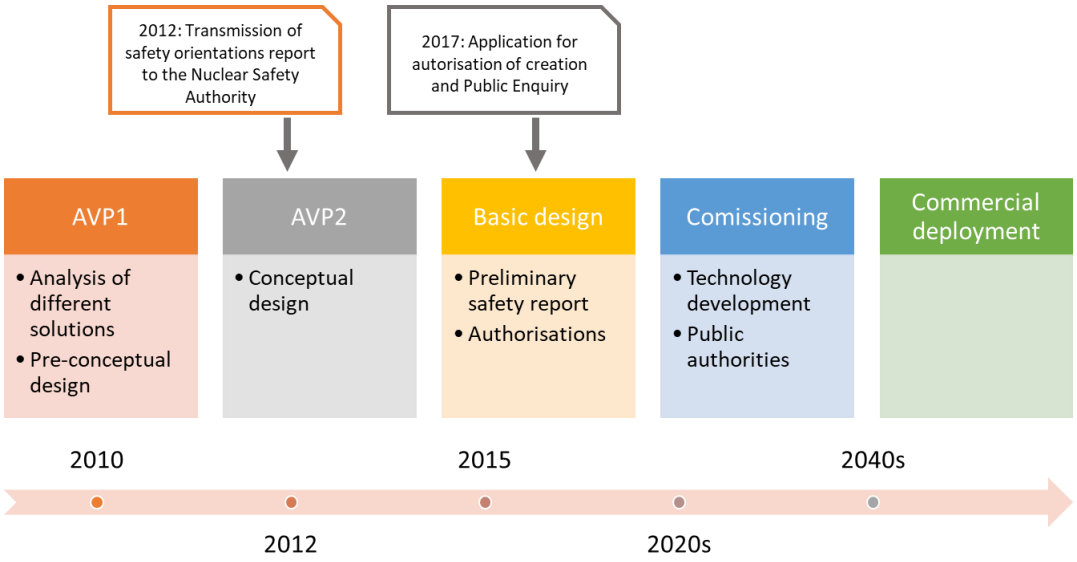

Figure 4: Overall schedule of the ASTRID project (CEA, 2012)[3]

The figure above presents the main tasks and milestones expected for the deployment of ASTRID. The design work has been structured in different phases (CEA 2012):

- The **pre-conceptual design (AVP1)** started in 2010 when the CEA analysed different solutions and chose the most appropriate one. The pre-conceptual design phase compromised the core engineering studies (beginning of 2010), the nuclear island engineering studies (September 2010) and studies of power conversion system (June 2011) among other tasks.
- In 2012, the second phase called **the conceptual design (AVP2)** started. The objective was to consolidate the design of ASTRID by validating different hypothesis and was expected to be completed at the end of 2014. The studies performed by different engineering groups have started at different times so there were different level of advancements.
- The third step was called "**basic design**" (2015-2019). It includes the preliminary safety analysis report, more detail studies and authorisations for building ASTRID. Modelling in 3D will be required for detailed studies and associated calculations. In addition, design studies will take into account uncertainties thanks to the first qualification tests for the ASTRID product concept definition.
- **Commissioning** is expected in the early 2020s, but it strongly depends on technology development and public authorities.

[3] This document does not include the current political uncertainties around ASTRID and is based on the information published by the CEA in 2012.

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- This will follow a ten-year operation until the **commercial deployment** of SFRs which could be possible in the 2040s at the earliest (CEA 2015).

TECHNICAL ADVANCEMENTS

ASTRID design benefits from favourable high safety margins to prevent events such as Fukushima. It has the main objective of demonstrating the SFR at industrial scale and therefore, some advancements will be achieved in the following areas (SNETP 2013b; CEA 2015):

- ASTRID is a **self-sufficient reactor** that will enable the optimisation of uranium resources, multi-recycling of plutonium and the half-life and toxicity of ultimate waste (actinides).
- SFR has a **high level of resistance to loss of coolant accidents** (LOCA) since it is possible to install a guard vessel around the main one. The main vessel contains the primary system and it is not pressurised.
- The secondary system uses sodium loops to transfer energy from the primary circuit to the main heat exchangers, providing an **additional safety barrier**.
- The size and mass of the primary system and the physical proprieties of primary coolant provides a large thermal inertia to the reactor and so, it allows larger grace times when putting in operation the decay heat removal (DHR) systems.
- ASTRID's core presents an advantage from the classical fast neutron reactor designs. It is made of a **heterogeneous MOX core** called "low sodium worth core" (LWC), which has a sodium void coefficient of reactivity closed to cero, that is to say, the core tends to remain stable.

CHALLENGES

Although the SFR will introduces many innovations, it has to deal with several challenges mostly related to safety and resistance to hypothetical accidents. The main issues are summarised below (SNETP 2013b; CEA 2012a).

Reactor safety

One of the main concerns spins around the core meltdown probability and how to reduce it at the lowest achievable. Some options are being currently under studied such as the "coeur à faible vidange" (CFV) or LWC. This concept focuses on optimising the core neutron feedback parameters (reactivity coefficients) to obtain an improved natural core behaviour during core heating or other unstable conditions. In other words, the reactivity effect due to sodium expansion is negative in the scenario of a total loss of primary coolant and so, the overall void effect is negative whether a boiling phase is reached.

There are several preliminary studies that have shown the potential of how the CFV core can fulfil high safety standards. However, there is still much work to be done in order to demonstrate this concept in terms of future simulation and experimental validation.

In addition, there are other safety devices that will be tested on ASTRID. Sepia is a passive-type emergency shutdown system patented by the CEA that promises to enhance safety margins of the reactor. Nevertheless, further research and development (R&D) is required to analyse other alternatives and advancements. Moreover, improvement of instrumentation and measurement systems are ongoing in order to ameliorate the core and reactor monitoring (IRSN 2015; OECD Nuclear Energy Agency 2016).

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Sodium-water risk: power conversion system

There are also some risks related to the affinity of sodium-water reaction. To achieve higher acceptability and safety, new energy conversions systems are being considered. For instance, the Brayton Cycle[4] or the use of gas for thermodynamic transformations. However, more research is needed in order to introduce this concept into the core while excluding any kind of risk with the gas. In addition, further R&D has to be considered in terms of steam generator concepts to have better protection in case of sodium-water reaction and tube failure.

Resistance to a potential mechanical energy release accident

A core-catcher will be designed to recover the entire core and ensure long-term cooling. However, other options need to be investigated in terms of core-catcher technologies, locations and performances. Likewise, significant effort is required to design a container that can resist mechanical energy release without any other reaction outside the site boundary in case of accident.

In-service inspection and reparability

Periodic inspection is fundamental in order to ensure a high level of safety. Further R&D mainly on optical and ultrasonic systems is needed to develop new technologies more efficient and effective for the inspection of the primary system of the reactor.

Higher availability and reduced shutdown times

The improvement of fuel handling system's performances is also crucial. Transmutation fuels, cooling times, fuel loading and unloading are some of the characteristics to be taken into consideration. Although sodium provides a great operation flexibility in normal and accident conditions, further studies on handling, cleaning, repair and requalification operations need to be performed.

1.2.ALFRED: LEAD-COOLED FAST REACTOR (LFR)

CONCEPT & OBJECTIVES

ALFRED stands for an advanced Lead-cooled Fast Reactor European demonstrator that has emerged as a solution for the next generation of nuclear power plants. ALFRED is an industrial scale lead-cooled fast reactor (LFR) of 300 MWth that will generate 100-125 MW_e (SNETP 2013b). The project is supported by the European Sustainable Nuclear Industrial Initiative (ESNII) and it will be the first LFR plant connected to the grid and fulfilling the Generation IV goals. MYRRHA is the first prototype of a nuclear reactor driven by a particle accelerator that will work in collaboration with ALFRED for enhancing safety requirements and design.

Like ASTRID, one of the objectives is to reach a **closed-fuel and sustainable cycle** in the long term. The ideal closed-fuel cycle is achieved by a highly efficient use of the energy content of the fissile material and the recycling

[4] http://web.mit.edu/16.unified/www/SPRING/propulsion/notes/node27.html

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of the used fuel of the reactor. In those terms, **the amount of long-life radioactive waste produced can be drastically reduced** while assuring a longer term fuel availability (OECD Nuclear Energy Agency 2016) .

The **heavy liquid metal coolant** used in this reactor is lead and presents several important features such as high scattering and low absorption cross sections that enable a simplified design. The use of lead-bismuth eutectic (LBE) as a coolant is also under study. Unlike sodium, lead does not react violently with water or air so that the system can be simpler, reducing the total cost of the construction and maintenance. Also, the intrinsic features of the coolant provides a very high level of safety ("FALCON, a Consortium to Build a Gen IV Lead Fast Reactor Demo in Romania" n.d.). These and other features are gathered on Table 2.

Table 2: Features of the Lead-cooled Fast Reactor of the ALFRED project (Smith 2017)

CORE AND FUEL	
Electrical power (MWe)	300
Fuel material	MOX, potentially with high minor actinides content
Cladding material	15-15 Ri coated (T91 coated as an option)
Fuel assembly	Hexagonal wrapped
COOLANT	
Type (Option 1)	Lead (Pb)
Melting Point (°C)	327
Boiling Point (°C)	1737
Chemical reactivity (w/air and water)	Practically inert
Type (Option 2)	Lead-bismuth eutectic (LBE)
Melting Point (°C)	125
Boiling Point (°C)	1670
Chemical reactivity (w/air and water)	Practically inert

STATUS OF THE PROJECT AND TIMELINE

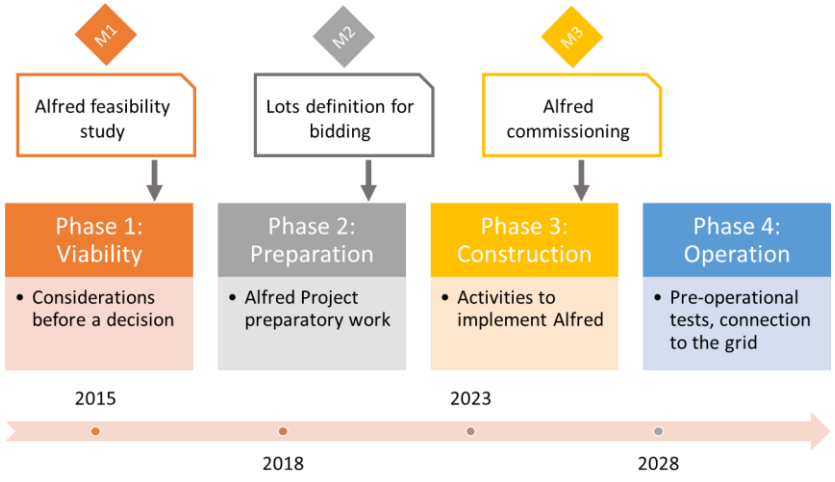

Figure 5 : General roadmap of ALFRED project (Turcu et al. 2017)

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In December 2013, a consortium compromising Italy's National Agency for New Technologies, Energy and the Environment (ENEA), Ansaldo Nucleare and Romania's Nuclear Research Institute was set up for the development and construction of ALFRED. ALFRED will be built in Mioveni (Romania) next to a fuel manufacturing plant already existing. The project has been divided into four phases (Turcu et al. 2017):

- **Phase 1. Viability**
  Several options will be considered by the consortium before taking the final decision. The objective was to have a clear commitment to ALFRED as a major nuclear program.
- **Phase 2. Preparation**
  ALFRED enters in preparation phase and lots definition for bidding.
- **Phase 3. Construction**
  Construction activities to implement ALFRED are expected to start around 2023.
- **Phase 4. Operation**
  The final step includes pre-operational tests and connection to the grid. ALFRED will start operation around 2028.

TECHNICAL ADVANCEMENTS

ALFRED brings several key advances compared to previous technologies (IRSN 2015; SNETP 2013a; Smith 2017):

- **A simplified design**. The inert nature of the lead coolant, the low absorption, and high scattering features allow the use of a decay heat removal system based on a well-known water technology.
- **High safety features**. The development of MYRRHA and ALFRED will upgrade the safety level of this kind of reactor thanks to both inertness and intrinsic properties of lead. The high boiling point (1737°C) at atmospheric pressure provides high margins under normal operating conditions and the risk of coolant boiling is reduced.
- **In-service inspections techniques.** New inspection techniques will be developed and studied during the project.
- **Thermal capacity of lead combined with large mass of coolant.** As a result, the need for intervention is significantly delayed and almost eliminated.
- Neutrons are poorly moderated in the coolant thanks to the high atomic mass of lead. As a result, **neutrons remain in the high-energy** range and are able to generate new fissile fuel (Pu).
- **LBE has a major advantage** in comparison to lead. It has much lower melting/freezing point (125°C) versus 327°C so it allows lower temperature operation and reduces engineering issues.

CHALLENGES

ALFRED is a promising project that will bring innovation and will work for the sustainable development of IV generation of reactors. However, it must overcome several challenges related to:

Materials qualification

For many decades, the **unavailability of qualified materials** for operating in a heavy liquid metal environment at high temperatures (500-550°C) has postponed the development of LFR. As a result, it has forced the selection of lead-bismuth eutectic (LBE) as a primary coolant. However, ALFRED will represent the very first time that a critical heavy liquid metal cooled reactor would provide electricity to the grid (SNETP 2013b).

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Erosion and corrosion

Another barrier is the **high corrosion level of lead-cooled** in contact with stainless steel. It is a major problem especially for the primary pump. Several R&D programmes have been set up to find solutions. For instance, the creation of an iron oxide layer on the surface of stainless-steel structures. Nevertheless, this process is complex and have different constrains in term of operating temperature and coolant purification (IRSN 2015). Other programmes are also studying the potential of graded composites and Silicon and Aluminium enhanced materials.

Degradation of mechanical properties

Lead also affects **the resistance of structures** to creep and wear. To overcome this issue, protective coatings are being studied and several options will be developed during the project. Steels can also become embrittled in contact with lead, which is not advantageous and needs new materials' research.

High melting point (327°C)

The risk of lead freezing also limits the reactor operation temperature. LFR safety is thus highly dependent on operating procedures, proper engineering will be crucial to avoid lead freezing.

1.3. MYRRHA: FLEXIBLE FAST SPECTRUM RESEARCH REACTOR

CONCEPT & OBJECTIVES

MYRRHA stands for Multi-purpose hYbrid Research Reactor for High-tech Applications. Currently under development by the Belgian Nuclear Research Centre (SCK•CEN), it is a multipurpose research facility based on the Accelerator-Driven System (ADS) concept. The ADS is a system which combines a subcritical reactor core (i.e. there is not enough fissile material to spontaneously maintain the fission) fed by a proton accelerator via a spallation target (Baeten et al. 2014). The accelerator will have a beam energy of 600 MeV with a beam current of 2.4 to 4 mA, while the reactor will have a power of 65 to 100 MWth and will be pool-type with a lead-bismuth eutectic coolant (Angulo 2017).

Different objectives are pursued through the MYRRHA project. MYRRHA will allow fuel and material developments for GEN IV and fusion reactors, as well as radioisotope production for medical and industrial applications. Furthermore, the fact that MYRRHA will be cooled by an LBE coolant will allow important synergies with LFR reactors development, in particular with the ALFRED project. In addition, MYRRHA will be crucial for studying transmutation of minor actinides contained in high level waste: a subcritical ADS could safely achieve transmutation of cores bearing high amounts of MAs (Baeten et al. 2014). Being a research reactor, MYRRHA is not designed for electricity production (although ADS can produce more electricity than consumed).

As mentioned for ALFRED, LBE has several advantages over sodium for use as a coolant: higher boiling point, no reaction with water or air, possible use for spallation target. However, LBE is more corrosive to steel than sodium, and alpha contamination needs to be monitored during refuelling.

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Table 3: Features of the ADS project MYRRHA (Angulo 2017)

CORE AND FUEL	
Thermal power (MWth)	65-100
Fuel material	MOX, potentially with high minor actinides content
Cladding material	15-15 Ti stabilised stainless steel at first (T91 ferritic-martensitic steel further)
Fuel assembly	Hexagonal wrapped
COOLANT	
Type	Lead-bismuth eutectic (LBE)
Melting Point (°C)	123.5°C
Boiling Point (°C)	1670°C
Chemical reactivity (w/air and water)	Practically inert

STATUS OF THE PROJECT AND TIMELINE

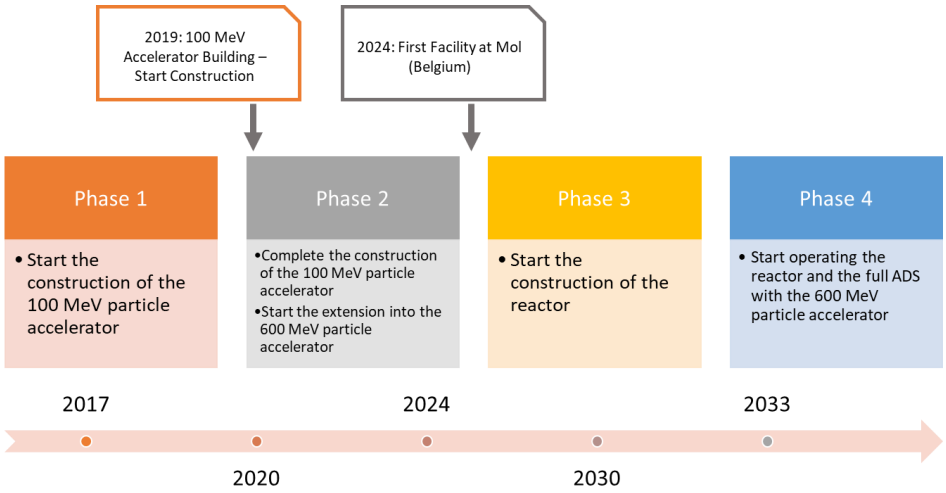

Figure 6: General roadmap of MYRRHA project (SCK-CEN 2018)

The figure above presents the general roadmap of the MYRRHA project. SCK•CEN distinguishes four phases, some of which take place in parallel (Angulo 2017):

- Phase 1: Phase 1 consists of prototyping and construction of the 100 MeV accelerator. Following a large effort of pre-licensing since 2010, the licensing of the 100 MeV accelerator started in 2016. Construction is expected to be accomplished by the end of 2022.
- Phase 2: After completing the construction of the 100 MeV accelerator, the 600 MeV particle accelerator will be constructed. Construction could start by 2024 and end in 2030.
- Phase 3: in parallel to phase 2, the reactor will be constructed between 2024 and 2030.
- Phase 4: By 2033, it is expected that MYRRHA will be able to start operations. Commissioning could begin in 2030 for the accelerator and the reactor.

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TECHNICAL ADVANCEMENTS

Although there are other R&D programmes regarding Accelerator Driven Systems (notably in India and China), MYRRHA will be among the first prototypes of ADS to be built (World Nuclear Association 2017). There are several features that distinguish it from conventional reactors (SCK-CEN 2018; Baeten et al. 2014):

- Being a flexible irradiation facility, MYRRHA **is able to work under both critical and subcritical conditions**. With a subcritical reactor core, the probability of a runaway reaction is minimal.
- MYRRHA is **multi-purpose**: it allows fuel development for innovative reactor systems, material development for GEN IV and fusion reactors and radioisotope production for medical and industrial uses.
- The **coolant used for MYRRHA is the LBE**, which is an option for LFRs such as ALFRED. Developments brought forward by MYRRHA could therefore benefit other GEN IV projects.
- MYRRHA could provide an **alternative option** (to critical fast reactors) for partitioning and transmutation of minor actinides (MA) present in used fuels. The core loading could contain a higher amount of MA.

CHALLENGES

In order to demonstrate the feasibility of an ADS demonstrator, MYRRHA will need to overcome some crucial challenges (CEA 2012f; World Nuclear Association 2017):

System complexity

ADS are complex systems since they associate a proton accelerator, a spallation target and a subcritical reactor. Feasibility on an industrial scale has not been demonstrated for each of these concepts and their coupling. Proving the feasibility of this coupling at a reasonable power level is crucial to allow operation feedback and eventually, scalability to an industrial demonstrator.

In addition, burning a significant portion of the MA that will be fed into MYRRHA would require several irradiations (not all the MA could be burned at once). Therefore, several installations around MYRRHA would be required (manufacturing of the MA-bearing fuel, recycling of this fuel…).

Production of volatile radioactive isotopes in the spallation target

Although ADS produce much less long-lived actinides than LWRs (and can even be used to burn efficiently minor actinides), the process generates radioactive isotopes from the target material, some of which are volatile, thus causing maintenance challenges. MYRRHA operators will need to deal with the added radiotoxicity induces by these elements.

Production of electricity

MYRRHA is not intended to generate more energy than it consumes. However, it is expected that the energy consumption will remain 'reasonable', allowing to believe that further industrial-scale demonstrator ADS could produce electricity, which is not proven as of today. Moreover, compared to classical LWRs, the power production is not as reliable due to forecasted accelerator downtime: MYRRHA's objective is to have a mean time between failure over 250 hours to prove its reliability. Finally, in the case of a FR deployment scenario, if one

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wants to burn the MAs through ADSs, this could result in a higher cost of electricity due to higher investments (compared to having only FR).

Erosion and corrosion

As discussed for ALFRED, another barrier is the **high corrosion level of lead-cooled** in contact with stainless steel (MYRRHA uses a similar coolant). Several R&D programmes have been set up to find solutions. For instance, the creation of an iron oxide layer on the surface of stainless-steel structures. Nevertheless, this process is complex and have different constrains in term of operating temperature and coolant purification (IRSN 2015). MYRRHA could mutualise some R&D effort with ALFRED (and other projects and programmes) to help solve this issue.

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2. FUEL CYCLE SYSTEM STUDIES

This section presents the **analysis of different fuel cycle systems based on three different reactor concepts (ASTRID, ALFRED and a general LWR)**. It has to be noted that this is a theoretical study based on a steady-state cycle around a reactor. In reality, the number of reactors would have to cover electricity requirements in which the equilibrium of GEN IV reactors would be done in several stages with reactor generation mixtures. MYRRHA as a standalone concept cannot be directly compared to other fast reactors since its aim is not to produce electricity but to treat minor actinides. Nevertheless, the coupling of fuel cycles to MYRRHA is relevant and has been included in this section. A general LWR was incorporated to the study so as to highlight the differences between fast reactors (ASTRID and ALFRED) and LWRs.

As a first step, it is important to understand the differences between these three fuel cycles and their options: what is the type of fuel used, the in and out fluxes, minor actinide generation, etc. For each reactor concept, different configurations have been analysed:

ASTRID

- ASTRID (SFR) with a mixed oxide fuel (MOX) that will recycle itself the actinides generated;
- Integration of MYRRHA in the SFR cycle: MYRRHA would treat the minor actinides generated by SFRs and SFRs would multi-recycle its Pu by using a MOX fuel.

ALFRED

- ALFRED (LFR) with a MOX fuel that will recycle itself the actinides generated;
- Integration of MYRRHA in the European-LFR cycle: MYRRHA would treat the minor actinides generated by ELFR and ELFR would multi-recycle its Pu by using a MOX fuel

LWR

- LWR with a mix of UOX/MOX fuel (current twice-through cycle seen for instance in France);
- Integration of MYRRHA into an LWR cycle: MYRRHA would treat the minor actinides generated by the LWR which would function on either UOX of MOX fuel.

2.1. ASTRID

As presented before, ASTRID is the project that will demonstrate the deployment of a 600 MW_e Sodium-cooled Fast Reactor. The program also envisages the construction of a core fuel fabrication workshop (AFC) with a capacity of 5-10 t_{MTHM}/year to produce the reactor fuel. This installation will eventually take the role played by the fuel factory in Cadarache (France), now stopped and being dismantled. A nominal capacity of 10 tons of heavy metal (t_{MTHM}/year) can fabricate the fuel core of ASTRID within three years (CEA 2012b). Several options remain open regarding the management of spent fuel. Two fuel cycle options are described in this section (Special Issue 2016):

1. ASTRID (SFR) with a mixed oxide fuel (MOX) that will recycle itself the actinides generated

The whole fuel cycle is laid out in Figure 7. Information was provided by CEA to estimate **the flux of spent fuel generated by the ASTRID demonstrator of 600 MW_e Sodium-cooled Fast Reactor (SFR).** From a fast reactor fleet accounting for 60 GWe, calculations were made to estimate ASTRID's material flow, (600 MW_e) which is

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4.5 t/year for inputs and outputs (neutral balance). The outputs include 0.05 t of minor actinides, 3.10 t of U and 0.90 t of Pu that will be reused to produce new fuel. Within an SFR, a repeated recycling of fuel materials can be achieved. Pu is the vector by which the energy potential of the fuel can be entirely valued. However, the recycling of Pu in the ASTRID generates minor actinides (0.05 t/y), which are the main contributors to radiotoxicity. The fission products constituting the final waste represent around 0.45 t/y. It is relevant to highlight that for a given quantity of electricity, the amount of minor actinides formed is four times lower if Pu is recycled in a fast reactor compared to the recycling in a LWR fuelled with MOX (CEA 2012f). Moreover, MOX fuel can be recycled multiple times in an SFR while it can be recycled only once in an LWR.

In this scenario, the recycling of certain long-lived elements into the SFR was considered to reduce the long-term radiotoxicity and thermal power of the ultimate waste. This option only considers the ASTRID project. However, if a fleet of SFRs want to be deployed, the design of a fuel factory is fundamental to meet the structural and capacity requirements. In addition, to meet the objectives for a successful deployment, fuel fabrication must address various challenges. For instance, the ability to recycle all grades of Pu and U available in the current fuel cycle. Thus, manufacturing plants must be able to integrate the reprocessing of Pu maintaining a high level of safety while aiming for acceptable economic performance (CEA 2012c).

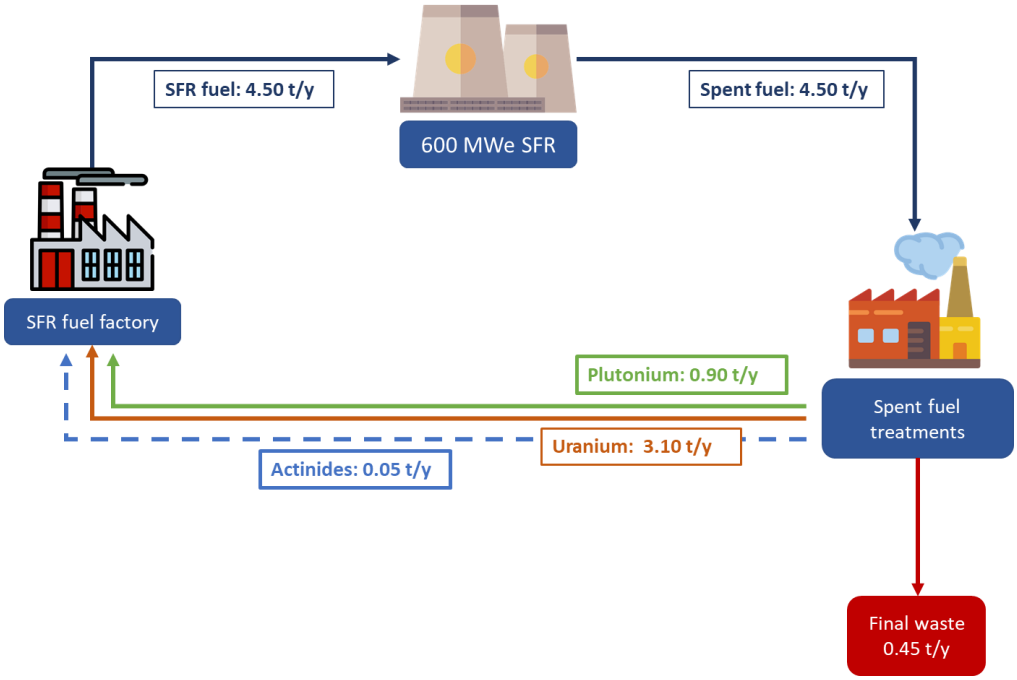

Figure 7 : ASTRID fuel cycle (LGI, 2018)

2. Integration of MYRRHA in the SFR cycle

This option is similar to the previous one, but MYRRHA would treat the minor actinides generated by SFRs and SFRs would multi-recycle its Pu by using a MOX fuel. There are several advantages (CEA 2012f):

- Reduction of radiotoxicity thanks to transmutation processes. Ultimate waste's radiotoxicity can be reduced by a factor between 20 and 100, depending on the time horizon considered. Only the transmutation of Am reduces radiotoxicity by a factor of 2.

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- Minor actinides do not go through the fuel factory and pass through MYRRHA. As a result, fuel factory reduces issues related to the management of minor actinides (compared to homogeneous transmutation).
- Fuel transportation is easier, cheaper and safer compared to a fuel containing minor actinides.
- MYRRHA's transmutation capabilities are almost 20 higher than those in an SFR, so it allows a higher content of actinides for transmutation purposes (30 to 50 wt. % MA content).

This option is when considering a large deployment of SFRs and not only the small scale project ASTRID. However, it brings an estimated additional cost (average cost per kWh) of the order of 20% (considering the treatment capacity necessary for a 60 GW_e fast reactor fleet) compared to transmutation carried out inside fast reactors (CEA 2012f). There are different transmutation modes that will be explained in the following sections. This scenario is based on the transmutation in ADS systems, where the management of minor actinides are decoupled from U and Pu management cycle.

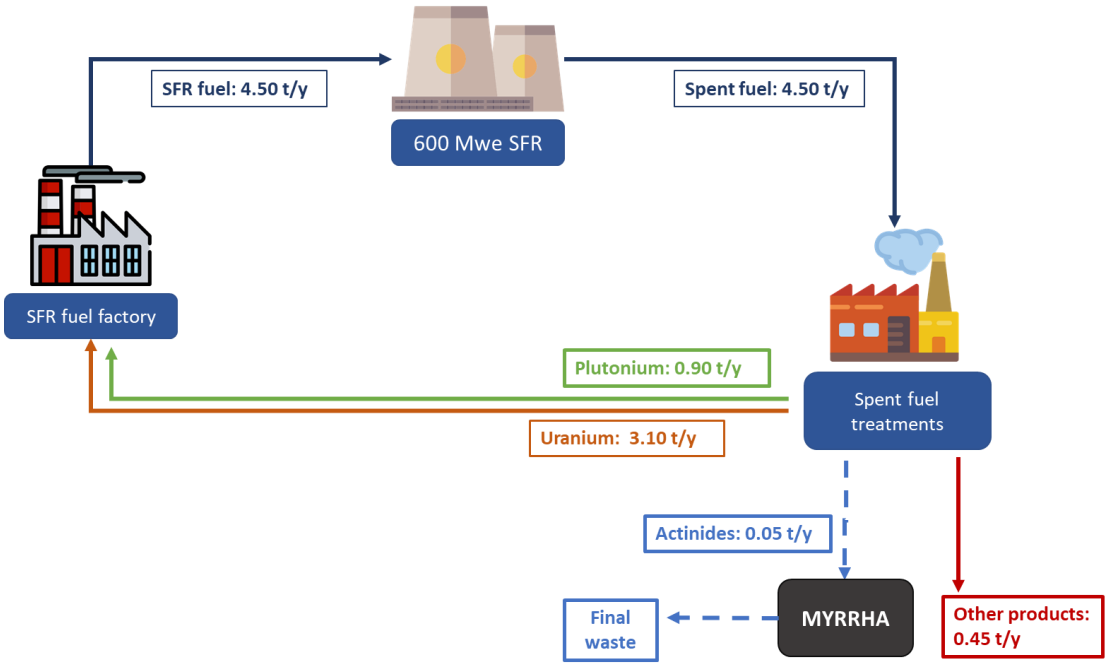

Figure 8 : ASTRID fuel cycle + MYRRHA (LGI,2018)

This scheme (Figure 8) does not include the installations around MYRRHA (MA-bearing fuel fabrication, recycling facility for the MYRRHA spent fuel…).

From the analysis of the fuel cycle of ASTRID and its possibilities, some findings stand out. One of the main objectives of the ASTRID demonstrator is proven: **transmutation of minor actinides inside the reactor is possible**. However, if a larger deployment of SFRs is foreseen, structural and capacity requirements must be fulfilled. Fuel fabrication must deal with different challenges such as the ability to recycle all grades of U and Pu present in the current fuel cycle while maintaining the economic viability. **Regarding MYRRHA, the coupling is more advantageous for fulfilling the needs of a larger deployment of SFRs** because the capabilities of MYRRHA are almost 20 times higher and allow a larger content of actinides (30 to 50% MAs content) than in an SFR. Transmuting minor actinides generated by the ASTRID demonstrator in MYRRHA does not bring a better performance than transmutation inside the reactor.

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2.2.ALFRED

ALFRED is the European project demonstrator of the Lead-cooled Fast Reactor (LFR) concept. The project envisages two possible pathways for the commercial development of the LFR concept:

- The first option considers **the demonstration of a small scale of LFR (100-125 MW_e)**, realized with engineering and technological solutions with a high readiness level and so it can be available in shorter-term. However, this option only envisages to evaluate an open-fuel cycle due to lack of interest in closing the cycle for a small scale.
- The second option considers **a larger reactor "European LFR" (600 MW_e)** relying on more advanced solutions. This alternative is envisaged for a longer-term than the first one. Considering a closed-fuel cycle is relevant because it is a more interesting option to study from the waste management's perspective and minor actinides partitioning and transmutation.

A final decision regarding the final pathway to follow has not been made yet at this stage of development. Therefore, this report considers both options: the ALFRED small-scale demonstrator of LFR to show that transmutation is possible inside the reactor and the European LFR (ELFR) is suitable for transmutation purposes in MYRRHA.

For ALFRED and all follow-on systems, pure lead is envisaged. Mostly because of economics and physical protection against Polonium hazard. In a hypothetical scenario, a MOX fuel with a similar composition as for the SFR demonstrator can be chosen. A general decision of the most suitable composition for LFR is not still made but it will depend on the scale of the project. The following table presents the % range of U and Pu for ALFRED and the ELFR.

Table 4 : Fuel composition for ALFRED and ELFR (Data provided by ENEA)

	ALFRED	ELFR
U	70-80 wt.%	78-82 wt.%
Pu	20-30 wt.%	16-20 wt.% [a]
MAs	N/A	1-3 wt.% [b]

The same two options previously presented for ASTRID, can be applied with similar results due to the common nature of a fast reactor ("Fast Neutron Reactors | FBR - World Nuclear Association" n.d.).

1. ALFRED (LFR) with a MOX fuel that will recycle itself the actinides generated

The first option considers the ALFRED demonstration project and a MOX within the range previously mention (20-30 wt.% Pu and 70-80 wt.% U). Calculations from the National Agency for New Technologies (ENEA) outlines the flux of ALFRED. The following schema gathers all these elements being 1.52 t/y the input flux and 1.08 t/y U, 0.32 t/y Pu, 0.01 t/y of minor actinides and 0.11 t/y of other fission products as outputs. The demonstration of the MA transmutation performances of the small-scale LFR is part of the mission of ALFRED.

As for an SFR, a repeated recycling of Pu to exact the maximum energy potential of the fuel is possible. However, it generates minor actinides (in this case 0.01 t/y) that are the main contributors to radiotoxicity. Compared to a general LWR, MOX fuel can be recycled several times and the quantity of minor actinides generated is up to four times lower.

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This option acknowledges the recycling of long-lived elements as main objective to reduce the long-term radiotoxicity of the ultimate waste. In the same way as ASTRID, if a fleet of LFRs is envisaged the design of a fuel facility is fundamental and must deal with several challenges such as the ability to recycle all grades of U and Pu of the current cycle while being economically viable.

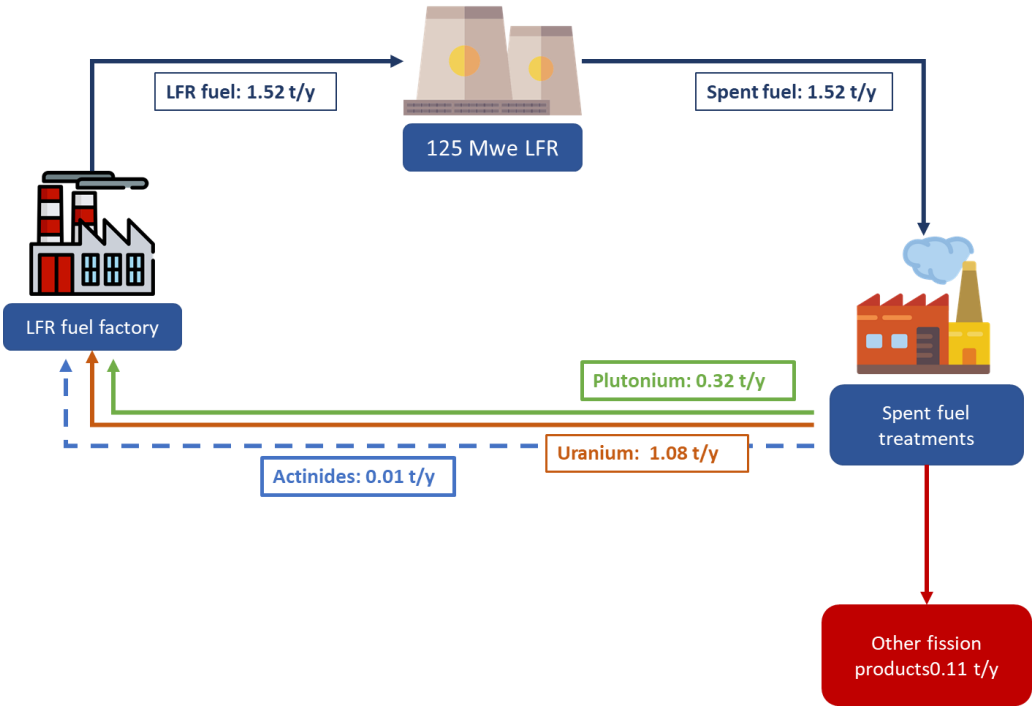

Figure 9 : ALFRED fuel cycle (LGI, 2018)

2. Integration of MYRRHA in the ELFR cycle

The second option considers that MYRRHA would treat the minor actinides generated by ELFR and the ELFR would perform the multi-recycle of Pu by using a MOX fuel (78-82 wt.% U, 16-20 wt.% Pu and 1-3 wt.% Mas). Although the coupling of ALFRED (small-scale) to MYRRHA is theoretically possible, it would have a very limited added value at small scale in terms of MA transmutation because:

- ALFRED is a small-scale demonstrator and is not meant for fleet deployment.
- The minor actinides burning capabilities of MYRRHA are limited to special elements that will be comparable to those foreseen in the special elements of ALFRED itself.

Therefore, a deployment of ELFR is more relevant for this option. Different transmutation modes are possible, but this case is based on an ADS system, where the management of minor actinides are decoupled from U and Pu management cycle. The following schema represents the fluxes of the cycle being the generation of minor actinides 0.16 t/y. Similar outcomes than an SFR fleet can be highlighted here:

- Transmutation processes carried out in MYRRHA reduces significantly the radiotoxicity of ultimate waste. Americium is target because it is the main contributor to long-term radiotoxicity. By transmutation it can be reduced by a factor of two.

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- Issues related to minor actinides management are reduced because they go directly to MYRRHA and do not pass through the LFR fuel factory. Therefore, transportation is safer and cheaper.
- MYRRHA allows 20 times higher content of actinides than the transmutation inside the LFR.

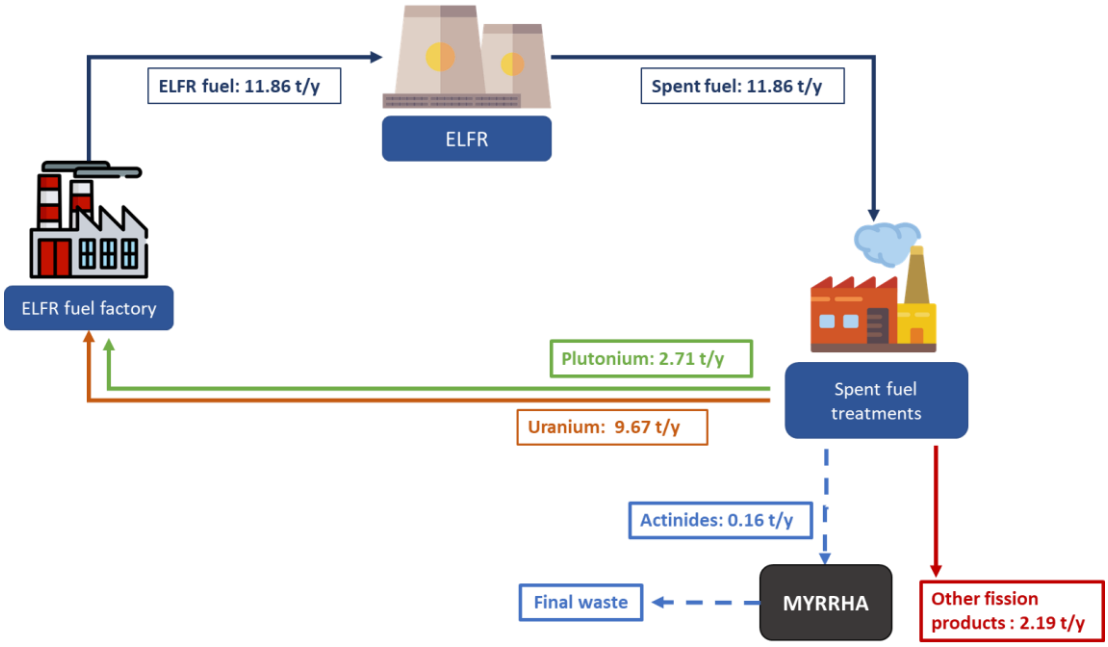

Figure 10 : ELFR + MYRRHA (LGI, 2018)

This scheme (Figure 10) does not include the installations around MYRRHA (MA-bearing fuel fabrication, recycling facility for the MYRRHA spent fuel…) to simplify the representation.

From the analysis of ALFRED several outcomes can be highlighted. As shown in Table 5, **the generation of minor actinides is ten times higher in the ELFR than the ALFRED demonstrator**, but this is because of the larger scale of the reactor. ALFRED's electric output is 100-125 MW_e and ELFR 600 MW_e. **ALFRED demonstrator of the LFR confirms that the transmutation of minor actinides is possible inside the reactor**. However, **the coupling to MYRRHA is only relevant for the ELFR** since there is no interest in applying this process in a small-scale reactor.

Table 5 : Flows of ALFRED and European LFR (Data provided by ENEA)

	ALFRED		ELFR	
	IN	OUT	IN	OUT
U	1.16 t/y	1.08 t/y	11.86 t/4y	9.67 t/4y
Pu	0.36 t/y	0.32 t/y	2.69 t/4y	2.71 t/4y
MAs	0 t/y	0.01 t/y	0.19 t/4y	0.16 t/4y
FPs	N/A	0.11 t/y	N/A	2.19 t/4y

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2.3.LWR

Light-Water Reactors (LWRs) are among the most common designs in the existing fleet of reactors, and the most widespread in the EU. They use light water (purified ordinary water) as moderator and coolant and require enriched uranium fuel to function. There are currently two types of LWRs: Pressurized-Water Reactors (PWRs) and Boiling-Water Reactors (BWRs) (nuclear-power.net 2018). However, since there are very few differences in the fuel cycle of these two designs, a generic LWR concept was explored here. FRs are supposed to replace LWRs sometime in the future, it is interesting to compare the fuel cycle of these two designs.

1. LWR with a mix of UOX/MOX fuel

Fuel cycles in which LWRs are fuelled with UOX and MOX are found currently in different countries in Europe and the world (e.g. France). In this fuel cycle option, the plutonium is recycled once ('twice-through' cycle). It will be used as a reference for comparison purposes with the other scenarios investigated in this deliverable.

For this cycle, information was provided by the CEA about the typical flows of a 60 GW_e reactor fleet, and then converted for a 600 MW_e typical reactor. Although 600 MW_e is not the typical power of a LWR, such power was chosen here in order to compare the flows with 600 MW_e fast reactor concepts (same power). In addition, no single reactor receives both UOX and MOX at the same time, but average flows are described here to give a broad picture of the 'twice-through' cycle (for a reactor fleet) (CEA 2012f).

The fuel cycle is the following: the LWRs are primarily charged with a UOX fuel. The irradiated fuel contains U (95% wt.), Pu (1 wt. %), fission products and minor actinides (mainly Am, Cu and Np, 4 wt. %) (Haut Comité pour la Transparence et l'Information sur la Sécurité Nucléaire 2018). Then, the fuel is treated, and the U and Pu are separated from the rest of the waste. The retrieved Pu is used to manufacture MOX fuels (mixed oxides fuels, containing a proportion of plutonium, usually not higher than 10 or 20 wt.% of Pu oxides depending on countries), and the retrieved U can also be used to manufacture fuels. Such a fuel cycle is described below: to simplify, in this case, there is no further use of the irradiated and separated uranium, and the depleted uranium is not represented. Also, note that the reactor receives either UOX or MOX fuel.

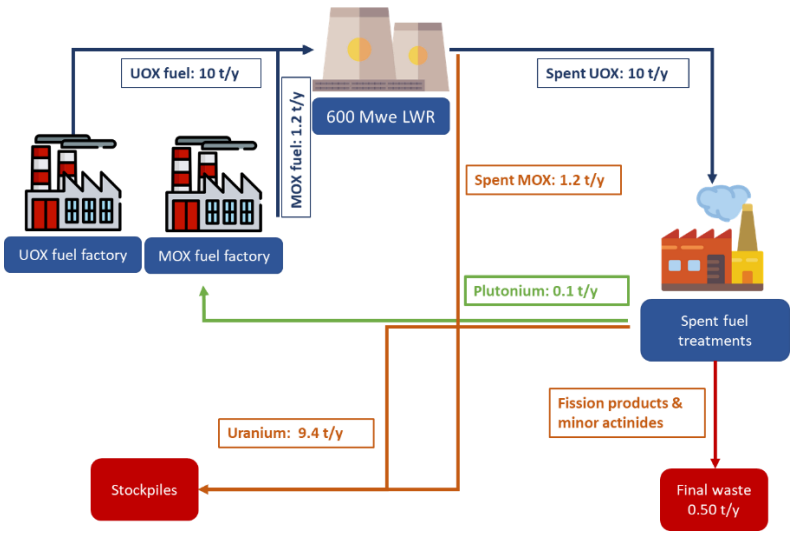

Figure 11: LWR fuel cycle (with stockpiling of irradiated uranium), adapted from (CEA 2012b)

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The main advantage of this cycle is that it is industrially mature: it has been in place in different countries for decades. It is also relatively simple and – compared to an 'open cycle' – it allows the recycling of the uranium and plutonium contained in spent fuel. However, such a cycle has a few drawbacks:

- It does not allow for the multi-recycling of plutonium (however, there are ongoing studies in some countries assessing the possibility for industrial-scale Pu multi-recycling in PWRs).
- The recovered uranium (from irradiated fuel) is not always exploited depending on economic conditions (Haut Comité pour la Transparence et l'Information sur la Sécurité Nucléaire 2018).
- Fission products and minor actinides are considered as final waste: no transmutation is considered. Therefore, final long-lived waste amounts are higher (from 2 to 100 times) than what could be achieved with MA transmutation.

2. Integration of MYRRHA into an LWR cycle

As described in section 2.3, MYRRHA is an accelerator-driven system which combines a particle accelerator, a spallation target and a sub-critical fast reactor. This section deals with the integration of MYRRHA within the fuel cycle of an LWR (as described in the previous paragraph).

Within the LWR 'twice-through' cycle, MYRRHA would be placed after the sequential treatment processes (PUREX or equivalent) which partition the spent fuel by separating U, Pu from minor actinides and other fission products. More partitioning would then be necessary to separate the minor actinides (Np, Am, Cm) from the fission products. The recovered minor actinides would then be transmuted by MYRRHA and the final wastes would then consist of remaining minor actinides and fission products (CEA 2012f). The cycle is described in Figure 12. Although MYRRHA could treat the minor actinides coming from MOX fuel, it has been feed it only with spent UOX fuel to simplify the figure.

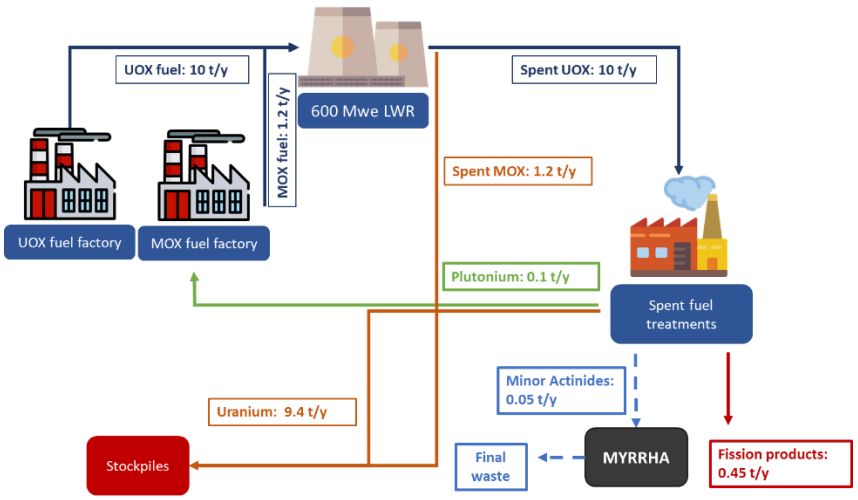

Figure 12: LWR cycle coupled with MYRRHA (adapted from (CEA 2012b))

This scheme (Figure 12) does not include the installations around MYRRHA (MA-bearing fuel fabrication, recycling facility for the MYRRHA spent fuel…) to simplify the scheme.

Compared to the previous LWR cycle, adding MYRRHA brings advantages (Baeten et al. 2014; CEA 2012f):

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- Potential for the transmutation of minor actinides, reducing the volume of minor actinides in the final wastes. Final wastes would then be composed of remaining minor actinides and fission products, which are shorter-lived.
- Being subcritical, MYRRHA's can be stopped much faster than critical systems: this reinforces the safety along the fuel cycle.
- Separating the fabrication of minor actinides-bearing targets and recycled nuclear fuel enable to limit the impact of minor actinides (MA) on the nuclear fuel cycle in terms of radiotoxicity, logistics (MA-bearing fuel requires more precaution for transport) and complexity of the fuel fabrication (CEA 2012f).

However, some drawbacks can also be mentioned:

- Due to a high MA content (30 to 50%), the fabrication and transportation of the MA-bearing targets is much more difficult than in the case of homogeneous/heterogeneous transmutation in fast reactors.
- Since MYRRHA has a limited power (100 MW_{th} maximum), it limits the exposition time and therefore the total amount of wastes it can handle. This is a constrain if the treatment should deal with the waste flow from a full reactor fleet.

In a nutshell, it is important to point out that ADS concepts such as **MYRRHA could efficiently transmute the MA** contained in the spent fuel of current LWRs, making it possible to reduce the long-term radiotoxicity of the spent fuel without having to wait for the deployment of a fleet of FRs. In addition, depending on the capacity of the ADS concepts, they can treat the spent fuel of **a large number of LWRs** and/or the stockpiled historical spent fuel, making their use more economical than a coupling with a single LWR.

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3. TREATMENTS FOR SPENT FUEL RECYCLING

This section presents different **processes for recycling the spent fuel**. Tremendous effort has been put on developing new reprocessing strategies for reducing radiotoxicity of used nuclear fuel to achieve a more sustainable nuclear energy sector.

The aim here is to describe the main aspects of potential European spent fuel treatment options and provide information about what components are recycled, how the process functions, what the steps are, among other relevant features. Separation research has mainly focused on hydrometallurgical processes because of the ability of these processes to fit into current treatment schemes, but also from a certain number of strengths as evidenced by the feedback from industrial experience: very high separation performance and limited quantities of technological waste. The waste treatment options presented here are especially important for the EU. They have been first presented and classified into three different groups depending on the elements recycled:

- **Partitioning processes**
  - **Sequential separation**
    - PUREX
    - DIAMEX-SANEX
    - I-SANEX
    - 1c-SANEX
    - EXAM
  - **Grouped separation**
    - GANEX (1st and 2nd cycle)
- **Transmutation processes**
  - Transmutation in fast reactors
  - Transmutation in accelerator-driven systems (ADS)

3.1. PARTITIONING PROCESSES

Partitioning considers the processes that remove certain elements from the final waste. It is considered along with transmutation as a way of reducing the burden on a geological disposal. Plutonium and minor actinides are the main responsible for the long-term radiotoxicity so that removing these nuclides from the waste (partitioning) and the fissioned (transmutation) reduces most of the long-term radiotoxicity. Two different type of partitioning processes have been included: sequential separation when selected elements are extracted from the waste separately and grouped separation when all the transuranic elements (Pu and all minor actinides) are removed as a block.

3.1.1. SEQUENTIAL SEPARATION

Sequential separation is the process of recovering specific and selected elements from the whole spent fuel by treatments that do not aim at recovering of all transuranic elements (Pu, Np, Cm, Am) at once (CEA 2012b). Three different and complementary processes are described below.

3.1.1.1. PUREX

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The PUREX process (Figure 13) applies a liquid-liquid extraction technique to separate and purify U and Pu. It uses an extractant molecule called tri-n-butylphosphate (TBP) that has the property of exacting cations at precise oxidation levels that corresponds to the oxidation of U and Pu (CEA 2012f). Other fission products including americium, curium and neptunium have different oxidation levels, which is why they remain in the process raffinate, making the extraction of U and Pu selective. Thus, PUREX mainly consists in extracting U and Pu, independent of each other, from the fission products. It is necessary to develop or add new treatments capable of extracting minor actinides to build a successful waste treatment strategy for fast reactors.

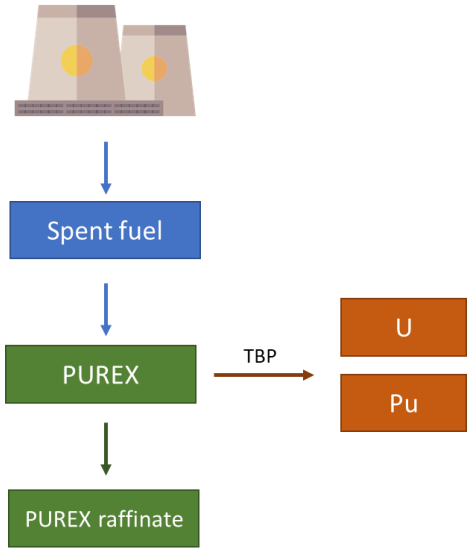

Figure 13 : PUREX process (LGI, 2018)

3.1.1.2. DIAMEX-SANEX

DIAMEX-SANEX (Figure 14) is a two-step sequential process that aims at: first, co-extracting lanthanides (Ln) and actinides from PUREX raffinates and second, separate lanthanides from actinides.

The DIAMEX process (DIAMide Extraction) is a solvent extraction process that co-extracts trivalent actinides and trivalent lanthanides from PUREX raffinate. The DIAMEX process was first developed at the CEA and tested at laboratory scale between 1993 and 2003. It was also continuously further developed within European collaborative projects (3rd, 4th and 5th Framework Programmes). It has been under study by ITU (Institute for Transuranium Elements, EC) and FZJ (Forschungszentrum Jülich, Germany) since 1998. Main information on the process can be found in the following table.

Table 6 : DIAMEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
An & Ln	DMBDTDMA	16	No data	TRL 6-7

The second step, R-SANEX (Selective Actinide Extraction) is a solvent-extraction process aiming to extract minor actinides from a DIAMEX raffinate. It separates lanthanides and actinides. It has been developed in the frame of European programmes since the early 2000s by many European research centres, namely ITU (Germany), the

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University of Reading (UREAD – UK), KIT (Germany), JULICH (Germany), CEA (France), and Chalmers University (Sweden). Main features are laid out on Table 7.

Table 7 : SANEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
An(III), Ln(III)	DMDOHEMA/ CyMe4-BTBP Or TODGA/BTBP	20	Not studied yet	TRL 4-5

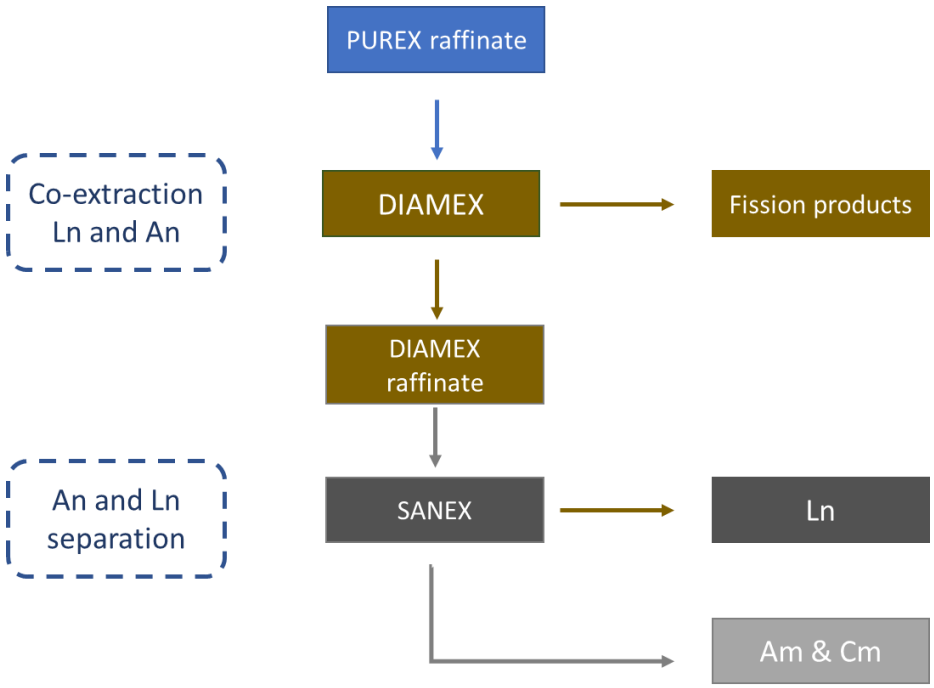

Figure 14 : DIAMEX-SANEX process (LGI, 2018)

3.1.1.3. I-SANEX

I-SANEX (innovative-Selective Actinide Extraction) is another type of SANEX process developed in the frame of the ACSEPT European project (FP7). The objective of the process is to extract selectively minor actinides (Am and Cm) directly from a PUREX raffinate so that the DIAMEX step can be avoided. Studies started in 2008 within the frame of different European programmes.

Table 8: i-SANEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
An (III), Ln	TODGA/CDTA SO_3-Ph-BTP or TWE21/HEDTA/CDTA SO_3-Ph-BTP	32	Not studied yet	TRL 4-5

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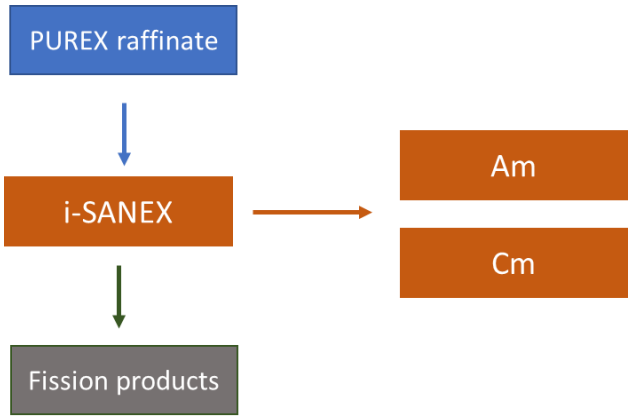

Figure 15 : i-SANEX process (LGI, 2018)

3.1.1.4. 1C-SANEX

1c-SANEX (1 cycle-Selective Actinide Extraction) is another solvent-extraction process developed in the frame of the European ACSEPT project (FP7). Its goal is also extract Am and Cm directly from a PUREX raffinate, avoiding the DIAMEX step. It has been developed since 2008 within the frame of European programmes by many European research centres headed by KIT (Germany). The main differences that distinguish this process from the other SANEX is the elements removed (Ln are not removed here), the main solvent used and the TRL status.

Table 9 : 1c-SANEX (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
An(III)	CyMe-Ph-BTBP/TODGA	32	Not studied yet	TRL 3-4

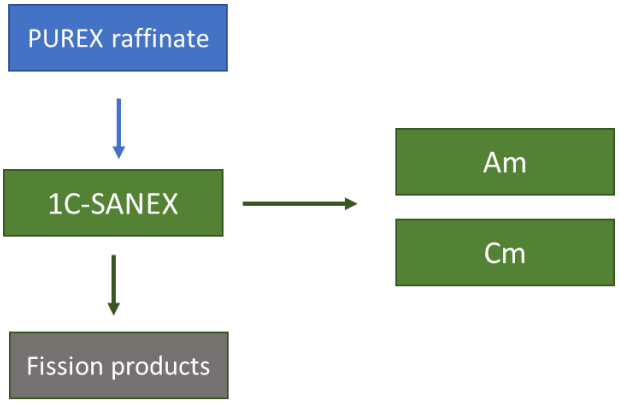

Figure 16 : 1c-SANEX (LGI, 2018)

3.1.1.5. EXAM

EXAm (Extraction of Americium) is a solvent-extraction process developed in France in the framework of the 2005 Waste Management Act. It aims at recovering directly americium alone. It has been developed in order to reuse the Am with uranium to produce $UAmO_2$ pellets. The process has been mainly studied at Atalante facility

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by CEA Marcoule for chemistry and process development, and by CEA Saclay for analytical purposes (Joly et al. 2015).

Regarding the status, it has been first studied at laboratory scale in 2008-2009. In 2009, the CEA first modelled a flowsheet for the process and cold tested it. In 2010, the scientists from CEA hot tested the EXAm process at Atalante facility. Since 2010, CEA studies have been essentially focussed on the improvement of the flowsheet with the purpose of treating concentrated PUREX raffinates (which would allow smaller dimensions of further industrial equipment). The following tabled explains the main information regarding this process.

Table 10 : Main information of the EXAm process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
Am	DMDOHEMA +HDEHP in HTP	68	Not studied yet	TRL 4-5

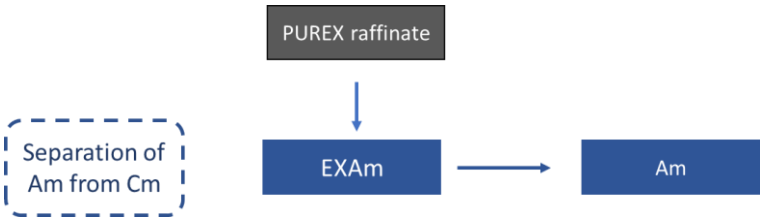

Figure 17 : EXAm process (LGI, 2018)

3.2.GROUPED SEPARATION TREATMENTS

Grouped separation treatments are block recycling processes. They recover groups of elements firstly, the bulk of U and then all the transuranic elements together (Pu, Np, Am, Cm).

3.2.1. GANEX 1ST CYCLE

The GANEX (Group Actinide Extraction) 1st cycle is a solvent-extraction process that aims to remove the bulk of uranium from high-level wastes. This concept has been developed in Marcoule (France) by the CEA. The results are promising, and it has been successfully tested at laboratory scale. This first cycle is common to the different possibilities in the second cycle (Joly et al. 2015)

Table 11 : GANEX 1st cycle (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
U, GANEX 2nd cycle feed	DEHiBA	28	Not studied yet	TRL 6-7

3.2.2. GANEX 2ND CYCLE

There are three possibilities that can follow the GANEX 1st cycle:

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- CEA GANEX

  The CEA GANEX is a solvent-extraction process aiming to extract all the transuranic elements from the GANEX 1st cycle raffinate. It has five steps and although the CEA has performed different tests at laboratory scale the results and the Actinide/Lanthanide separation rate is much lower than expected due to a too long equilibrium reaching time.

Table 12 : CEA GANEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
Np+Pu+Am+Cm	DMDOHEMA + HDEHP in HTP	48	Not studied yet	TRL 4-5

- EURO-GANEX

  The EURO-GANEX (European GANEX) is a variation of the CEA GANEX developed within the European project ACSEPT. With the same purpose of the CEA GANEX, this process was developed in order to co-separate all transuranic elements from the GANEX 1st cycle (Joly et al. 2015). This process has especially been studied at CEA (France), JULICH (Germany), KIT (Germany) and is currently under testing at UNISTRA (France), CIEMAT (Spain) and NNL (UK).

Table 13 : EURO-GANEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
Np+Pu+Am+Cm	DMDOHEMA & TODGA	30	Not studied yet	TRL 4-5

- CHALMEX

  CHALMEX (Chalmers GANEX) is a variation of the CEA GANEX developed within the European project ACSEPT. Like the CEA GANEX, this process allows to co-separate all transuranic elements from GANEX 1st cycle. The theoretical feasibility of CHALMEX has been assessed. However, tests don not have been performed.

Table 14 : CHALMEX process (Joly and Boo 2014)

Elements Removed	Main Solvent	Number of Stages	Evolution of the Volume of Wastes	TRL scale
Np+Pu+Am+Cm	$CyMe_4$-BTP & TBP	Not known for now	Not studied yet	TRL 2-3

The whole GANEX cycle (1^{st} and 2^{nd}) is summarised in the image below:

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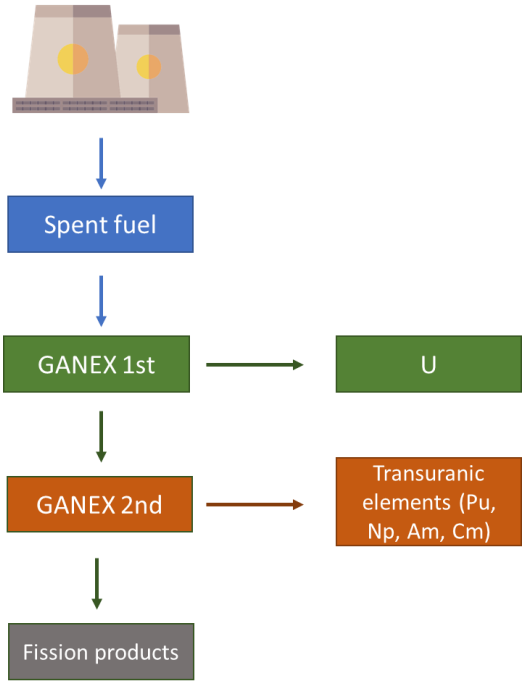

Figure 18 : GANEX 1st and 2nd cycle (LGI, 2018)

3.3. TRANSMUTATION OF MINOR ACTINIDES

Nuclear transmutation is the process of transforming one chemical element or isotope into another, by modifying the number of protons and/or neutrons in its nucleus. Regarding transuranium elements (i.e. Pu and minor actinides), there are two competing reactions for transmutation:

- Fission: the initial nucleus splits into lighter nuclei.
- Neutron capture : the initial nucleus merges with one or more neutrons, forming a heavier nucleus (CEA 2012e).

Fission is the reaction which is sought after in most cases, since neutron capture often only displaces the problem while fission often solves it (the lighter nuclei having less activity and/or shorter half-lives). This is one reason for which fast neutrons are preferred for transmutation purposes: they are more likely to produce fission reactions than thermal neutrons (CEA 2012e). Therefore, transmutation of minor actinides is linked with the deployment of fast reactors.

Different transmutation strategies exist and are presented hereafter.

3.3.1. TRANSMUTATION IN FAST REACTORS

The first option is to associate the transmutation of minor actinides to the production of electricity inside the core of a fast reactor. However, incorporating minor actinides into the core of a reactor changes its safety parameters, and therefore there are constraints on how to integrate them into the nuclear fuel. Some of the options available are (CEA 2012e):

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- **Homogeneous transmutation** : minor actinides are diluted in the fuel (not more than a few % for safety reasons). The advantage is to have a high neutronic flow for the transmutation. However, the exposition time of MA is necessarily the same as that of the fuel. Besides, the constraints linked with minor actinides (manufacture, transport, storage, etc) impact the whole nuclear fuel cycle which is more problematic (CEA 2012e). Existing fuel reprocessing facilities are not fit to deal with such constraints, so new capacities should be envisaged if this type of transmutation mode is to be chosen.

Homogeneous transmutation can be described in a simplified scheme (Figure 19). Minor actinides are treated along with uranium and plutonium.

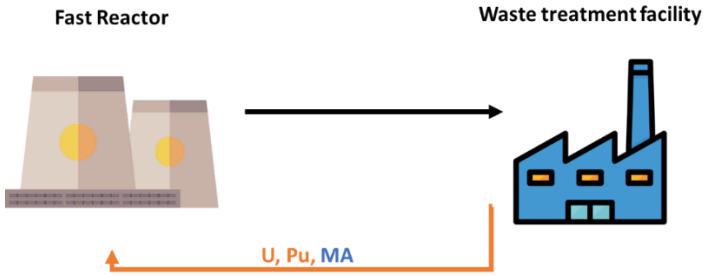

Figure 19: Homogeneous transmutation of minor actinides, adapted from (CEA 2012f)

- **Heterogeneous transmutation**: minor actinides are physically separated from the fuel. Manufacture of minor actinides bearing components are dealt with independently from fuel manufacture. There are two different possibilities for incorporating MA into the fuel:

    - Concentrating minor actinides in pockets near the core, with high concentrations (> 10%). With this composition, it could be possible to realise 'once-though' transmutation (minor actinides are transmuted in a single passage in the reactor). Even though this presents the advantage of incorporating high concentrations of minor actinides, the major drawbacks (compared to the blankets) are that it requires a longer time spent in the core (or a different neutron spectrum),and forbids the use of a uranium matrix for stability purposes (CEA 2012e).

    - Minor actinides-bearing blankets: the idea is to incorporate MA in blankets (10-20% concentration) placed at the periphery of the reactor core, so that safety is less affected. The advantage is to decouple fuel and blanket manufacture, so that minor actinides do not affect the whole fuel manufacturing processes. The drawback (compared to the pockets) is that the neutronic flow is less intense in the blankets, they have to spend more time in the reactor.

Heterogeneous transmutation, be it with MA in blankets or pockets, can be schematised as follows (Figure 20). Contrary to homogeneous transmutation, MA are treated separately from U and Pu.

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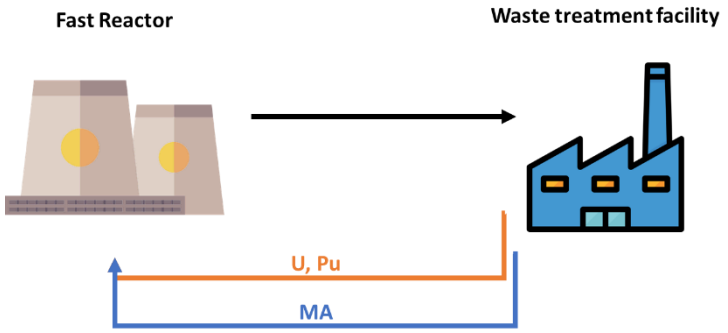

Figure 20: Heterogeneous transmutation of minor actinides, adapted from (CEA 2012f)

Compared to homogeneous transmutation, the heterogeneous mode makes it possible to:

- Limit the number of assemblies having to deal with MA.
- Not affect the core of the reactor.

Therefore, existing fuel reprocessing capacities could be reused for heterogeneous transmutation, with the addition of additional dedicated modules for MA-bearing fuel components. This reutilization is not possible in the case of homogeneous transmutation.

3.3.2. TRANSMUTATION IN ACCELERATOR-DRIVEN SYSTEMS (ADS)

Another option is to decouple the transmutation of minor actinides from the production of electricity. In this case, transmutation is performed in dedicated systems, separated from nuclear reactors. These systems are called accelerator-driven systems (ADS). As detailed in section 1.3, ADS are sub-critical entities that encompass a proton accelerator, a spallation target and a sub-critical fast reactor (with a lead-bismuth eutectic coolant in the case of MYRRHA).

Due to the sub-criticality of the system, it is possible to incorporate MA at a much higher level in the fuel: 30 to 50% (depending on some parameters of the ADS (CEA 2012e, 2), without affecting the safety of the reactor core. However, the drawback is having to deal with such a highly charged fuel during manufacture and transport steps.

Compared to transmuting MA in fast reactors, using MYRRHA brings some advantages (Baeten et al. 2014; CEA 2012f):

- Being subcritical, MYRRHA's can be stopped much faster than critical systems: this reinforces the safety along the fuel cycle.
- Separating the fabrication of MA-bearing targets and recycled nuclear fuel greatly reduces the impact of MA on the nuclear fuel cycle.
- It allows incorporating higher quantities of MA (30 to 50%) than homogeneous or even heterogeneous transmutation.

However, some drawbacks can also be mentioned:

- Due to the high MA content, the fabrication and transportation of the MA-bearing targets is much more difficult than in the case of homogeneous/heterogeneous transmutation in fast reactors. Some research

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and development is still needed in this field to ensure the safety of manufacture and transportation steps.

- Since MYRRHA has a limited power (100 MW_{th} maximum), a longer exposition time is required for a given quantity of MA and therefore the total amount of waste that MYRRHA can handle is limited. This is a constrain if the treatment should deal with the waste flow from a full reactor fleet.
- In the case of a fleet of fast reactors, CEA has calculated that transmutation in ADS rather than fast reactors means an overcost of 20% per kWh of electricity for 400 MW_{th} ADS concepts (CEA 2012f).

The transmutation in ADS is depicted hereafter, in the case of a fast reactor. The fast reactor would recycle itself its U and Pu, while the MA would be dealt with by the ADS, with a dedicated plant to manufacture the MA-bearing fuel.

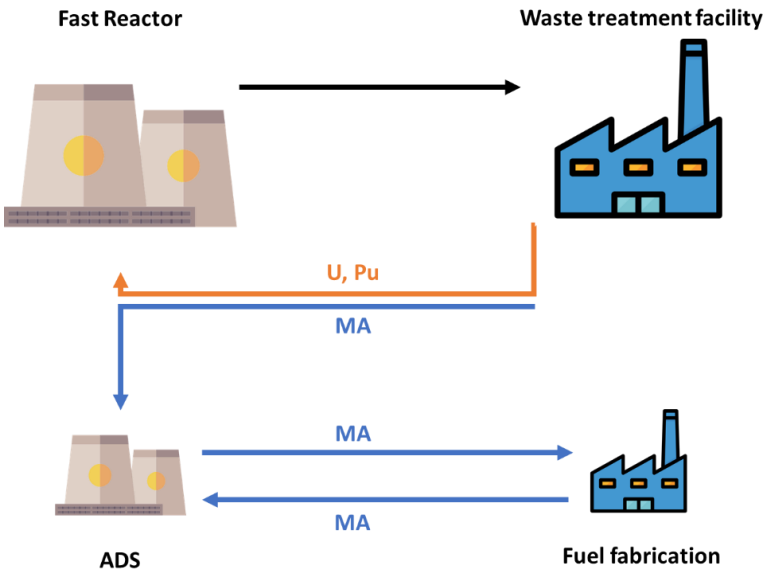

Figure 21: Transmutation of minor actinides in an accelerator-driven system, adapted from (CEA 2012f)

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4. RECYCLING SCENARIOS & REPROCESSING FACILITIES

After reviewing several spent fuel partitioning and transmutation possibilities in the previous section, this section focuses on the **description of three potential scenarios** for spent fuel recycling and waste treatment facilities. Information from Section 2 and 3 have been related to **determine which fuel cycle option and reprocessing processes can be applicable to the three recycling scenarios**. In addition, when information is available, each case explains how waste treatment facilities might be affected in the present and future, and the impact on several significant parameters such as the evolution of radiotoxicity of actinides and the final waste disposal surface (footprint).

Three scenarios are considered here and have been classified by the relevant elements recycled.

- **Scenario 1. Multi-recycling of U and Pu**
  Multi-recycling of U and Pu in fast reactors is key for improving the sustainability of the nuclear energy sector. It supports a closed-fuel cycle strategy because U and Pu can be reused to produce new fuel. Successive recycling of Pu modifies its isotopic composition and it becomes enriched in heavier isotopes. This is particularly true for fuels charged in LWRs, which are not very effective in Pu isotopes fission. They can only recycle MOX fuel once and so, less than 1% of the total energy potential of the initial uranium is actually used. On the contrary, fast neutron reactors favour fission over capture, limiting the accumulation of heavy isotopes and thus, allowing repeated recycling to fully exploit the potential of natural uranium (CEA 2012d). PUREX fits well into the first scenario and performs the separation of U and Pu for multi-recycling purposes.

- **Scenario 2. Multi-recycling of U, Pu and Am**
  The second scenario combines the recycling of U, Pu and one minor actinide only (Am). This strategy has been proposed because americium has been identified as the first target for a possible partitioning and transmutation strategy with the goal of reducing the long-term radiotoxicity and harmfulness of ultimate waste. As highlighted in results obtained by CEA in 2012, americium is the main contributor to radiotoxicity and decay heat for over 100 years in vitrified ultimate waste. In addition, the French national evaluation commission recommended in its 8th report that "*active, structured research on separation and transmutation has to be continued by focusing on americium first of all*" (Commission Nationale d'évaluation 2012). The recycling of americium is also quite interesting regarding the impact on the waste disposal surface (footprint). It is estimated that Am reduces by a factor of seven the surface for final waste disposal (CEA 2015).
  A combination of sequential strategies such as first, PUREX (for the extraction of U and Pu) and then, EXAM (selective extraction of Am) is one of the options envisaged for this scenario. Concerning transmutation, different options can be applied: a dedicated system (ADS) or transmutation in heterogeneous mode. Homogeneous mode was set aside due to complexities to reuse existing waste treatment capacities.

- **Scenario 3. Multi-recycling of U, Pu and all minor actinides (Am, Cm and Np)**
  The last scenario concerns the recycling of all major and minor actinides (U, Pu, Am, Cm and Np). Minor actinides other than americium have less significant long-term impact and the radiological and thermal threat is lower. However, a scenario that considers heterogeneous recycling of all minor actinides is relevant so as to have an integrated approach when dealing with spent fuels. For this case, several waste treatments can be envisaged. For instance, a combination of sequential treatments such as PUREX (U and Pu extraction), DIAMEX-SANEX (for the extraction of Am and Cm together). The variations of SANEX

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process (i-SANEX and 1c-SANEX) are also applicable here. Likewise, a grouped separation can also be considered by implementing GANEX 1st and 2nd cycles (EURO-GANEX). Depending on the strategy chosen, transmutation options can vary, going from homogeneous mode (all actinides are treated together), heterogeneous mode (major and minor actinides are treated separately) and transmutation in a dedicated ADS system (minor actinides are transmuted separately from uranium and plutonium management cycle) (CEA 2012f, 5).

Fuel cycle systems described in section 2 will now be **connected to the three recycling scenarios** presented above. A summary is presented in Figure 22. Due to the similarities and to avoid redundancies between ASTRID and ALFRED, they will not be studied separately in this section but treated as general Fast Reactors (FR).

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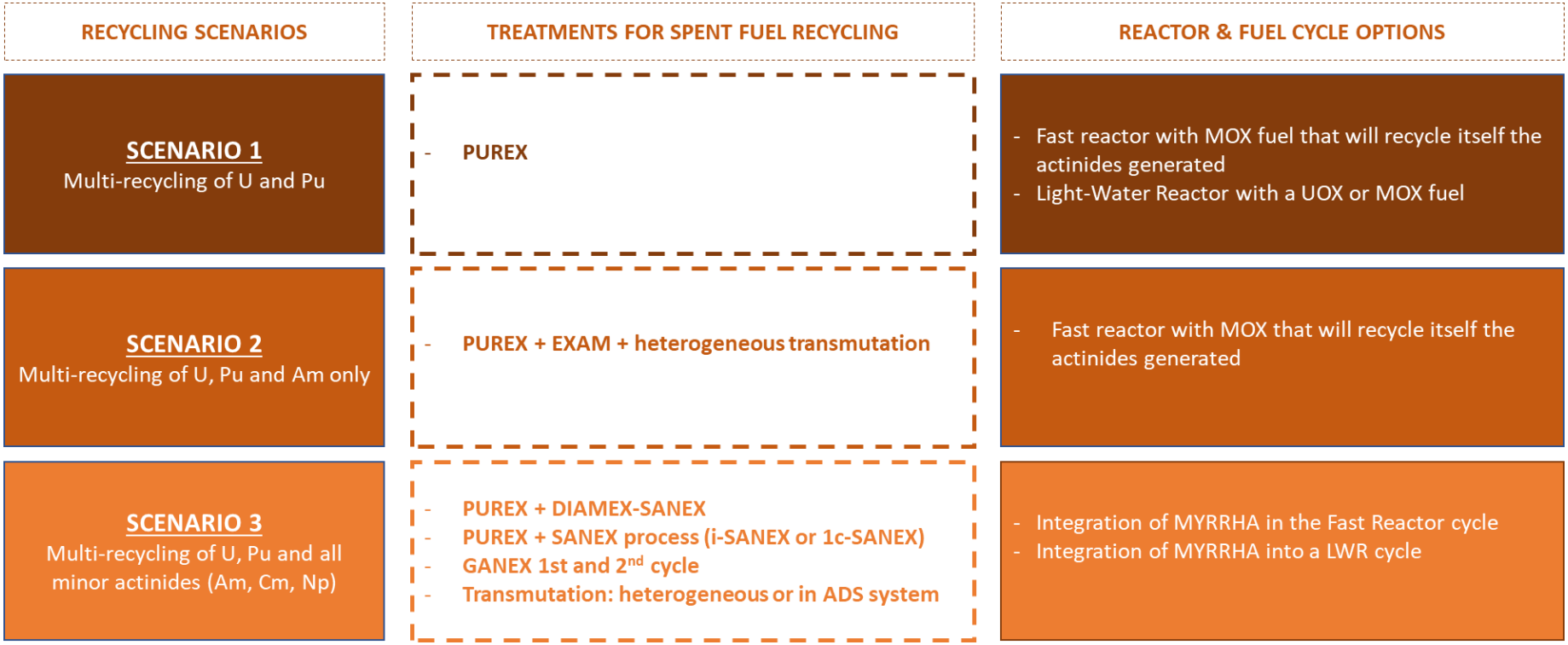

Figure 22: Summary of recycling scenarios, reprocessing processes and associated fuel cycles (LGI, 2018)

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4.1.FAST REACTORS

1. Fast Reactor with MOX that treating the actinides inside the reactor

Fast reactors (ASTRID and ALFRED project) are able to recycle the major and minor actinides present in the spent fuel. A repeated recycling of fuel materials can be achieved and so the extraction of the energy potential can be more valued than in LWRs. One consequence of the multi-recycling of Pu is the generation of minor actinides that is estimated at 0.05 t/year for ASTRID and 0.01 t/year for ALFRED. **Fast reactors will manage itself the transmutation of minor actinides**. Therefore, the most suitable recycling scenarios are:

- **Scenario 1** (multi-recycling of U and Pu): PUREX
- **Scenario 2** (multi-recycling of U, Pu and Am only): PUREX + EXAM + heterogeneous transmutation

Concerning fuel manufacturing, several solutions have been studied and tested in the Phoenix reactor to allow spent fuel to be recycled. It has been concluded that further research and development is necessary to optimise processes and so conceive a plant capable of supporting fuel treatment capabilities similar to those of the current nuclear fleet. There are several recycling issues that a **fuel fabrication plant for ASTRID and ALFRED** and later, for a potential fast reactor fleet, must address (CEA 2012c):

- The fuel plant must be able to recycle all grades of plutonium and uranium available in the current and future fuel cycle. In general, fuel manufacturing plants must be able to integrate the reprocessed plutonium from the UOX and MOX fuels from thermal reactors and spent fuels from fast reactors.
- It must maintain a high level of safety.
- It must operate with excellent availability to reliably supply reactors with just-in-time fuel while aiming for acceptable economic performance.

To demonstrate **the transmutation potential of minor actinides in fast reactor**, two possibilities can be considered: **the installation of an entirely new workshop next to them, or dilution treatment in an existing plant.**

The commissioning of a **stand-alone workshop** could be installed at the reactor place and take place in about ten years after the official start of operation of a fast reactor. For instance, the nominal capacity of the fuel fabrication workshop is expected to be around 5-10 $t_{heavy\ metal}$/year (CEA 2012b) to fulfil ASTRID's needs. It must be designed to allow future evolutions of processes and technologies that will enable fuel manufacturing of different specifications. First, it will be only dedicated for the multi-recycling of uranium and plutonium (Figure 23) for strategic purposes. This is actually an essential step when demonstrating the closed fuel cycle of fast reactors. An evolution towards the recycling of minor actinides is also expected when further technical improvements on transmutation experiments are reached. The priority should first be first to the recycling of Am only (CEA 2012b).

The second option envisages treating the spent fuel in **existing treatment plants** such as the one located in La Hague which would use existing units for a large part of the treatment operations. However, this plant might be decommissioned at the time a fast reactor starts operating. Likewise, technical modifications particularly at the plant head level are needed to optimally treat the spent fuel (CEA 2012c).

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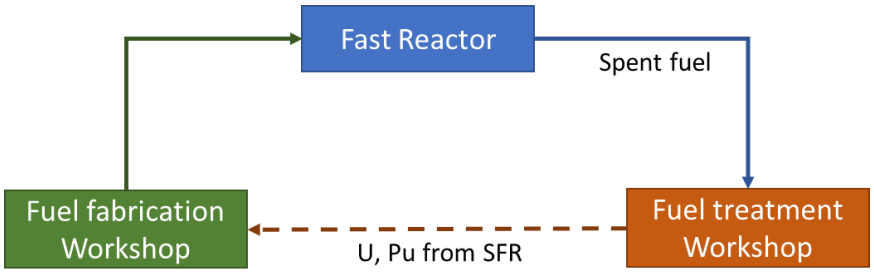

Figure 23 : Simplified schema of a fast reactor with MOX where actinides are treated in the reactor (CEA 2012b)

Focusing on the **transmutation of minor actinides**, homogeneous or heterogeneous modes can be possible at first sight. However, due to the complexity of homogeneous transmutation and constraints in terms of safety, number of assemblies having to deal with MAs, etc. **heterogeneous mode is seen as the most suitable solution**. Existing fuel reprocessing capacities could be reused for heterogeneous transmutation, with the addition of additional dedicated modules for MA-bearing fuel components. On the contrary, existing reprocessing facilities cannot deal with the issues of a homogeneous transmutation so new facilities should be envisaged if homogeneous transmutation was chosen. As mentioned in section 3.3, transmutation studies in fast reactors are particularly focusing on two options: minor-actinide-bearing blankets (MABB) or concentrating minor actinides in pockets near the core, with high concentrations (> 10%). Concerning economic viability, an additional cost of 5% to 10% per kWh is estimated for the implementation of transmutation options in a critical reactor (homogeneous or blankets, all minor actinides or americium alone).

Several studies have been conducted to **evaluate the feasibility of the existing facilities** (such as Atalante) for instance, fuel production and Am targets for irradiations in ASTRID (Figure 24). As a general result, they all need to be upgraded with new equipment to be able to fulfil this need. In the longer term, it could be decided to extend the implementation of specific fuel fabrication and treatment workshops (Figure 25) to ensure the separation of actinides and manufacturing the fuel associated (CEA, 2012c)

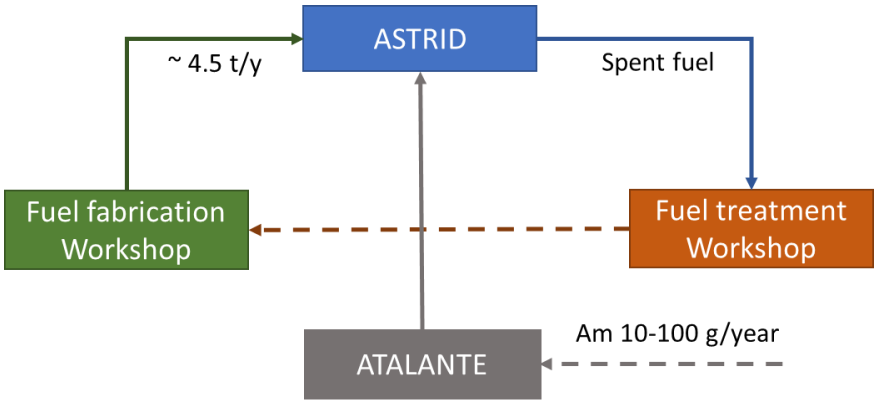

Figure 24: ASTRID reactor where minor actinides are treated at Atalante facility (CEA 2012b)

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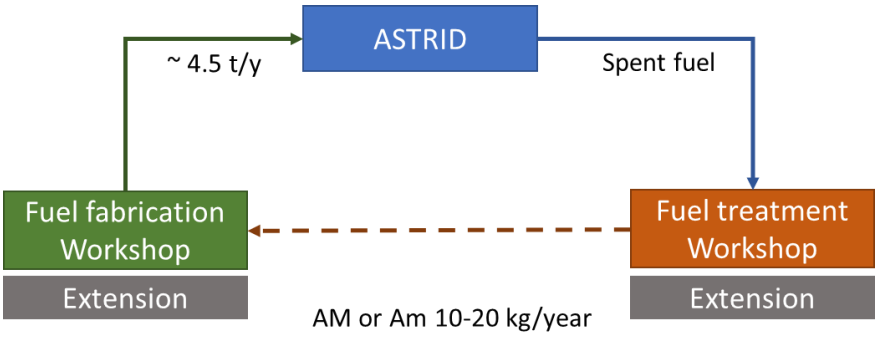

Figure 25 : ASTRID with MOX (extension of fuel fabrication and treatment workshops) (CEA 2012b)

2. Integration of MYRRHA in Fast Reactors' cycle

The second option proposed includes the coupling of a fast reactor to MYRRHA for transmutation purposes. Estimations of flow and actinides generated during the cycle have been presented in section 2 for ASTRID and ALFRED. Therefore, this section focuses on the transmutation and management of minor actinides.

The **recycling scenario 3: multi-recycling of U, Pu and all minor actinides (Am, Cu and Np)** seems to be the one that best fits its needs. The treatment strategy selected includes **GANEX 1st cycle** for the extraction of U and then, **GANEX 2nd cycle** for the extraction of all transuranic elements together (Am, Cm, Nm) and Pu. Concerning transmutation, it would be implemented in a dedicated ADS system (MYRRHA).

The implementation of transmutation options brings several advantages and issues that have to be taken into account (CEA 2012f, 5):

- The transmutation of minor actinides **reduces drastically the long-term radiotoxicity** by a factor of 20 or 100, depending on the time horizon considered. The main contributor to long-term radiotoxicity is Americium, whose radiotoxicity can be reduced by a factor of two. Figure 26 describes the evolution of the radiotoxicity of ultimate waste, considering three scenarios: multi-recycling of Pu in a fast reactor with transmutation, open cycle and multi-recycling of Pu in a fast reactor without transmutation. All these evolutions are compared to the baseline of natural U. As the graph shows, the greatest drop of radiotoxicity over time is achieved when the multi-recycling of Pu is combined with transmutation.
- The transmutation of minor actinides **is not an alternative to deep geological disposal but could be a long-term path for progress**. As shown in Figure 27, **the transmutation of americium is enough to achieve a 10-fold reduction in the footprint**. In addition, by transmuting both americium and curium, it can be obtained a long-term reduction in radiotoxicity. Thus, in less than 500 years, the radiotoxicity of the waste inventory returns to a level equivalent to that of all the uranium extracted to manufacture the current fuels.
- Transmutation of all the actinides results in a **reduction of the underground storage area for HLW by a factor 3 to 6** (depending geology and design) compared to the direct disposal of spent fuel (Lensa, Greneche, and Institut für Energieforschung 2008).
- On the other hand, the transmutation of minor actinides leads to an increase in their inventory in reactors and fuel cycle facilities. Considering a 60 GW_e fast reactor fleet, the inventory goes from twenty tonnes (without transmutation processes) to 60 to 160 tonnes depending on the transmutation scenario considered.

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- **Homogeneous recycling presents many constraints** regarding core safety parameters and remains difficult to assess at the current stage of development. Moreover, it leads to issues during handling and manufacturing operations of before and after of the irradiation of assemblies in the reactor.
- When **curium** is present in the assemblies to be transmuted, several issues appear regarding irradiation, thermal parameters, etc. On the contrary, recycling of americium brings less constrains.
- **MYRRHA allows a higher content of minor actinides** compared to transmutation carried out in fast reactors. It can range from 20 up to 50%.

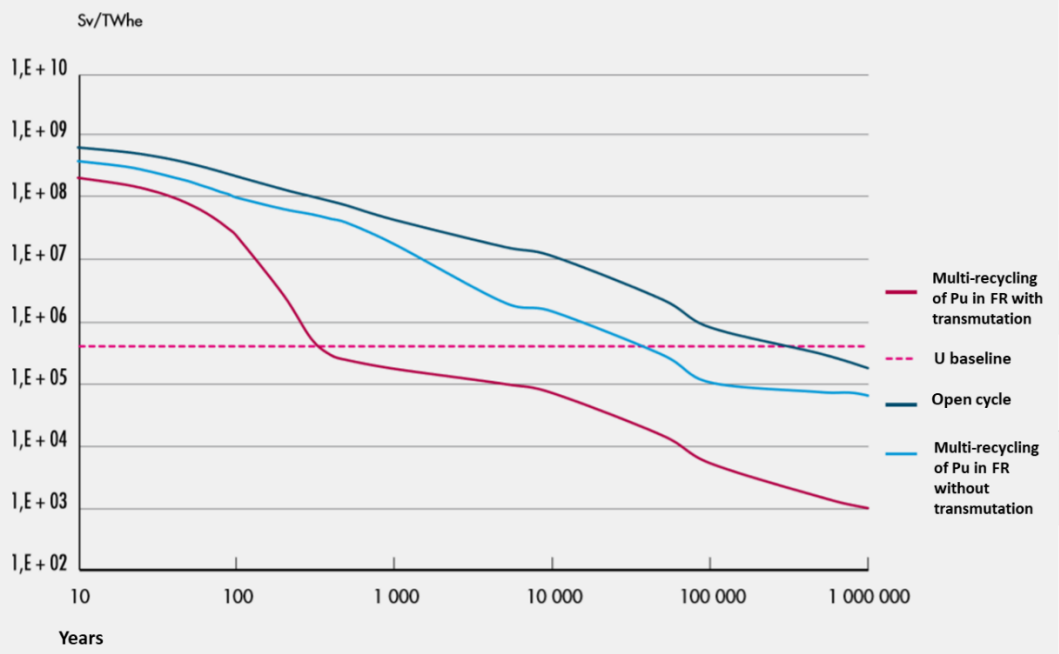

Figure 26 : Evolution of radiotoxicity of ultimate waste in different scenarios (CEA 2012f)

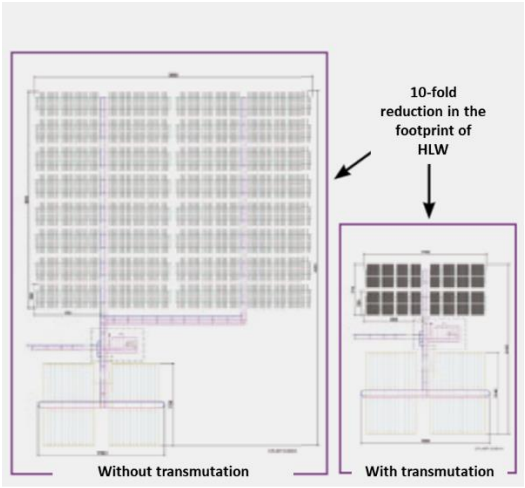

Figure 27 : Footprint comparison with and without transmutation (CEA 2012f)

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4.2. LIGHT-WATER REACTORS

1. Light-Water Reactor with a UOX or MOX fuel

As detailed in section 2.3, the LWR considered in this configuration is a 600 MWe reactor fuelled with either UOX or MOX (the same power as ASTRID for comparison purposes). In the case of UOX, the input flow is 10t/year, generating 9.4t/year of U and 0.1t/year of Pu, along with fission products and minor actinides. Then, depending on the country, the U and Pu can be retrieved thanks to a sequential separation process (PUREX or an equivalent). The Pu can be used to manufacture MOX fuel (with about 10% Pu) while the recovered U can be enriched to be used as a fuel again, depending on economic conditions (Haut Comité pour la Transparence et l'Information sur la Sécurité Nucléaire 2018). Such a strategy is followed by different countries around the world (with some variations), e.g. France, Japan (Lensa, Greneche, and Institut für Energieforschung 2008).

As mentioned in the description of scenario 1, successive recycling of Pu modifies its isotopic composition with more heavier isotopes. However, LWRs cannot fission efficiently these isotopes (low fission probability), therefore they can only recycle Pu once, through MOX fuels. Another study from the IAEA concluded that recycling Pu as MOX could be possible **at least twice**, but further recycling would require blending the Pu ex-MOX with other Pu sources (Hesketh, n.d.). In any case, **multi-recycling of Pu cannot be considered for a fleet of LWRs** (although there are ongoing studies assessing its feasibility) and the scenario 1 should be modified here to only include mono-recycling of Pu, as it is the case for the 'twice-through' cycle (CEA 2012e, 2).

Similarly, the transmutation of one or all the minor actinides is achieved more efficiently in fast reactors than in LWRs, principally because fast neutrons trigger more fission reactions than thermal neutrons (CEA 2012e, 2). Therefore, scenarios 2 and 3 will not be considered for LWRs (without ADS).

For the scenario 1, the relevant treatment process here is **PUREX**, described in section 3.1.1.1.

Regarding fuel manufacture, there are different steps for a classical UOX fuel:

- Uranium conversion, from natural uranium oxides to UF6.
- Uranium enrichment, where the percentage of uranium 235 is increased from 0.7% to about 4%.
- Fuel fabrication based on enriched uranium.

After being irradiated in LWRs, the fuel is treated in a waste reprocessing facility, where the PUREX process is applied to the irradiated fuel. Regarding the capacity of this reprocessing facility, it is important to mention that it should be adapted to the need for MOX fuels in the reactor fleet: it is this need that conditions the minimum capacity of the plant in terms of tons (of Pu) per year. For instance, the French reprocessing site in La Hague produces about 10 tons of Pu (heavy metal) per year, and **can treat a maximum of 1700 t of irradiated fuel** (Haut Comité pour la Transparence et l'Information sur la Sécurité Nucléaire 2018).

The fission products and minor actinides represent a **0.4 t/year** flow for a 600 MWe reactor. They are considered as a final waste in this configuration and are vitrified. In France, they are stored temporarily at La Hague reprocessing facility in order to cool down and let their activity diminish. The aim of the country is to deposit them in an underground storage facility (but not before 80 years of temporary storage (Haut Comité pour la Transparence et l'Information sur la Sécurité Nucléaire 2018)). The total volume occupied by these vitrified wastes is **1.5 m3** for a 600 MWe reactor. 1.3 m3 of structural wastes are also produced.

2. Integration of MYRRHA into a LWR cycle

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This sub-section presents the integration of MYRRHA within the LWR cycle. Calculations of flows were already detailed in section 2.3, this part focuses on the strategies of partitioning and transmutation of minor actinides.

Integrating MYRRHA into a LWR cycle makes it possible to treat a high proportion of minor actinides (Np, Am, Cm). Instead of vitrifying them along with fission products in order to store them as final wastes, the idea is to put into place a **transmutation of minor actinides in a dedicated structure: the ADS**.

Similar to the configuration with a SFR and MYRRHA, the **scenario 3: multi-recycling of U, Pu and all minor actinides (Am, Cu and Np)** seems to be the one that best fits here. In addition, the waste treatment strategy selected includes **GANEX 1st cycle** for the extraction of U and then, **GANEX 2nd cycle** for the extraction of transuranic together (Am, Cm, Nm, Pu).

Transmutation in an ADS requires a fuel composed of both Pu and minor actinides in an oxide form (CEA 2012e, 2). This is why the GANEX processes seems to be a good partitioning candidate for such a configuration since they co-extract Pu and MA in the 2nd step of the process. In the case of MYRRHA, the fuel considered is a MOX fuel bearing up to 30% Pu, and from 20 to 50% of MA (Angulo 2017).

However, fuel manufacture for MA transmutation in an ADS is a very particular and complex endeavour (CEA 2012e, 2):

- Fuels need to be free of uranium.
- They will bear a high MA content (potentially up to 50%), resulting in increased constraints in fuel manufacture and transportation due to high thermal energy and radiotoxicity.
- Behaviour of such fuels under irradiated conditions still needs some assessment.

Due to these specificities, fuel manufacturing for ADS will require **new reprocessing facilities or new modules in existing facilities**. As a matter of fact, the ADS fuel fabrication will require to collect (with partitioning processes) the Pu and minor actinides from the spent UOX and MOX fuels. After the irradiation in the ADS, more processes will be needed to minimise the losses of the cycle (Pu and MA remaining in the spent ADS fuel). In addition, special attention should be given to the matrix of the ADS fuel, with the possibility to recycle it (not considered here). Lastly, if the ADS fuel is chosen to be a nitrogen one, enriching the fuel in ^{15}N to avoid the creation of ^{14}C should be envisaged (we did not include this processing step in this study) (Lensa, Greneche, and Institut für Energieforschung 2008).

MYRRHA brings several advantages to the cycle, some of which are the same as for the coupling with the SFR:

- ADS can be **useful both for countries phasing out and countries carrying on with nuclear power**. As a matter of fact, countries phasing out could seek to eliminate their stockpiles of spent fuel by providing the ADS with both their Pu and MA to manufacture the ADS fuel. This could help dispose of the spent fuel by the end of the century. On the other hand, countries willing to pursue with nuclear energy could provide the ADS with their MA. By doing so, they **do not contaminate the reactor fleet** and they **do not need to be handled in deep geological repositories**. Both types of countries could share the use of the ADS, even though their policies are completely different (Hamid Aït-Abderrahim 2018).
- The advantages of transmuting minor actinides are similar to those described for the ASTRID scenarios (with the difference that ASTRID produces electricity). It diminishes the long-term radiotoxicity of the final wastes by a factor of 20 to 100, or at least by half for the transmutation of americium alone. Therefore, the remaining radiotoxicity would mostly come from fission products (apart from the remaining MA), which are much shorter-lived: in about 300 years, their radiotoxicity is comparable to that of natural uranium.

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- Regarding the footprint of the final wastes, it is important to mention that transmutation of MA by MYRRHA does not imply that projects for deep geological repositories of final wastes are no longer needed – the current vitrified wastes still need to be stored safely. However, removing elements with an important decay heat such as americium reduces the total footprint of the storage (the radioactive parcels can be stored closer to each other if they produce less heat).

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CONCLUSION

One of the European Commission's target is to move towards a more sustainable nuclear sector. Advanced fast reactor technologies will make the maximum of the available energy while new spent fuel management solutions will ensure higher safety, reduced long-lived and high-level radioactive waste and reduction of the final waste surface disposal.

The aim of the task 8.3 "Process mapping studies" and the present deliverable D8.3 was to offer an overview of the status for each ESNII concepts (ASTRID, ALFRED, MYRRHA), propose different recycling routes and assess the current and future possibilities of existing EU waste treatment facilities.

Within this study, **six fuel cycles options** from ASTRID, ALFRED project and general LWRs were described. Two of the fuel cycle options proposed (MOX fuel with actinides recycled in a fast-neutron reactor and integration of MYRRHA in the fuel cycle) can be applied to both ASTRID and ALFRED since the operating principle of fast reactors is quite similar. Differences come from the coolant, technical specifications and the consortium leading the project. The two other fuel cycles were proposed to **compare the possibilities between a fast reactor and a general LWR**, which is currently one of the most used technologies. It was shown that LWRs are limited in terms of partitioning and transmutation of actinides. LWRs cannot fission efficiently heavy isotopes and thus, they can currently only recycle Pu once, through MOX fuels. Therefore, **multi-recycling of Pu cannot be currently considered for a fleet of LWRs**. On the contrary, **fast reactors (ASTRID and ALFRED) allow the multi-recycling of Pu and minor actinides,** making a better use of the energy potential.

The possibilities and impact of partitioning and transmutation processes have been part of this analysis. **Nine different reprocessing processes were presented** and classified into sequential separation, grouped separation and transmutation processes. Then, **three recycling scenarios** have been proposed to link the information collected from fuel cycle options and treatments. From this analysis, different conclusions came up:

- **Fast reactors (ASTRID and ALFRED) fit better the recycling scenarios proposed**. Multi-recycling of Pu and Am alone is only possible for fast reactors through a combination of PUREX, EXAM and heterogeneous transmutation.
- **Removal and transmutation of minor actinides from ultimate waste reduces significantly the total long-lived radiotoxic inventory of the waste.** It can reduce the long-term radiotoxicity by a factor of 20 or 100, depending on the time horizon considered. The main contributor to long-term radiotoxicity **(Americium) can be reduced its radioactivity by a factor of two**.
- **Geological repositories to host the remaining waste remain necessary** regardless the preferred waste treatment procedure. However, only strategies including the **transmutation of americium achieve a ten-fold reduction of the final footprint**. Likewise, by transmuting both americium and curium, it can be obtained a reduction in radiotoxicity in long-live elements, which could be a factor of 100 beyond of a few centuries. In addition, results shown that the transmutation of all the actinides enables the **reduction of the underground storage area for HLW by a factor 3 to 6** (depending on geology and design).
- **The feasibility of spent MOX fuel processing for fast reactors has been demonstrated** both in pilot facilities and at La Hague plant but the deployment at industrial scale stills needs to be demonstrated.
- **Existing waste treatment facilities** could use existing units for treatment operations, but technical modifications will be needed to optimally treat the spent fuel from fast reactors. In addition, the existing facilities might be decommissioned by the time a fast reactor fleet is deployed so other alternatives have been considered. For instance, **the commissioning stand-alone workshops next to fast reactors**

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to deal with partitioning and transmutation purposes while avoiding waste transportation issues. In particular, ASTRID envisages this kind of workshop with a nominal capacity of 5-10 $t_{heavy\ metal}$/year. Facilities must be designed to allow future evolutions of processes and technologies.

- Coupling a fast-neutron reactor with an ADS such as MYRRHA makes it possible to **reduce safety constraints for the fast reactor and simplifies the logistics of its fuel cycle** (no need to manufacture MA-bearing fuel for the fast reactor itself, possibility to continue with existing facilities). However, ADS fuel manufacturing still needs demonstration at an industrial scale. In addition, the recycling of MA-bearing fuel will need more R&D since it is not industrially mature (this recycling is needed since several runs inside an ADS are required to transmute a significant part of the MAs).
- Transmutation through ADS systems allows to use **fuels with a higher proportion of MA** than transmutation in fast reactors, but they are also more complex to manufacture and to handle.
- ADS can be **useful both for countries phasing out and countries carrying on with nuclear energy**: countries phasing out could get rid of their stockpiles of fuel while countries pursuing it could solution their MA problems without contaminating their own fleet.

Scientific feasibility of partitioning and transmutation processes has been demonstrated. However, further research and development efforts are still required to demonstrate the viability of waste treatment facilities at industrial scale. Future work could focus on improving the reliability of the estimations on ecological, social and economic impacts, advanced fuel cycles and possible deployment scenarios of a fleet of fast reactors. This report will serve as input for next the task 8.5 “Impact studies” (M36) that will assess more in detail the impact of implementing a closed cycle across the full nuclear fuel cycle, identifying drivers, barriers, stakeholders involved, and deployment scenarios, among other topics.

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ANNEXES

SURVEY – T8.3 PROCESS MAPPING STUDIES (13RD JUNE 2018)

ASTRID

1. When is ASTRID (SFR) expected to be in operation? (in 2025 as planned or later?)

2. Fuel Material: Mixed oxide (U, Pu)O2
    a) What are the most used composition of MOX?
    b) Are there other compositions currently under study?

3. What is the content of minor actinides? Is it the same in the LFR?

ALFRED

1. What is the latest estimation of the commissioning for ALFRED (LFR)? 2025?
2. What are the possible fuel compositions?
    a. MOX: Pu content? Max 30%?
    b. MA-bearing fuels?
    c. Uranium nitrides?

MYRRHA

1. Existing transmutation processes: impact on existing spent fuel (MA)?
2. Potential coupling to SFR and LFR?

MOX fuel cycle

1. What are the most used MOX compositions for Astrid (SFR) and Alfred (LFR)? (select 3 ideally)
    a. Mixed oxide (U, Pu)O2 composition →
    b. ?
    c. ?

2. Spent fuel composition of each MOX. Same in ASTRID and ALFRED?
    a.
    b.
    c.

3. What minor actinides can be found in MOX?
    a. Americium
    b. Curium
    c. Neptunium
    d. Others?

4. The current MOX facilities are only in France, are there others?
    a. Areva's-Orano's MELOX Factory
    b. MOX Fuel Fabrication Facility Savannah River in South Caroline (2007)

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5. Where is MOX currently being used? In Light Water Reactors as well?

6. The reprocessing of MOX has been proved? In La Hague? Comments.

7. Do you know what are the waste treatment processes envisaged/possible routes? For Pu, U and actinides recycling.

   Options
   One-though recycling? Twice-through?
   Multi-recycling?
   Homogeneous recycling? (As a group, together with major actinides, by diluting them into uranium and plutonium)

   Our multi-recycling option
   - PUREX → for U and Pu recycling
   - DIAMEX-SANEX, GANEX, EXAM → Americium, Curium and Neptunium recycling

   Here you can find a potential multi-recycling strategy for the spent MOX fuel in Fast Reactors.

GANEX 1st cycle	CEA GANEX (Pu, Np, Am, Cm)
	EURO-GANEX (Pu, Np, Am, Cm)
	CHALMEX (Pu, Np, Am, Cm)

Waste treatment facilities

1. The current waste treatment facilities are in: La Hague (FR) and NDA Magnox (UK). Are there others?

2. Since these waste treatment facilities will be decommissioned by 2025 the strategy will be to build new ones and so, centralise the wastes from the Gen IV Fast Reactors and integrate all the processing and recycling treatments.
   a. What do you think?
   b. Do you see other possibilities?

SURVEY - ALFRED & LFRS

Concerning the coolant:

1. Has a decision been made regarding the coolant? Lead or Lead-bismuth eutectic (LBE)?
   If yes, why is this option finally chosen? (security, performance, etc)

Concerning the fuel:

1. What is the composition of the most suitable fuel for an LFR?
   a. MOX? What % of U, Pu and actinides?
   b. Uranium nitrides (% of components)
   c. Other?

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2. With this/these composition/s (at equilibrium), what would be the flows in terms of U, Pu, fission prod, minor actinides? (in tons/y)

Concerning waste management:

3. Can we couple ALFRED to Myrrha (ADS)? (for transmutation of minor actinides)
   Do you think this coupling could have added-values compared to a standalone fast-breeder reactor such as ALFRED?

4. Have any waste treatment strategies been already envisaged? (PUREX+DIAMEX, transmutation, GANEX, etc) If yes, Which one(s)?

5. What is the expected final waste storage/footprint?

SURVEY - MYRRHA

Concerning the fuel:

1. Regarding the fuel, is MOX still the considered fuel? What percentage of Pu? What percentage of minor actinides (MA) could we incorporate for transmutation purposes (Am, Cm, Np)?

2. What would be the annual capacity of MYRRHA (tons of metal/year)?

3. What could be the most suitable partitioning processes to produce an input fuel for MYRRHA? GANEX? DIAMEX-SANEX/1c-SANEX?

4. Could MYRRHA transmute all minor actinides (at least Am)? What could be expected efficiencies for transmutation processes (MA transmuted/total input MA)?

Concerning waste management:

5. What could be the value-added of coupling MYRRHA to a (fleet of) LWR? Of FRs? Do you think this coupling could have added-values compared to standalone fast-breeder reactors? What are the advantages or disadvantages?

6. How much is the final waste surface disposal reduced thanks to MYRRHA? What does it depend on?

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