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Feasibility of future prospects and transition scenarios in fast reactors for Minor Actinides transmutation and plutonium multi-recycling in the frame of the French law for waste management

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Waste management solutions can only be assessed within a global perspective that covers the entire fuel cycle (fuels, reactors, fuel cycle plants, logistics, disposal), not by only considering the performance of one of these aspects alone. Scenario studies making it possible to monitor changes in the fuel cycle over time are thus essential if we wish to highlight the advantages and disadvantages of the different options, while assessing the relevance and possibility of recycling plutonium and deploying transmutation methods.

Scenario studies give us an global overview of the fuel cycle and the waste produced over time, from the actual situation today up to a more balanced situation that may be achievable in several decades or so. Such studies allow us to determine the feasibility of transitioning from the current reactor fleet configuration to a future configuration that is either partially or completely composed of fast reactors. Such transitions are complex to manage in terms of material availability, recycling capacities, and accumulated waste stocks.

To assess the technical and economic feasibility of various options, a methodology has been developed which relies on the analysis of different criteria and indicators. This methodology has already been used to analyse the different transmutation options within the scope of the French Act dated 28 June 2006 and more specifically the Decree dated 16 April 2008 which specifies that: *"the CEA [...] must submit [...] a file by 31 December 2012 at the latest [...], which also includes the results of the technical and economic scenario studies taking into account possibilities for optimisation regarding the transmutation of long-lived high-level waste (LL-HLW), its interim storage, and its disposal in a deep geological repository. This file must examine the advantages of recycling minor actinides and their transmutation compared with their disposal in vitrified waste, the different recycling methods available (heterogeneous or homogeneous) [...], and the related technologies that can be used (new-generation critical reactors, sub-critical reactors controlled by accelerators)."*

The CEA then began focusing on the U-Pu fuel cycle and the industrial transition from the current French fleet to a future fleet incorporating sodium-cooled fast reactors (SFR). This was done by using the same methodology as that used to assess transmutation. This work falls under the French national radioactive materials and waste management plan dated 27 December 2013 with respect to the Act on the sustainable management of materials and radioactive waste.

This paper first describes the methodology and criteria proposed. The main computer tools used to provide input data for the criteria and indicators are then discussed. Thereafter, two applications of this methodology are detailed. The first assesses the results of having evaluated the transmutation options and the feedback collected. The second focuses on assessing the U-Pu fuel cycle and describes a possible scenario for the progressive deployment of SFRs in the French fleet.

1. METHODS AND CRITERIA

Several criteria need to be assessed in order to accurately analyse the various nuclear fuel and waste management solutions available. Numerous studies have been instigated with the following objectives in mind:

- i) Understanding the flow of materials in the nuclear fuel cycle, e.g. between the plants and the reactors
- ii) Analysing the prerequisites for the industrial-scale deployment of the relevant processes
- iii) Analysing the impact on waste storage and disposal solutions
- iv) Estimating the costs.

All these studies generate a great deal of results covering a broad range of fields; it is therefore essential that the most relevant data be chosen in light of our current state of knowledge in order to accurately evaluate the scenarios in question.

In an attempt to be as exhaustive as possible with respect to the criteria, we used an approach which involved formulating questions that covered fields; they are detailed and summarised below.

What are the characteristics of the materials and waste for each scenario and what are the related inventories?

This question covers:

- *The inventories of the materials used in the nuclear fuel cycle (plants and reactors)*
- *Waste assessment.* The waste in this case is high level waste (HLW) and long-lived intermediate level waste (LL-ILW) produced by the facilities that are required for each scenario. We must examine not only the quantities, but also some of the most important characteristics, e.g. radiotoxicity of HLW, decay heat of waste which has a direct impact on the requirements in terms of disposal solutions, interim storage, etc.

The radionuclides contained in HLW emit different types of radiation (alpha, beta and gamma), which have different biological effects on the human body. Measuring the dangerousness of radioactive waste is based on a principle that involves accounting for the contribution of each radionuclide and its impact on the human body, i.e. the radiotoxic inventory.

The radiotoxicity of a radionuclide - expressed in sieverts (Sv) - is determined by multiplying its own activity by a dose factor that is related to its ingestion or inhalation. Dose factors as defined by the International Commission on Radiological Protection (ICRP) depend on the type and energy of the radionuclide's radiation, on their behaviour in the human body once ingested or inhaled, and on the radiosensitivity of body tissue. The dose factors are the subject of regulatory documents applicable in France in line with the recommendations issued by the ICRP.

The radiotoxic inventory of waste represents the sum of the radiotoxicity values due to the ingestion of all the nuclides found in the waste in question. It corresponds to the dose received by a set of individuals having ingested all of the radioactive material in question. This value (associated with a source) is not based on a contamination scenario and assumes that the radionuclides are potentially free and available. Therefore, it must not be understood as a dosimetric impact that takes into account the barriers and transfer mechanisms from the source to the individual.

The radiotoxic inventory nevertheless remains one of the characteristics defining the inherent dangerousness of radioactive waste. For this reason, it has frequently been used in international exercises that compare different radioactive waste treatment and management solutions. This is also why we have chosen it as an indicator in our technical assessment of the different transmutation options.

What are the characteristics of the facilities (reactors, plants, storage facilities) and the transport requirements specific to each scenario?

This issue concerns the impact of scenarios on the design and function of industrial-scale facilities, e.g. fuel and fuel sub-assembly manufacturing requirements, treatment of spent fuel, etc.

The issue of transport is also taking into consideration seeing that, depending on the scenario, additional transport solutions may prove necessary with respect to configurations without transmutation. In some cases, the feasibility of the transport solution may also not be achieved. To answer such questions, our indicator in this case quantifies the annual number of fuel sub-assembly transport operations.

What impact will each scenario have on the deep geological repository?

The impact of these scenarios on the deep geological repository can be expressed in terms of surface area (underground horizontal surface occupied by the facility) or in terms of the total volume excavated to build the tunnels and cells.

These values reveal the size of the HLW and LL-ILW inventories for disposal, as well as the impact of their thermal characteristics. These characteristics will impact both the configuration and the layout of the disposal facility.

What kind of economic performance can be expected of each scenario?

This issue concerning the impact of the different cycle options on the cost of generating electricity has been covered in depth in several past studies (see ref. [OCDE-2006] et [RED-2007]). It can be expressed in terms of the production cost (in € per kWh electric), as well as in relation to a given reference scenario. The respective contributions of the fuel cycle and the reactors can also be represented for the different scenarios in question.

What industrial risks are inherent to each scenario?

This issue concerns the following subjects:

- Availability of the technologies deployed in each scenario.
- Level of potential technology progress in case of its integration into the nuclear fleet
- Flexibility of technologies when faced with context changes concerning the energy strategy (return to previous conditions, etc.), the regulations (releases, exposure limits, etc.) or technical choices.
- Consequences of failure (reliability of supply, etc.).

How much natural uranium is required for each option?

NB: Other criteria, such as proliferation resistance, are also relevant. Nevertheless, this criterion has not been applied in our methodology since the current state of knowledge in this field is deemed insufficient for the needs of our studies.

2. COMPUTER TOOLS

2.1 COSI COMPUTER CODE

The CEA uses the COSI computer code to simulate the fuel cycle (reactors and facilities). In this way, variations in the flows and inventories of nuclear materials used in the fuel cycle and the characteristics of the nuclear waste produced can be determined over time and in relation to variations in the nuclear power demand. Each of the relevant isotopes is monitored over time at each stage of the fuel cycle. Parameters such as the decay heat, activity or radiotoxicity are also assessed [1].

COSI is reference code worldwide in its field; it has been used in numerous benchmarking exercises, several European and within the scope of studies carried out by NEA experts.

A key advantage of COSI is its coupling with the simplified evolution code [VIDA-2006] which can be used to calculate changes in the fuel during irradiation on the basis of its initial composition, the burn-up rate at unloading and the irradiation time. CESAR relies on multi-parameter cross-section libraries stemming from neutronic calculations (APOLLO2 for light water reactors, ERANOS for fast reactors) [2].

CESAR is the reference evolution code used by Orano, e.g. for the reprocessing and recycling plant at La Hague.

There are uncertainties of various origins on the results, e.g. on the basic nuclear data, on the technical data, and on the modelling assumptions. It is reasonable to believe that these uncertainties will be more pronounced for fuels with high plutonium and minor actinide contents. Research focusing on reducing these uncertainties is currently underway at the CEA.

For our study in question, however, the level of confidence that can be given to these results is sufficient to highlight the main trends.

2.2 CASPAR TOOL FOR ECONOMIC STUDIES

CASPAR stands for "technical and economic calculations of fleet scenarios". It is an Excel program developed by the CEA to analyse the economic aspects of fleet scenarios. It has benefited from feedback from the former tool called ESTRAN used in studies for the 2012 file. CASPAR takes into account sizes effects, series-production effects and twinning effects with respect to the cost of reactors, which is largely preponderant compared with that of fuel cycle facilities.

2.3 CRISTAL CODE PACKAGE FOR CRITICALITY STUDIES

The calculations are mainly performed using the following two calculation methods:

- APOLLO2-MORET 4 in the CRISTAL v1.2 package

APOLLO2 code (version 2.5.5) with the CEA93 cross-section library (version 6) resulting from the European JEF2.2 assessment is used to calculate the k_{inf} , as well as to provide the cross-sections with 172 energy groups for fissile environments and structural materials, which can be used directly in the MORET 4 code.

The Pij module (collision probability method) in the APOLLO2 code is used to calculate the self-shielded, macroscopic cross-sections that are homogenised and possibly condensed. These cross-sections can then be used either in an APOLLO2 S_n calculation (resolution of the transport equation by means of the discrete-ordinates method) or by a MORET 4 calculation (by means of a Monte-Carlo method).

MORET 4 (version 4.B.4) is a Monte-Carlo code that can be used to calculate the effective multiplication factor (k_{eff}) of any kind of 3D configuration.

The different codes are linked according to validated procedures that have been qualified and adapted to each calculation configuration.

- TRIPOLI-4.4 is the reference code in the CRISTAL V1.2 code package.

TRIPOLI-4.4 is used with the JEF2.2 cross-section library and is a continuous-energy Monte-Carlo code used to calculate the effective multiplication factor (k_{eff}) of any kind of 3D configuration.

2.4 COMPUTER TOOLS FOR RADIATION PROTECTION STUDIES

The related source terms are determined by the DARWIN/PEPIN2 code package. The dose equivalent rates are then calculated using the TRIPOLI-4 code.

2.5 COMPUTER TOOL FOR THERMAL STUDIES

The ANSYS software package is used.

3. TECHNICAL AND ECONOMIC ASSESSMENT OF DIFFERENT OPTIONS FOR MINOR ACTINIDE TRANSMUTATION: THE FRENCH CASE

In a first phase, a decision was made to select contrasting material management scenarios in order to highlight the advantages and drawbacks of the various methods considered, and to identify the major trends. It must be pointed out that these scenarios have been designed to assess the options and not to attempt to describe the final industrial reality. The assumptions associated with these scenarios are based on industrial experience transposed as effectively as possible to these new material management options; they do not preclude any future process and technological developments.

The scenarios analysed in this first phase have in common the consideration that the current series of reactors would be renewed at constant installed capacity (60 GW_e) generating 430 TWh_e/year . 40 GW_e of light water EPR^{TM} -type reactors (European Pressurized Reactor) would be deployed between 2020 and 2040, followed by 20 GW_e of sodium fast reactors (SFR) between 2040 and 2050. The date of 2040 corresponds to a general hypothesis of a possible start of deployment of these SFRs which would also be consistent with the main renewal dates for power reactors and fuel cycle plants. The introduction of a second series of 40 GW_e SFR would take place from 2080 to replace the EPR^{TM} which have reached the end of their service life. Starting in 2100, nuclear power generation capability would consist entirely of fast reactors (Fig. 1).

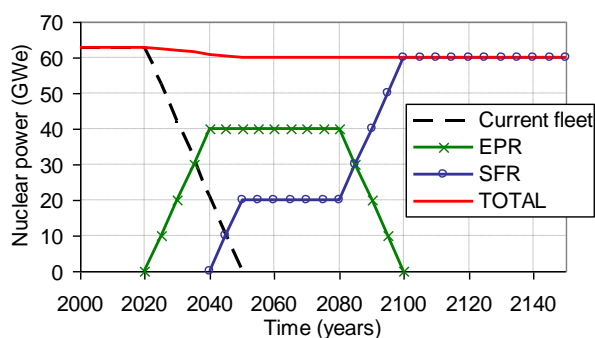


Figure 1: Production by reactor technology

Among the scenarios considering the deployment of SFRs, several differentiated alternatives have been selected:

- Recycling of plutonium only (scenario F4)
- Recycling of plutonium and transmutation of all or part of the minor actinides in homogeneous mode:
 - of all minor actinides (Np, Am, Cm) (scenario F2A)
 - of americium alone (scenario F2B)

- Recycling of plutonium and of all or part of the minor actinides in heterogeneous mode in radial blankets:
 - of all minor actinides (scenario F1G) (MABB: Minor Actinide Bearing Blankets)
 - of americium alone (scenario F1J) (AmBB: Americium Bearing Blankets)
- Recycling of plutonium in SFR and transmutation of minor actinides in a dedicated ADS stratum (scenario F7).

The SFR core concept considered has been developed by the CEA and French partners [3]. The accelerator driven system (ADS) model selected is the one designed for the EUROTRANS project (ADS Pb-EFIT) [4].

3.1 RESULTS OF SCENARIOS

3.1.1 Inventory

Contrary to a system of water cooled reactors, which inevitably produces increasing amounts of plutonium (reaching 1600 tonnes in 2150 for 60GW_e PWR fleet fed with UOX fuel), a system of plutonium recycling SFR would stabilise the Pu inventory at about 900-1000 tonnes. Stabilisation implies that SFRs are break-even core, namely, that they produce as much plutonium as they consume.

3.1.1.1 Main results at equilibrium and interim period

At equilibrium period, in the case of recycling Pu alone in SFRs without minor actinides (MA) transmutation, the minor actinide content in the irradiated fuel is about 0.4% (compared to about 0.1% in a PWR-UOX fuel).

Transmutation of MA means considering at equilibrium:

- A content of about 1.2% of MA in the fuels in the case of transmutation in homogenous mode; in this case all the SFRs are involved;
- or one row of radial blankets containing 20% of MA in 75% of the reactors of the system;
- or the deployment of 18 ADS of 385 MW_t in case of transmutation in dedicated reactors (an ADS with a higher power could certainly lead to more attractive performances).

Transmutation of americium alone requires at equilibrium in all SFRs:

- a content of about 0.8% of Am in homogeneous mode or ;
- one row of radial blankets containing 10% of Am .

The interim period proves to be more restrictive, because the quantity of MA to be transmuted and the number of fast reactors available are not necessarily matched. This is the case in particular between 2040 and 2080. This means:

- in homogenous mode, having to significantly exceed the limit of 2.5% of minor actinides in the fuel ;
- in heterogeneous mode, considering two rows of blankets containing 20% of actinides until around 2100.

These results are conditioned by the assumptions made in the scenarios, and in particular the limitations associated with them. Hence, they demand further optimisation of the interim phase to try to smooth the peaks encountered, which dimension the fuel cycle installations [5].

3.1.1.2 Plutonium availability

If the deployment of the first wave of SFR from 2040 raises no problem, it is important to anticipate the reprocessing of the spent fuels (cooling time reduced to 3.4 years instead of 5

years), or to alter the reactor concept in order to get the plutonium required to deploy the second wave of SFR (using radial blanket for example).

In these conditions, the deployment of a SFR reactor system as described in these scenarios appears feasible.

The deployment of ADS reactors is faced with a plutonium deficit of about 40 tonnes. Relaxing the limitations applied to the scenario would probably help circumvent this difficulty. Two alternatives can be examined, i.e. reducing the fuel cooling time and/or larger number of fertile assemblies.

3.1.1.3 Inventories of minor actinides

- Inventory in waste

The transmutation of MA helps considerably to limit the quantity present in waste, because with the exception of minimal losses, they are no longer automatically sent to waste.

If the non-recycling of MA means the continuous increase in their waste inventory (to reach nearly 400 tonnes in 2150), the transmutation of all the MA helps stabilise this inventory at around 60 tonnes, regardless of the transmutation mode selected (homogeneous, heterogeneous, ADS). Between these two extremes, the transmutation of Am implies a moderate increase in the waste inventory, due to curium and neptunium: the value reached in 2150 is about 150 tonnes.

- Inventory in the cycle

The immediate consequence of implementing the MA transmutation option is the increase of the MA quantities in the cycle inventory, as shown in fig. 2. It is worth to recall that the term "cycle" here designates all the installations of the loop followed by the actinides (fuel fabrication plant, reactor, reprocessing plant, spent fuel storage). The not recycled part of MA inventory is considered as waste and the sum up between recycled and waste forms the overall MA inventory. In the case of ADS, the high level of inventory in the cycle is explained as follow: although the reactor inventory is comparable with the transmutation in SFR, the short duration of the irradiation cycle due to the low power of ADS makes the inventory outside of the reactors (7 years: reprocessing + fabrication) proportionately larger. This term is dominant here.

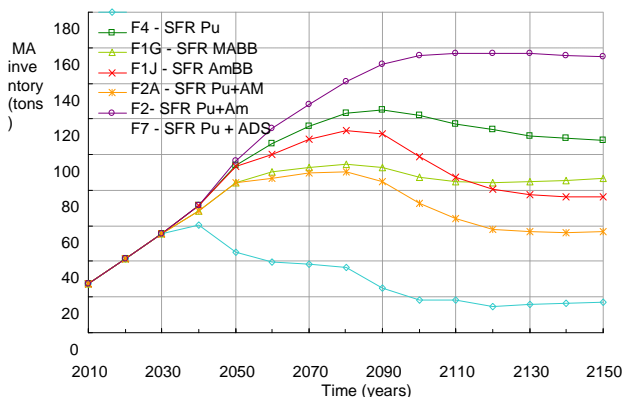


Figure 2: Inventory of minor actinides in the fuel cycle.

3.1.1.4 Some optimization of scenario

Transmutation in homogeneous mode:

The technical feasibility of the homogeneous transmutation remains limited to about 2.5~3.0% MA if one remains on the design of reference core SFR, with acceptable impacts on reactivity coefficients. Optimization of reprocessing strategy allows to limit the maximal MA content loaded in reactors: PWR MOX fuel assemblies are reprocessed over a longer

period thus more diluted, and since all fuels are reprocessed in “last-in-first-out” mode, the cumulated ^{241}Am production is reduced. Those changes lead to keep the MA content below 2.5% during the transient period of SFR deployment (Fig. 3). The same optimization gives a comparable reduction of Am content (maximum is 2.0%) for the scenario with only Am transmutation [6].

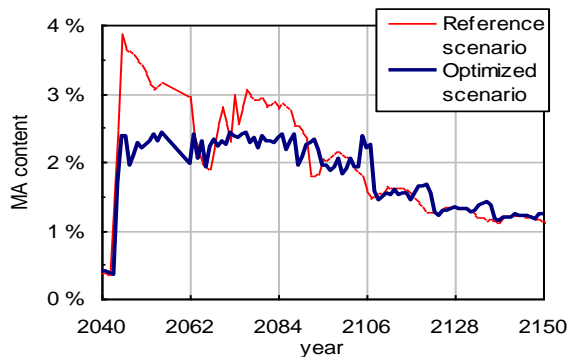


Figure 3: Evolution of the MA content in fresh fuel

Transmutation in heterogeneous mode:

The scenario has been optimized:

- by optimizing the reprocessing strategy (identical to the homogeneous case, see previous chapter).
- by reducing both initial Am content (from 20% to 10%) and irradiation time of AmBB (from 10 cycles to 5 cycles of 410 EFPD, which is the same duration than standard fuel) to increase loading frequency. This allows to reduce Cm build-up and consequently fuel decay heat.

It remains possible to use only one AmBB row with 10% Am in the optimized scenario. The thermal power of fresh AmBB is divided by 2, although Am inventory in fuel cycle is higher as illustrated: 115 tons at equilibrium instead of around 80 tons for the reference scenario. In our study, this new scenario is called “scenario F29” [6].

3.1.2 Impact on waste

CEA asked the French national radioactive waste management Agency (Andra) to assess the impact of high and intermediate level waste as produced by various transmutation options, on the sizing of a geological repository. Andra used repository architectures similar to those employed on the Cigéo project which is under development for current NPPs. Results allow to compare the underground footprint and the excavated volume for three scenarios (F4, F1G and F1J). The impact of the interim storage duration is also assessed. Solutions are proposed to optimize the footprint of the repository. An analysis of the advantages and drawbacks of transmutation options is provided [7].

HLW and LL-ILW are disposed of in separate underground zones. This arrangement offers independence in terms of (i) the management of the various types of waste and (ii) the phenomenological behaviour of each zone, in view of the specific characteristics of the waste contained.

The repository cells are constructed progressively with waste emplacement, according to a modular architecture which provides for a strict separation between mining and nuclear activities.

The underground facility includes (Fig. 4):

- A common infrastructure built prior to the operational phase of the repository,
- A disposal zone for LL-ILW,
- A disposal zone for HLW.

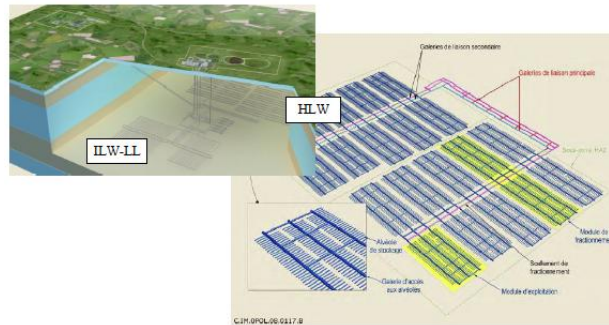


Figure 4: Potential layout of the repository

In the F4 scenario, the first study phase concluded that an increase of the interim storage period from 70 to 120 years would provide a gain of 25% on the footprint of the HLW zone and 7 % on the total excavated volume. The presence of americium in the waste restricts the densification of the repository because of a relatively low decay of the thermal power with time due to the long radioactive half-life of americium 241.

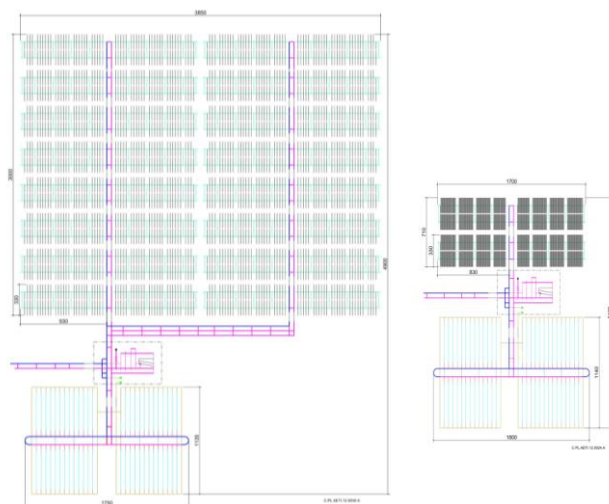
In the case of the transmutation of all minor actinides (F1G), the increase of the interim storage period to 120 years allows a larger gain (60%) of the footprint of the HLW zone and 12% of the overall volume excavated.

Based on the results of the first study phase, the second study phase consisted of a search for ways of optimizing the design with the objective of a more drastic decrease of the repository footprint. This second phase has considered only an interim storage period of 120 years for HLW. Indeed this assumption associated with transmutation scenarios provides a significantly higher gain than a 70 years interim storage period.

Figure 5 shows the repository layout as a result of this study.

Compared to the multi-recycling of plutonium in SFR, the transmutation of MAs associated with a design optimization of the repository would provide:

- a reduction by a factor up to 7.3 (Am) to 9.8 (MAs) of the footprint of the HLW disposal zone after an interim storage period of 120 years;
- a total reduction by a factor 3 of the repository footprint taking into account ILW-LL and common infrastructures;
- a total reduction by a factor 2 of the excavated rock volume.



Without transmutation (F4)

Transmutation of MA (F1G)

Figure 5: Underground repository architectures with an interim storage period of 120 years

Along with a reduction of the footprint of the HLW disposal zone, the partitioning and transmutation of actinides also decrease the duration of the thermal phase. After an interim

storage period of 120 years, the thermal phase is reduced to about 200 years against thousand years without transmutation.

The partitioning and transmutation of actinides do not reduce the long term radiological impact of the deep geological repository. Indeed this impact is dominated by long lived fission and activation products (APs) with a higher mobility in the geosphere (iodine 129, chlorine 36).

In the normal long term evolution safety scenario, the study shows that the densification of the repository as allowed by partitioning and transmutation does not significantly change the radiological impact of fission and activation products, despite concentrations in the near field increase with densification.

In an altered evolution scenario such as intrusive drilling, the impact of FPs and APs may increase to some extent because of the densification of the repository. Nevertheless this impact remains acceptable with regard to the dose limit provided by the basic safety guide issued by the French nuclear safety authority (0.25mSv/y).

3.1.3 Impact on the cycle facilities

3.1.3.1 Impact on manufacturing

Powder metallurgy processes comparable to that employed in the MELOX plant can be considered for the fuel manufacturing. Such processes include successive steps involving powder preparation, the manufacture of pellets via lamination and sintering, the manufacture of rods and the installation of the sub-assemblies. Nevertheless, the presence of minor actinides requires reinforced shielding systems (neutron emission and gamma radiation). Furthermore, the heat released by the radioactive materials involves making specific provisions for controlling temperatures during the manufacture of minor actinide-bearing sub-assemblies. However, as americium and the mixture of minor actinides are less penalizing than plutonium, controlling the criticality risk, which is possible by means of the customary control procedures, is not problematical.

Table 1 compares the thermal power and neutron emissions of the oxide powders used for manufacturing the fuels. These values are standardised with respect to the SFR fuel without minor actinides.

TABLE 1: Thermal power and neutron emission of powders during manufacturing (at equilibrium)

	MA transmutation				Am transmutation	
	F4	F2A	F1G	F7	F2B	F1J
Fuel	Fr-core	Fr-core	MABB	ADS	Fr-core	AmBB
Thermal power (W/kg)	1.5	9	55	160	3	7
Neutron emission (relative value)	(1)	(120)	(1600)	(3500)	(1.4)	(3)

It is clear that the scenarios involving the transmutation of all MA are extremely impeded by the presence of curium, and particularly of its ²⁴⁴Cm isotope. The manufacture of the fuels would obviously require the construction of a shielded enclosure with remote-controlled devices. With high Cm content, whole new technology development will be required (more pronounced for ADS option). The scenarios with transmutation of Am are less restrictive for manufacturing operations [8].

3.1.3.2 Impact on manufacturing

Minor actinide-bearing sub-assemblies are processed in facilities designed to receive the different types of spent fuel to be recycled (UOX, MOX, SFR). Processing operations are assumed to be based on the hydro-metallurgy processes such as it is the case at La Hague, including for ADS fuel. ADS fuel can also be processed by pyrometallurgy. The processing facility is broken down into workshops. Most of the workshops are dedicated to separate the nuclear material from metal structures and to place it into solution, to separate via extraction cycles with solvents the chosen elements, to convert the separated elements into oxides and to condition structural waste and fission products.

The scenarios including the transmutation of MA obviously involve the implementation of a process for separating chosen elements. According to the scenarios, this process may consist in a sequential separation process (DIAMEX-SANEX), in which the actinides are recovered separately, in a combined separation process (GANEX), in which plutonium and MA are extracted together, or in an individual separation process (EXAM), in which Am can be recovered selectively. All of these processes are still being developed. Their feasibility has been demonstrated at the laboratory scale, however, their industrial implementation still requires a long R&D process.

Studies on reprocessing facilities for core fuels recycling (MA or Am) or bearing blankets recycling (MA or Am) do not show important difficulties. But scientific and technical feasibility has to be investigated for ADS spent fuels.

Criticality constraints have been analysed in preliminary studies: additional analyses are required for specific functions as conversion of product containing curium. Thermal and radiation constraints have also been considered: the controlling systems and biological shielding are to be reinforced, particularly during the minor actinide conversion step [8].

3.1.3.3 Impact on transport

The heat released by new or spent actinide-bearing sub-assemblies can be a problem for transport operations between the reactors and the fuel cycle plants (Table 2).

TABLE 2: Thermal power of new and spent sub-assemblies (equilibrium will be reached around 2130)

	F4	MA transmutation			Am transmutation	
		F2A	F1G	F7	F2B	F1J
Fuel	Fr-core	Fr-core	MABB	ADS	Fr-core	AmBB
New S/A (kW/ass)	0.3	1.3	9	7	0.6	1.3
Spent S/A (5 years) (kW/ass)	1.3	2.2	11	7	1.8	5

So impacts of transmutation scenarios on fresh and spent fuels annual transportation have been evaluated. Thermal, radiation and criticality constraints have been taken into account to propose cask concepts for normal conditions as shows figure 6.

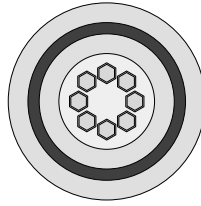


Figure 6: Example of a cask concept for 8 S/A

No difficulties appear for Am transmutation scenarios (homogeneous or heterogeneous). When fuels contain curium, transport uncertainties increase because of important heat release requiring dividing fresh fuels and technological innovations development (MABB and ADS).

The number of canister to be transported and the number of transport journeys required are significant factors for assessing the scenarios (Figure 7) [8].

It must nevertheless be pointed out that these assessments only take into account routine transport conditions. It would be advisable to further enhance the accuracy of this study by taking into account accident conditions. In this case, the issue of transporting sub-assemblies with minor actinides would be exacerbated.

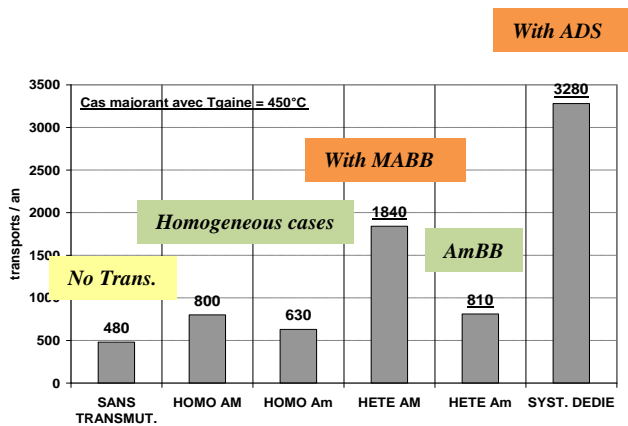


Figure 7: Annual transport operations of fresh and used fuels.

3.1.4 Impact on the reactor

3.1.4.1 Impact on the core

The homogeneous transmutation process will degrade the safety coefficients of the core more or less significantly according to the minor actinide content being considered. A limit of 2.5 % is generally accepted for a large SFR core.

In the case of the heterogeneous transmutation (MABB and AmBB), the MA to be transmuted are introduced on the periphery of the core and, even with a high content, their impact on the reactivity coefficients remains marginal.

3.1.4.2 Impact on the handling of fuel

Some specific biological shielding or cooling system will probably be necessary for handling sub-assemblies containing MA and particularly for MABB.

The handling of spent sub-assemblies is greatly conditioned by their residual power. At present, there are two limits:

- For unloading S/A out of the reactor vessel, the maximum acceptable powers amount to 7.5 kW if the environment is gaseous (possible solution for the European Fast Reactor) or 20 kW if the operation is performed in the liquid sodium coolant (case of a transfer bucket such as on Phénix or Superphénix). Research studies have been conducted on the possibility of increasing this last value to 40 kW.
- Concerning the underwater storage of the S/A, it will be necessary to clean them beforehand in order to remove all traces of sodium. At present, the maximum power for cleaning is of 2.5 kW. Research work is being performed to reach 7.5 kW.

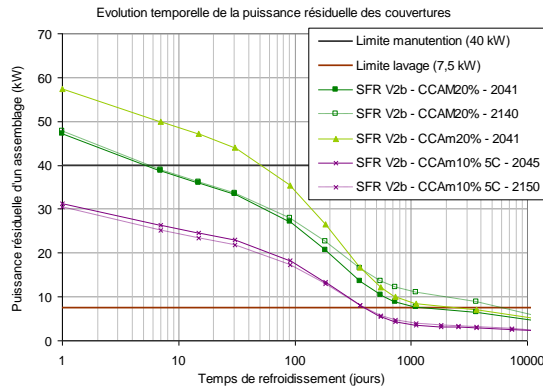


Figure 8: Residual power of spent sub-assemblies

The impact of the implementation of MA bearing blankets is obvious on figure 8. Even when considering a transfer in sodium, the AmBB with 20% of Am could not be unloaded out of the reactor vessel before around fifty days. Therefore, providing a storage area inside the vessels appears to be a compulsory requirement.

Cleaning the MABB and AmBB with 20% of MA or Am could not be envisaged before at least 15 and 7 years, respectively, and probably later since the uncertainty on the residual power has to be taken into account. This means that an intermediate storage area (outside of the reactor vessel) in sodium is necessary and it must be sized accordingly.

The optimization of the AmBB concept as we explain in 3.1.1.4 allows to reduce these drawbacks (AmBB, 5 cycles instead of 10 cycles with 10% of Am). In this case, there is no technical impact on the reactor vessel of the reactor.

3.1.4.3 Impact on the external storage

The external storage is a buffer used to store the new assemblies temporarily before loading and to store the spent assemblies while waiting for their decay heat to become compatible with the washing device. The size of the external storage depends on precise knowledge of the decay heat of the spent fuel assemblies and on the allowable power of the washing device. To account for these different uncertainties, we chose two wrapper values for the power of the washing device: 7.5 and 4 kW. With this hypothesis and with the value of the decay heat (see Fig. 8), we can define the size of the external storage and the cooling delay for the type of assemblies. The results are given in Table 3.

The assessment of the size of the external storage shows the impact of the couple decay heat and washing device, especially in the case of a low, allowable power of the washing device. In the worst case, it was necessary to wait for 60 years before washing, which reveals the very long life time of the external storage: about 120 years, 60 years for reactor life time plus 60 years for the cooling of the latest fuel assemblies [9].

TABLE 3: Main features of the external storage

Allowable power by the washing device	7.5 kW		4 kW	
	unities	years	Number of positions	years
Only Pu fuel	0.5	242	1.2	362
AmBB 20%	7	310	60	928
MABB 20%	15	382	60	928
AmBB 10%	1.1	248	3	572
Homogeneous MA/Am (at equilibrium)	0,6	242	1,5	543

3.1.4.4 Impact on reactor availability

The introduction of minor actinides leads to an increase in the number of actions during the refueling operation, such as reshuffling or increasing movement of the MA assemblies. Therefore, the availability is slightly reduced. An assessment of the handling time with the CEA tool, OCTET leads to an increase in this time of one equivalent day at every reloading operation. As the opening-and-closing time is 400 days, the availability is reduced by 0.25% [9].

3.1.5 Economics

The purpose of these studies is not to raise issues about the economics of the nuclear industry in France and in the world. Therefore, providing power production costs in absolute terms is not meaningful, as these depend on the particular industrial and marketing environments in which the plants will be constructed. Besides, several fuel cycle related process will need new technology development and subsequent industrial assessment, that increases cost evaluation uncertainties. Comparing production costs in relative terms is quite sufficient for the exercise conducted in this paper which consists in a simple inter-comparison analysis of the different scenarios.

The comparison of the levelised costs of electricity per MWh for each scenario is conducted when the equilibrium is reached (when a total SFR fleet is deployed). For the computation of cycle costs in the levelised cost of electricity per MWh, we took into account each fuel step (front end and the back end). For each step, we evaluated the unitary cost and, for discounting purposes, the mean time interval in regard to the fuel irradiation time.

Two sets of discount rates have been chosen enabling to assess the sensitivity of the results to this parameter:

- The first case is representative of a "private estate" economic approach. The discounting is performed with a rate of 8 % over the first 30 years and then 3 % after that.
- The second case corresponds to a "public" or "public interest" economic approach. The discounting is performed with a rate of 4 % over 30 years, which then decreases to 2 %.

In the first approach (referred to as approach A), economic calculations were carried out with a unit cost database for reactors and cycle operations. In this database, the operating costs are assessed by analogy with existing facilities while assuming advanced cost-saving developments. The second approach (referred to as approach B) does not account for the advanced developments considered in approach A due to the fact that they might not provide suitable industrial solutions (due to modified regulations, for instance). Furthermore, approach B is based on hypotheses, which are much more cautious regarding certain major parameters (which are not clearly understood) such as, e.g., the plant availability factor (Kd).

Results with Approach A

Table 4 presents the average production costs for the different scenarios. An index of 100 has been allocated to the average production costs for 2120-2150 (representative of the fleet at equilibrium) of scenario F4 (without any transmutation). The index 100 does not represent the same value for the two sets of discount rates.

Table 4: Standardised production costs

Scenarios	Discounting rate of 8%-3%		
	Total	Reactor	Cycle
Without MA	100	94	6
Het. all MA	106	96	10
Het. only Am (20%, F1J)	104	95	9
Het. only Am (10%, F29)	105	95	10
Hom. all MA	108	95	12
Hom. Am	105	95	10
In ADS	126	116	10

Scenarios	Discounting rate of 4%-2%		
	Total	Reactor	Cycle
Without MA	100	91	9
Het. all MA	107	92	14
Het. only Am (20%, F1J)	105	92	13
Het. only Am (10%, F29)	107	92	15
Hom. all MA	109	92	17
Hom. Am	106	92	13
In ADS	124	110	14

With the hypotheses which have been selected, approach A shows that:

- The cost overrun related to the transmutation in SFR would amount to around 4 to 9 % overall. As for the cycle alone, cost overruns could reach 50 to 60 %. However, the cost overrun for the reactor item would not exceed 2 %;
- There would be no large distinction from an economic viewpoint between the homogeneous and heterogeneous transmutation options;
- Only the transmutation in the ADS could generate a significant cost overrun of around 26 %.

It has also been observed that the relative order of merit of the scenarios is only slightly affected by the choice of the discount rate due to the fact that the cash flow chronologies are very similar for the different scenarios.

A sensitive study indicates that the most important items are SFR investment and operating costs, and then to a lesser degree, fuel processing and manufacturing costs. The same conclusions apply to the 4%-2% discounting rate.

Results with Approach B

The economic calculations have been modified and more penalizing values have been used, particularly concerning the items which have been identified as important in the sensitivity analysis. Different values have been chosen compared to approach A for three types of reasons:

- Technical divergences leading to different cost assessments.
- Different philosophies, particularly concerning future requirements regarding safety and radiation protection. It is assumed that the technological developments will not be able to deal with such matters.
- A greater caution concerning certain aspects which are not clearly understood at this stage (Kd, etc.).

Figure 9 compares the results between Approach A and B.

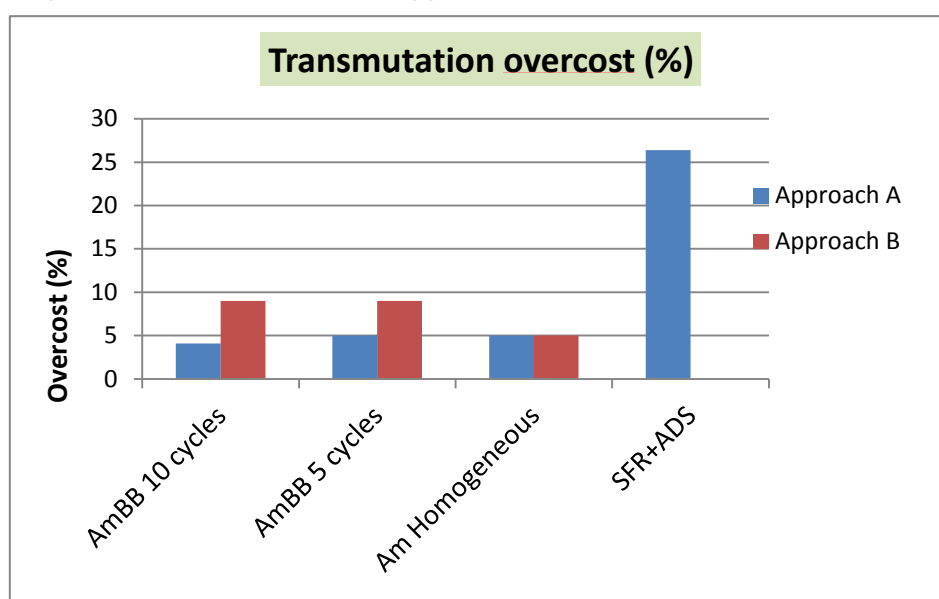


Figure 9: Transmutation overcost between Approach A and B

The cost overrun related to the transmutation in SFR would amount to around 4 to 9 % overall and around 26% in the case of ADS. So, except ADS, the economy is not very discriminating on the choice of the transmutation option.

3.1.6 Technological maturity, flexibility, progressivity

All of the scenarios described in this paper refer, to variable degrees, to processes or technologies which have not been implemented at present, notably for industrial use, and in some cases, which are still concepts.

The maturity of the processes and technologies to be implemented in the different scenarios is a major parameter which is required for understanding the technical and economic risks related to the different options. Nevertheless, risks will be limited if the selected options provide sufficient flexibility and if the new technologies are implemented progressively. For instance, processes which are continuously updated through successive addition of new functions involve less risks than processes requiring dedicated facilities, which are, in most cases, not easily adaptable to new options.

It is obvious that these risks will be all the more high as the technologies employed will not be mature. This is the case for the scenario involving the transmutation of minor actinides in ADS. Although significant progress has been made regarding the feasibility demonstration,

ADS are complex systems whose development requires the design of components with a high technicality and whose feasibility is not yet guaranteed.

The technological maturity of the cycle operations for the transmutation of MA has not been reached and still requires a long R&D process, specifically regarding the transmutation scenarios of all of the minor actinides. As expected, the presence of minor actinides at the manufacturing and processing steps will cause design modifications for dealing with radiation protection and thermal problems. The transmutation of curium raises design constraint issues for all of the facilities.

The progressivity and flexibility of the options are probably key factors for their industrial development. The progressive addition of americium in the blankets involves much less industrial risks than, for instance, the combined extraction of plutonium and minor actinides with a homogeneous recycling in all of the fuel flowing through the electronuclear fleet. A progressive implementation of the options as they become technically mature and economically acceptable, will always be a safer solution from an industrial viewpoint than the sudden introduction of extremely innovative technically challenging options.

3.2 CONCLUSIONS

These studies carried out in collaboration with EDF, AREVA and Andra have let to obtain many important and “objective results” on a set of defined scenarios and with an improved and satisfactory methodology related to the advantages and drawbacks of MA transmutation options and consequences on fuel cycle plants.

Only the transmutation of all minor actinides, through multi-recycling operations in SFR, enables to stabilise their inventory over time. This involves two elements which must be identified:

- The transmutation of minor actinides significantly reduces their inventory in the geological repository due to the fact that they are no longer transferred to the waste packages. The amount of minor actinides present in the waste is essentially determined by the quantity which is already present when the transmutation is implemented.
- The immediate consequence of the reduction of the minor actinide content in the waste is an increase of the MA inventory in the cycle (reactors and plants). As transmutation is a relatively slow process, the minor actinides accumulate in the facilities prior to reaching an equilibrium level. The calculated inventories vary from 60 to 160 tons according to the transmutation concepts. It is recalled that, at the same time, the plutonium inventory amounts to around 1,000 tons.

Reducing the thermal load of high-level waste packages due to the transmutation of americium greatly reduces the underground area covered and the excavated volume of the high-level waste disposal modules without questioning the performances and safety of the repository. The transmutation of Am and MA would result in a reduction by a factor up to 7.3 (Am) to 9.8 (MAs) of the footprint of the HLW disposal zone after an interim storage period of 120 years, a total reduction by a factor 3 of the repository footprint taking into account ILW-LL and common infrastructures.

As can be expected, calculations reveal high contents of minor actinides at the fuel manufacturing and processing steps, which will generate significant design modifications for dealing with obvious thermal and radiation protection problems. The complexity of the operations carried out during the operating phase (loading / unloading, interim waste storage, transport) will also be increased. Scenarios involving the transmutation of all minor actinides are extremely impeded by the presence of curium and the implementation constraints often exceed that of the scenarios, in which only americium is transmuted, by one order of magnitude.

Economic studies have been conducted on the different scenarios in order to determine the impact of the transmutation of the cost of electricity production. They show that the cost

overrun related to the transmutation process could vary between 5 to 9% in SFR and 26 % in the case of ADS according to the assumptions being considered.

The transmutation of curium raises design constraint issues for all facilities whereas its impact on reduction of high-level waste zone is low compared to the impact of Am transmutation.

4. FAISABILITE ANALYSIS OF SCENARIOS WITH THE PROGRESSIVE DEPLOYMENT OF PU MULTIRECYCLING IN SFRs IN THE FRENCH NUCLEAR POWER FLEET

Recycling as carried out in France today makes it possible to recover the fissile material that remains in spent fuel, mainly plutonium and uranium which are recycled as MOX or ERU (Enriched Reprocessed Uranium) fuel respectively. The other elements (fission products and minor actinides) amount to only a few per cent of the mass of spent fuel (representing the ultimate waste resulting from nuclear power generation). They are conditioned in a glass matrix that traps the radionuclides in the long term. Deterioration of plutonium's fissile isotope content during its irradiation in reactor conditions does however make it more difficult - after a first round of recycling - for any subsequent recycling in water reactors. In practical terms, plutonium recycling is limited to one round (once-through recycling) and "second-generation" plutonium contained in spent MOX fuels is currently not recovered. Fast reactors use uranium and plutonium better because not only they more efficiently transform ^{238}U into plutonium, but they also make the fission of all plutonium isotopes possible. The integration of fast reactors into the nuclear power plant fleet therefore seems to be the most logical solution for using the plutonium contained in MOX fuels (currently not used) and more generally for ensuring the recurrent recycling of recoverable materials.

Rather than suddenly switching over to the large-scale deployment of fast reactors as assumed in past scenarios, it is now deemed preferable to ensure the progressive implementation of this technology through successive phases: each phase involves the more significant deployment of fast reactors with its own growth objective.

These phases can therefore be summarised as follows:

- Phase A: Once-through recycling in PWRs
- Phase B: Recycling of spent MOX fuel
- Phase C: Stabilisation of the Pu inventory
- Phase D: Independence with respect to natural uranium.

A phase 0 was also defined. This phase corresponds to a hypothetical French fleet having operated in an open-cycle configuration only.

The phases were assessed according to a multi-criteria approach by taking into account the inventories between the different phases, as well as the capacities of the fuel cycle facilities required, the transport of nuclear fuels, and the resulting radwaste. This assessment will subsequently be supplemented by economic data (see §1).

The initial conditions of the scenarios correspond to the current French fleet with its 58 PWR units amounting to 63.1 GWe and generating 430 TWhe/year (average load factor of 78%).

The working assumptions for the study were based on a constant nuclear power generation over time, as well as a future reactor lifespan (PWRs and FRs) of 60 years for the duration of the scenarios.

Four fuel loading configurations were chosen for PWRs: 100% UOX, 100% ERU, 30% MOX and 100% MOX. The latter is only required for phases C and D.

A self-breeder core concept with a low sodium void effect (CFV) was chosen for fast reactors; these are the characteristics chosen for the ASTRID integrated technology demonstrator design. Two industrial models were taken into consideration: 1000 MW_e and 1450 MW_e. For the latter model, we also considered an alternative version including two rows

radial blankets making it possible to transmute americium or to reach a breeding gain of about 0.2 for certain scenario options [10].

A lifespan of 50 years was assumed for the fuel cycle plants (reprocessing and manufacturing), which places discussions on the future of current plants at around 2040-2050 at a time when the first commercial fast reactors could be deployed.

4.1 DESCRIPTION OF THE PHASES

4.1.1 Phase A

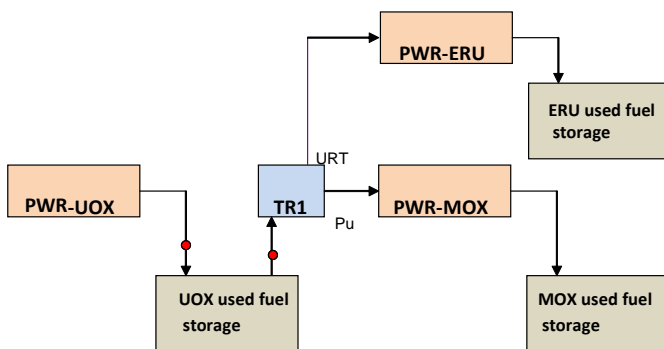
This phase makes it possible to recover any residual uranium and plutonium contained in spent UOX fuels, which actually stabilises the interim storage needs for such fuel.

UOX fuels are reprocessed as they are produced (red dots in the figure 11) and all of the plutonium recovered during reprocessing is recycled as MOX fuel with a rate of 30%¹ in PWRs. This scenario corresponds to a *status quo* with respect to the current situation (PWRs having reached the end of their lifespan are replaced with EPR-type PWRs).

Uranium recovered from reprocessing is enriched and recycled (known as ERU fuel) in specific PWRs. This recycling method is an industrial reality already implemented in the current fleet (using a significant quantity of URT (reprocessed uranium) in 4 specific reactor units), which explains why this assumption was chosen for this phase. The actual recycling varies depending on the natural uranium market, assumed to be favourable in the timeframe of this study.

Fission products and minor actinides are immobilised in glass.

Spent MOX and ERU fuels are placed in interim storage since the PWR fleet cannot recycle the plutonium and second-generation uranium contained in these spent fuels under satisfactory conditions. These spent fuels are recycled in the following phases: they may become radwaste if phases B to D are not implemented.



(TR1: reprocessing and recycling of UOX fuel)

Figure 10. Schematic diagram of phase A

The characteristic fleet for phase A is composed of 100% PWRs, i.e. 38 EPR-type reactors generating 423 TWh_e/year. The net installed capacity is 58.1 GW_e, which can be broken down as follows:

- 24 UOX-PWRs (36.7 GW_e)
- 11 PWRs with 30% MOX (16.8 GW_e)
- 3 PWRs with 100% ERU (4.6 GW_e)

All in all, 8.7% of the nuclear power is generated by MOX fuels, 7.9% by ERU fuels and 83% by UOX fuels.

¹ Fraction of the MOX fuel in the PWR core (identical to the current French fleet)

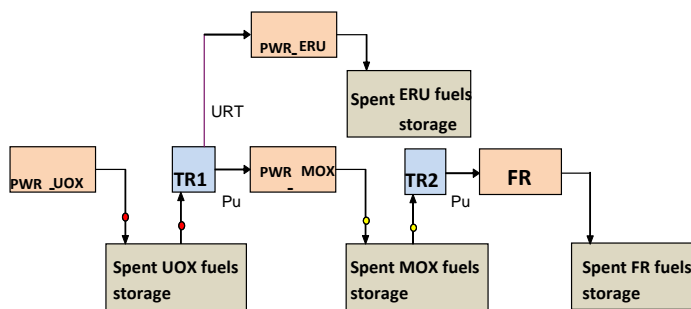
The natural uranium consumption of such a fleet - based on the assumptions retained for enrichment operations² - is about 6,300t/year, i.e. savings of about 17% compared with a fleet operating with an open fuel cycle (phase 0). The plutonium inventory increases at a rate of 7.4t/year. The large majority of this inventory is found in spent fuel placed in interim storage, which is particularly the case for MOX fuels. The total minor actinide inventory increases at a rate of 3.2t/year and is practically evenly distributed between vitrified waste resulting from the reprocessing of UOX fuels and spent fuel placed in interim storage. Compared with an open fuel cycle, the integration of MOX-PWRs that burn plutonium therefore leads to a lower Pu inventory for phase A, but an increase in the minor actinide inventory. The total inventory of transuranium elements (plutonium and minor actinides) is generally reduced by about 15%.

4.1.2 Phase B

The objective of phase B is to use the plutonium contained in spent MOX fuels by deploying a limited number of fast reactors. This makes it possible to stabilise the interim storage of spent MOX fuels.

This phase also includes "industrial apprenticeship" - in a few units - of a fast reactor component integrated into a fleet that is still mainly composed of PWRs.

Fast reactors are only supplied by plutonium resulting from the reprocessing of MOX-PWR fuels and spent fast reactor fuels are not recycled (Fig. 11).



(TR1: reprocessing and recycling of UOX fuels)

(TR2: reprocessing and recycling of MOX fuels)

Figure 11: Schematic diagram of phase B

The fast reactor power required for this phase comes directly from the plutonium flow produced by the reprocessing of MOX-PWR fuels. It will mainly depend on two parameters: the unit power of fast reactors and the reprocessing order of spent MOX fuel.

A power of around 1000 MW_e was chosen with respect to the unit power of fast reactors for this phase. These reactors will be the first of a kind of a Generation IV industrial-scale fast reactor in France. This power level lies between the ASTRID prototype power of 600 MW_e and the expected fully mature industrial-scale reactor power of 1450 MW_e. It has been deemed to be a realistic assumption from an industrial viewpoint.

Concerning the MOX spent fuel reprocessing strategy, a middle-road solution was chosen between the two strategies under investigation during the parametric studies, i.e. a mix of cold and hot spent fuels in practically equal proportions. This "50/50" strategy makes it possible to reduce technical difficulties expected to arise with conservative strategies, i.e. a high fraction of ²³⁸Pu (exceeding 4%) for the "hot-first" strategy which is restrictive in terms of radiation protection and thermal conditions in the fuel cycle; and a high plutonium content (~30%) in fast reactor fuels for the "cold-first" strategy, which results from the significant isotopic deterioration during interim storage (loss of ²⁴¹Pu by radioactive decay). Lastly, it is

² Assumption: content of 0.2% of ²³⁵U in depleted uranium

possible to adapt the proportion of the mix in order to stabilise the interim storage conditions of spent MOX fuels with a whole number of reactors deployed.

The nuclear power fleet during phase B therefore includes 35 PWRs and three 1000 MW_e fast reactors, which produce a total of 410 TWh_e/year. The net installed capacity is 56.5 GW_e, which can be broken down as follows:

- 22 UOX-PWRs (33.7 GW_e)
- 10 PWRs with 30% MOX (15.3 GW_e)
- 3 PWRs with 100% ERU (4.6 GW_e)
- 3 fast reactors (3.0 GW_e)

In this configuration, 8% of the power is generated by MOX fuels, 8% by ERU fuels, 5% by fast reactor fuels and 79% by UOX fuels.

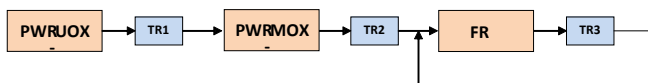
This phase is marked by the deployment of a small number of fast reactors, representing about 5% of the total power generated by the nuclear power fleet. The integration of several fast reactors will only marginally affect the flow of materials and inventories estimated for phase A but provides a drastic reduction in required spent fuel interim storage.

4.1.3 Phase C

Phase C corresponds to the implementation of plutonium multi-recycling in a fleet comprising PWRs and FRs with the objective of stabilising the plutonium inventory.

This phase therefore considers the reprocessing of all types of spent fuel, particularly FR fuels and the multi-recycling of plutonium thus recovered. Two main options seem feasible: the multi-recycling of plutonium can be carried out either in fast reactors only (option 1), or both in PWRs and fast reactors (option 2).

Option 1: recycling Pu in fast reactors only



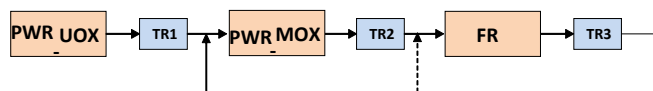
TR1: reprocessing and recycling of UOX fuels

TR2: reprocessing and recycling of PWR-MOX fuels

TR3: reprocessing and recycling of FR-MOX fuels

The principle in this case is to recycle plutonium in fast reactors only, with these reactors being by nature the most suitable for performing multi-recycling. Yet as PWRs are net producers of plutonium, a 'burner' mode must be adopted for fast reactors (which must be "net consumers" of plutonium) to stabilise the overall plutonium inventory.

Option 2: recycling Pu in MOX-PWRs and fast reactors



TR1: reprocessing and recycling of UOX fuels

TR2: reprocessing and recycling of PWR-MOX fuels

TR3: reprocessing and recycling of FR-MOX fuels

The principle here involves deploying a greater proportion of MOX-PWRs which are net consumers of plutonium but cannot multi-recycle it - as previously mentioned - owing to the deterioration of its isotopic composition with each recycling. Fast reactors can thus be used to "rectify" this isotopic composition and allow for its reuse in PWRs. In this way, fast reactors no longer have to be operated in burner mode because all the plutonium is consumed in the PWRs in the form of MOX fuel.

It was this second option that was first investigated where plutonium is produced by UOX-PWRs before it is consumed in MOX-PWRs, with the fast reactors ensuring the correction of its isotopic composition to enable its recycling in MOX-PWRs.

During this phase which includes a higher number of FRs in the fleet, it was also considered that the uranium recovered during the reprocessing of UOX fuel (URT) could be used to manufacture FR fuels. There is therefore no more enrichment of URT, which means that the fleet no longer includes PWRs loaded with ERU fuels (enriched URT fuels) during this phase.

The nuclear power fleet during phase C (option 2) therefore includes 22 PWRs and 16 fast reactors, which produce a total of 420 TWh_e/year. The proportion of MOX fuel is 40% in PWRs, which leads to the recommendation of 100% MOX PWRs (foreseeable with EPRs) to supplement the standard load at 30% chosen for the previous phases.

The net installed capacity is 57 GW_e, which can be broken down as follows:

- Nineteen 1530 MWe PWRs loaded with 30% MOX (29.1 GW_e)
- Three 1600 MWe PWRs with 100% MOX (4.8 GW_e)
- Sixteen 1450 MWe fast reactors (23.2 GW_e)

In this configuration, 65% of the nuclear power generation is ensured by MOX fuels (PWRs or FRs) saving as much natural uranium, and 35% by UOX fuels.

To make possible the recycling of plutonium from FRs in PWRs in the form of MOX, there must be a first stage where this plutonium is recycled in FRs; this initial stage will be all the longer since the plutonium from FRs is not very well corrected. To limit the length of this stage, the performance of FRs could be adapted to make them slight net producers of plutonium (breeding gain of about 0.2), with the isotopic quality of plutonium therefore being easier to rectify so it is acceptable for PWRs while the system remains self-sufficient (by construction) overall.

The consumption of natural uranium at this stage is estimated at 2700t/year. This is greatly reduced compared with the previous phases (by about -53% compared with phase A and -65% compared with stage 0 – no recycling).

The implementation of phase C will necessarily require fuel cycle plants that employ the appropriate technologies for manufacturing and treating the required quantities of FR fuels (about 250t/year). By way of comparison, the French experience in manufacturing FR fuels amounts to a little more than a total of 100 tonnes between 1963 and 1999. The reprocessing of FR fuels amounts to 25 tonnes, which was mainly carried out between 1973 and 1991.

The plutonium inventory is stabilised when phase C reaches equilibrium (useful Pu inventory is about 700 t). The total minor actinide inventory increases at a rate of about 3.6t/year. This inventory is mainly conditioned in glass canisters produced during fuel reprocessing processes. This growth is similar to that observed for phase B. The majority of the minor actinides are produced during the irradiation of MOX fuels in PWRs.

4.1.4 Phase D

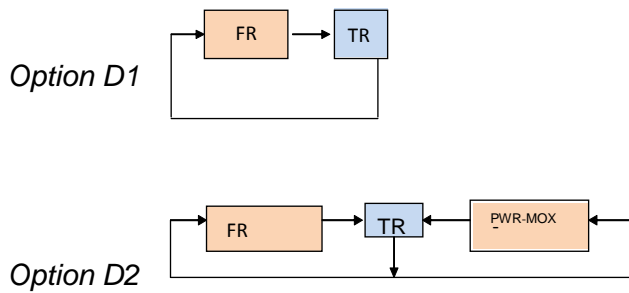
Phase D aims at eliminating any need for natural uranium to supply the nuclear power fleet. Like in the previous phase, it also aims at stabilising the Pu inventories.

The objective of gaining independence with respect to natural uranium requires that the only fuel be a plutonium-based fuel (PWR-MOX or FR-MOX), with the addition of depleted uranium which can also come from URT stocks or result from the reprocessing of ERU fuels.

Two approaches are possible:

- a homogeneous 100% FR fleet (version D1) which could comprise forty-one 1450 MW_e self-breeder fast reactors to produce 430 TWh_e/year

- a mixed fleet composed of 28 fast breeder reactors and 10 100% MOX PWRs (version D2) to produce 424 TWh_e/ year.



TR: reprocessing and recycling of FR-MOX

TR: reprocessing and recycling of PWR-MOX and FR-MOX

Figure 12: Schematic diagram of phase D

4.1.5 Phase main characteristics of the phases

Table 5 summarises the characteristics of the different phases under investigation in relation to the fleet composition, flows and characteristic inventories.

The transition from phase A through to phase D improves - at each phase - the quantities which are characteristic of the sustainable management of materials, particularly regarding the consumption of natural uranium or the interim storage of materials (e.g. plutonium in spent fuels) or the waste produced.

Each phase makes it possible to improve the industrial maturity of the fast reactors whose integration into phase B remains very minor (5% of the fleet).

An issue remains to be resolved for phase C with respect to the design and its complex operation. It may be decided to move directly from phase B to phase D on the condition that there is a suitable driving force in place.

TABLE 5

Main characteristics for each phase

	A	B	C	D1	D2	0
Fraction of FRs in the fleet	0%	5%	40%	100%	75%	0
Unat consumption (t/year)	6300	5800	2700	0	0	7600
UOX(+ERU) manufacturing (t/year)	800 (+75)	730 (+75)	335			960
MOX manufacturing (t/year)	83	75	220		255	
FR (core) manufacturing (t/year)		26	215	480	375	
UOX (+ERU) reprocessing (t/year)	820	750	570 (+20)			
MOX reprocessing (t/year)		65	170		255	

FR (core+radial blankets) reprocessing (t/year)			275	480	470	
Pu inventory (t/year)	↗ 7,4	↗ 7,1	Stabilized	Stabilized	Stabilized	↗ 10,5
AM inventory (t/year)	↗ 3,2	↗ 3,1	↗ 3,6	↗ 2,2	↗ 3,3	↗ 2,5

4.2 FUEL CYCLE TRANSITION SCENARIOS

After having characterised each phase, the transition between these phases was analysed according to various assumptions (especially time). It was possible to assess the technical, economic and industrial feasibility and realism of the different options leading to the global scenarios comprising each phase.

As a result of this process, four global scenarios were defined: 1) scenario ABCD1; 2) scenario ABD1, which is an alternative to ABCD1 where the deployment of a 100% SFR fleet is accelerated (due to the increasingly higher cost or shortage of natural uranium); 3) scenario ABCD2 where the final fleet is symbiotic and composed of SFRs and 100% MOX EPRs; 4) scenario ABD2, which is an alternative to ABCD2 where the deployment of a symbiotic fleet is accelerated.

The year 2050 was chosen as the commissioning date for the first SFR in phase B. Phase C starts with the implementation of spent SFR fuel treatment and the deployment of additional SFRs. According to our assessment, there appears to be no pressing incentive to launch phase C any earlier, which led us to choose a B-to-C transition from 2090 corresponding to the renewal of the fuel cycle plants. Nevertheless, the deployment of three additional SFRs before 2090 is considered during this transition since it does not seem to be industrially realistic to build a new series of SFRs from 2090 at a sustained pace following a 30-year period without any construction.

The transition between C and D is estimated to take about 30 years for D1 and 50 years for D2. Equilibrium is then reached around 2190-2210 depending on the option chosen for phase D. The early transition from B to D makes it possible to reach this equilibrium much faster and to overcome our dependency on natural uranium resources sooner. In this case, our independence with respect to natural uranium is gained about 60 years earlier. This early transition from phase B to D requires an increase in the treatment capacities of plants so as to recover the plutonium contained in the spent fuels, which is needed for the accelerated deployment of a 100% SFR fleet while maintaining its total power generation at a constant level.

4.2.1 Total plutonium inventory

The variation in the overall plutonium inventory is shown in Figure 13. Contrary to scenario A, the transition scenarios make it possible to reach the objective of stabilising the plutonium inventory. The accelerated transition scenarios (ABD1 and ABD2) stabilise the Pu inventory at a lower level due to the fact that the spent fuel stocks are used to optimise the transition, with the additional possibility of being able to increase the fuel treatment capacity during the 2090-2120 period.

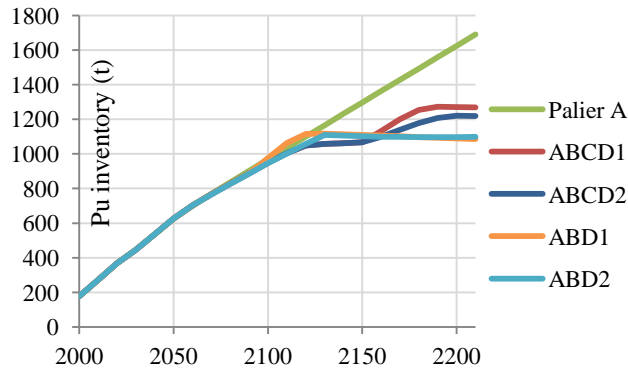


Figure 13: Variation in the total Pu inventory for the transition scenarios

4.2.2 Storage of spent fuels

The accelerated transitions make it possible to significantly reduce the storage of spent fuels since it is necessary to recover as much plutonium as possible from these stockpiles. Figure 14 shows the variation in these stockpiles for the different scenarios.

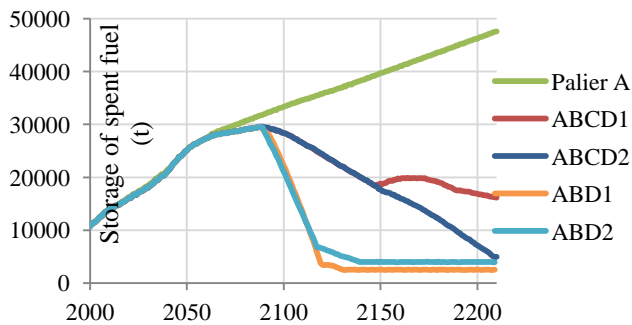


Figure 14: Variation on the storage of spent fuels for the transition scenarios

4.2.3 Natural uranium requirements

The consumption of natural uranium obviously stabilises faster in the scenarios with an accelerated transition. Scenario ABD1 makes uranium savings of about 18% compared with scenario ABCD1, while scenario ABD2 makes savings of about 16% compared with scenario ABCD2. Regardless of the transition scenario, the natural uranium requirements are significantly reduced compared with a French fleet that continues to operate with the once-through recycling of Pu in EPRs (scenario A).

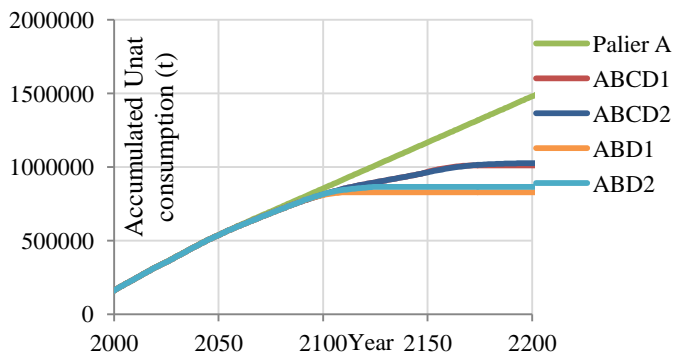


Figure 15: Accumulated consumption of natural uranium for the transition scenarios

4.2.4 Transport of SFR fuel sub-assemblies

The feasibility of transporting SFR fuel sub-assemblies does not appear to be called into question by our preliminary studies. Based on current regulations and transport cask designs similar to those used today, however, the number of SFR sub-assemblies will have to be reduced per transport cask. This will result in an increased number of transport operations. According to our studies, the criticality risk limits the number of SFR fuel sub-assemblies that can be transported in a cask. In terms of heat releases, SFR fuel sub-assemblies in transport casks must be kept in a helium atmosphere to promote thermal conductivity and heat removal. It will also be necessary to improve the resins used in these transport casks with respect to their heat removal performance. This issue requires further research within the scope of optimising transport casks for SFR fuels.

The table below lists the number of transport operations for three transition scenarios with respect to scenario A, i.e. continuing the current strategy of once-through recycling of Pu in PWRs. The total has been calculated for the 2050-2210 period.

As expected, it can be seen that the differences with respect to scenario A increase with the proportion of SFRs integrated into the fleet. The faster deployment of SFRs (scenario ABD1) also logically results in a higher total number of transport operations, excepting from UOX fuels and new radial blankets.

TABLE 6

Total number of transport operations between 2050-2210 compared to an extended scenario A

Proportion of SFRs	75%	100% from 2181	100% from 2122
Scenario	ABCD2	ABCD1	ABD1
UOX + ERU + new radial blankets	0.6	0.5	0.4
New MOX	8.1	8.9	12.0
Spent sub-assemblies	2.8	3.0	3.9
New and spent sub-assemblies	1.9	2.0	2.5

4.2.5 Waste and geological disposal

The total disposal surface area required for the three transition scenarios has also been calculated on the basis of the 2005 Andra disposal concept. These results do not currently take into account the latest concept changes recently recommended by Andra which incorporate thermo-hydro-mechanical criteria. The disposal surface areas have thus been calculated on the basis of thermal criteria, which consider a limit temperature of 90°C on the walls of the hottest waste package in the disposal facility. Scenario ABD1 requires the smallest surface area but it must be stressed that the differences in surface areas are of the same order of magnitude as the uncertainties on the study assumptions. This information will be specified in greater detail in future studies, taking into account the latest results from Andra.

TABLE 7

Estimate of HLW and LL-ILW footprint over the 2050-2210 period

	Scenario ABCD1	Scenario ABCD2	Scenario ABD1
LL-ILW	1.1	1.1	1.3
HLW	15.9	17.4	13.8
Total	17.0	18.5	15.1

As a comparison, scenario A requires an overall surface area of about 6.5 km² (assumption: 722 HLW waste packages per year for 160 years). It is important to point out that, in this

case, the disposal requirements for spent fuel will continue to rise to reach more than 47,000 tons in 2210. The deep geological disposal of these spent fuels is expected to represent an additional surface area of about 21.5 km².

4.2.6 Economic competitiveness

The economic analyses all indicate that the economic competitiveness of SFRs is highly dependent on the additional investment cost of the reactor compared to PWR and on the price variations of natural uranium. The competitiveness of SFRs may be reached sometime between 2080 and 2110 if an additional investment cost of 30% is made in comparison with PWRs. If this extra investment cost was increased to 50%, the date of competitiveness of SFRs would be shifted by fifteen years or so. This explains why we chose an ABD-type scenario over an ABCD scenario. Once the ABCD scenario has reached equilibrium (around 2150), the ABCD1 scenario appears to be more expensive than ABCD2 owing to the greater number of SFRs in the fleet; the contribution of the reactor cost to the fleet cost is greater than the fuel cycle cost.

4.3 CONCLUSIONS

On the basis of these promising results, these studies will continue to explore certain avenues of research and new options. Among the needs that have been identified, the following areas are being investigated:

- Robustness of the conclusions reached in 2015 with respect to the assumptions applied
- Industrial overview of the transition from phase A to B, particularly with respect to the industrial fuel cycle facilities and integration of the Astrid technology demonstrator into this transition.
- The results of the disposal surface area required, which are being updated to take into account the latest developments recommended by Andra, e.g. incorporation of thermo-hydro-mechanical criteria.
Andra has been asked to reassess the estimations of the disposal surface areas and excavated volumes required for each of the different phases.
- Tech watch of other reactor concepts capable of recycling plutonium from spent UOX and MOX fuels and of new MOX products, including their integration into the above scenarios.
- The transport capacities not only for spent and new SFR and PWR fuels, but also for PWR fuels recycling Pu. These capacities are currently being analysed and refined.
- The costs of fuel cycle facilities and reactors, and new methodologies for improved economic assessments.
- The environmental impact of the different options and the life cycle analysis (LCA).

These new results will allow us to meet the different requirements governing radioactive material and waste management issued by the French government (PNGMDR plan) under the 2006 Act on radioactive waste management.

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References

1. C. Coquelet-Pascal, M. Tiphine, G. Krivtchik, D. Freynet, C. Cany, R. Eschbach, C. Chabert, "Cosi6: a tool for nuclear transition scenarios studies and application to SFR deployment scenarios with minor actinides transmutation", Nuclear Technology, Vol 192, 91-110 (2015).

2. J.-M. Vidal et al., "Cesar5.3: an industrial tool for nuclear fuel and waste characterization with associated qualification", Proceedings of the WM2012 conference, Paper 12067, (2012).
3. P. Sciora, et al., "A break even oxide fuel core for an innovative SFR: CEA neutronic studies", Proc. of GLOBAL 2009, Paris, France, Sept. 6-11, 2009.
4. W. Maschek, et al., "Design, safety and fuel developments for the EFIT accelerator driven system with CERCER and CERMET cores", Proc. of IEMPT 10, Mito, Japan, Oct. 6-10, 2009.
5. C. Coquelet-Pascal, et al., "Comparison of different scenarios for the deployment of fast reactors in France – Results obtained with COSI", Proc. of GLOBAL 2011, Makuhari, Japan, Dec. 11-16, 2011
6. C. Coquelet-Pascal, et al., "Some scenarios for minor actinides transmutation in the frame of the French act for waste management", in *IEMPT*, Prague, 2012
7. C. Chabert, D. Warin, J. Milot, A. Saturnin and A. Leudet, "Impact of minor actinide transmutation options on interim storage and geological disposal," in *IEMPT*, Prague, 2012
8. A. Saturnin et al., "Transmutation Scenarios Impacts on Advanced Nuclear Cycles (fabrication/reprocessing/transportation)", International Conference on Fast Reactors and related Fuel Cycles, FR13 (2013)
9. G.M. Gautier, F. Morin, F. Dechelette, E. Sanseigne, C. Chabert, « Technical and economic impacts of the transmutation of minor actinides in a SFR », Proceedings of ICAPP2012, Paper 12142 (2012)
10. L. Buiron, L. Dujcikova, "Low void effect (CFV) core concept flexibility: from self-breeder to burner core", Proceedings of the ICAPP 2015 Conference, Paper 15091 (2015).
11. L. Buiron et al, «Heterogeneous minor actinides transmutation on a UO₂ blanket and on (U,Pu)O₂ fuel in sodium-cooled fast reactor. Assessment of core performances» *GLOBAL*, Paris, 2009.
12. G. Martin, G. Krivtchik, M. Tiphine, R. Eschbach, "French transition scenarios toward Sodium Fast Reactors in this century", Proceedings of the Global 2017 Conference, Paper 098 (2017).
13. G. Martin, M. Tiphine, C. Coquelet-Pascal, "French transition scenarios toward a symbiotic nuclear fleet", Proceedings of the ICAPP 2016 Conference, Paper 15732 (2016).
14. M. Tiphine, C. Coquelet-Pascal, G. Krivtchik, R. Eschbach, C. Chabert, B. Carlier, M. Caron-Charles, G. Senentz, L. Van Den Durpel, C. Garzenne, F. Laugier, "Simulations of progressive potential scenarios of Pu multirecycling in SFR and associated phase-out in the French nuclear power fleet", Proceedings of the Global 2015 Conference, Paper 5326 (2015).
15. C. Chabert, A. Saturnin, G. Krivtchik, G. Martin, M. Tiphine, G. Mathonnière, S. Gabriel, A. Allou, E. Fillastre, FX. Giffard, P. Sarratb, JL. Giroto, H. Hancock, B. Carlier, G. Senentz, F. Laugier, "Feasibility of future prospects and transition scenarios for the French fuel cycle", Proceedings of the Global 2017 Conference, Paper 087 (2017).
16. C. Chabert et al., "Considerations on industrial feasibility of scenarios with the progressive deployment of Pu multi recycling in SFRs in the French nuclear power fleet", Proceedings of the Global 2015 Conference, Paper 5351 (2015).