cea

Nuclear Energy Division

Advances in research on partitioning-transmutation and plutonium multi-recycling in fast neutron reactors

Summury of results

June 2015

SUMMARY OF RESULTS

1. Sustainable management of materials: multiple recycling

Spent nuclear fuel unloaded from water reactors usually contains the following:

- about 95% residual uranium;

- approximately 4% fission products which can be described as fragments resulting from the fission of uranium-235 (^{235}U) and plutonium-239 (^{239}Pu);

- 1% of transuranium elements, mainly plutonium (most of the isotopes present are fissile), with the remainder being minor actinides which are thus named because they barely represent 0.1% of the total content of the spent fuel.

The spent fuel processing and recycling strategy was first implemented in France more than 30 years ago and represents the first key phase to sustainably managing radioactive materials and waste. This involves processing spent fuel to salvage recoverable material (uranium and plutonium), whereas the other compounds (fission products and minor actinides) are considered as "ultimate waste".

Therefore:

- all of the plutonium recovered by spent fuel reprocessing is recycled in mixed uranium-plutonium oxide (MOX) fuels, which is used in 24 of the 58 reactors operating in France today;

- the recovered uranium can be re-enriched and reloaded, which is currently possible in 4 of the French fleet's reactors;

- fission products and minor actinides (comprising only 4 to 5% of the content of spent fuel) are currently embedded in glass and poured into steel containers which are stored in cells pending their disposal in a deep geological repository.

This strategy has a number of advantages in terms of resource saving, plutonium inventory control, reduced ultimate waste (which does not contain large quantities of plutonium) and safe packaging of this waste. It therefore helps meet the objectives defined by the French Act of 28 June 2006 on the sustainable management of radioactive materials and waste. There are, however, a number of obstacles (explained further on) which prevent the implementation of multiple plutonium and uranium recycling in the current fleet of water reactors. Once-through recycling implemented in the current fleet allows France to recycle about 10 tonnes of plutonium in MOX fuels which, once unloaded, is placed in interim storage pools.

The possibility of recycling uranium and plutonium multiple times up to their almost complete transformation, is a key issue for the 4th generation of nuclear systems, as detailed below.
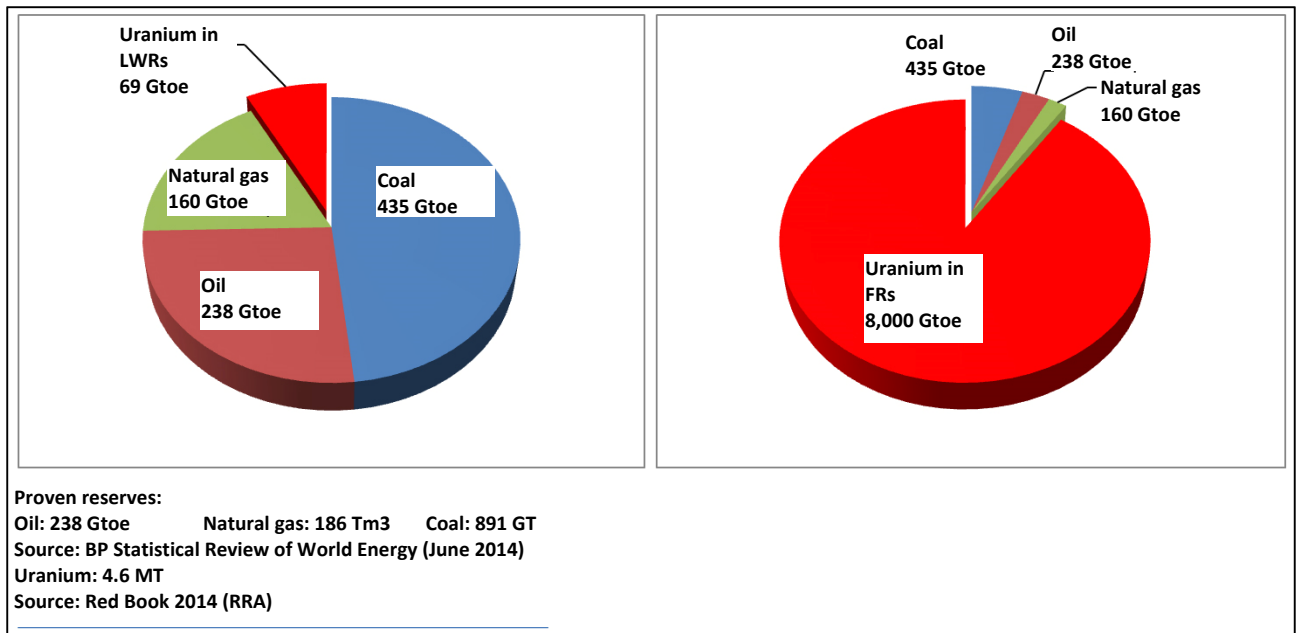
Recovery of conventional global uranium reserves in thermal neutron reactors (left) and in fast reactors (right)

The figure above illustrates the energy recovery potential (expressed in Gtoe) for conventional resources identified in fossil fuels. The recovery of uranium resources is shown in red; on the left for the current nuclear reactors and on the right for fast reactors with the multiple recycling of uranium and plutonium.

2. Sustainable management of materials: fast reactors

Fast reactors have a number of advantages when it comes to radioactive material management while being complementary with the current French thermal reactor fleet which includes pressurised water reactors (PWR):

- Fast reactors can use the plutonium produced by water reactors (or by themselves) indefinitely, thus getting the most out of its energy potential while avoiding the long-term disposal of spent fuel containing plutonium.

- By exploiting all of the uranium extracted from mines (all of its isotopes including ^{238}U, which is the main isotope), fast reactors increase the energy that can be extracted from a given mass of natural uranium by a factor of almost 100. This would have a profound impact on the estimated global energy reserves as it would be a significant advantage for energy generation in the long term.

- Fast reactors are able to operate without natural uranium, once the necessary quantity of plutonium has been provided at the start. They only require a single top-up in ^{238}U, a non-fissile isotope of uranium that is currently not exploitable in water reactors, despite being found in large concentrations in depleted uranium resulting from enrichment operations. Whereas the French nuclear fleet uses about 8,000 tonnes of natural uranium each year and scraps about 7,000 tonnes of depleted uranium, a fast reactor fleet of equivalent power would only require about 50 tonnes of depleted uranium. The depleted uranium stocks in France on the Pierrelatte site alone, i.e. about 250,000 tonnes, would provide a practically inexhaustible source of energy independence for a fast reactor fleet.

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- The fast neutron spectrum also opens the door to minor actinide transmutation and thus the possibility of reducing the inventory of these radionuclides in waste (if this option is chosen), while reducing the size of the deep geological repository needed for the disposal of ultimate waste.

Fast reactors therefore appear as a key link in the closed-cycle strategy by making it possible to fully exploit the materials contained in spent fuel.

3. Fourth generation of fast reactors

The Generation IV International Forum (GIF) established the foundations for discussions on advanced nuclear systems in the early 2000s. The main criteria have been defined - safety, economics, proliferation resistance and sustainability - which all highlight the relevance of recycling in fast reactors. The six most promising systems have already been identified though the technological maturity of the different concepts chosen by the GIF is somewhat variable.

In view of the objectives defined under the 2006 Act, the CEA's efforts are focussing primarily on sodium-cooled fast reactor technologies (SFRs) due to their advanced level of maturity and high potential for progress.

The 'nuclear reactors' standing group of experts which met on 10 April 2014 confirmed that: "*to date, among the different nuclear systems investigated under GIF, only the sodium-cooled fast reactor (SFR) technology is sufficiently mature to foresee the construction of a 4th generation industrial prototype in the first half of the 21st century* ". SFRs offer several valuable technical characteristics in terms of safety and radiation protection and are the only reactors that can benefit from a substantial corpus of operating experience (OPEX). The twenty or so prototypes and demonstrators which have been built throughout the world represent more than 400 combined reactor years of operation, with about 100 reactor years for the four SFRs of significant power which operated efficiently on an industrial level. In France, the Phénix reactor was shut down in 2009 after more than 35 years of operating, thus offering a wealth of knowledge and experience to the scientific community.

On the basis of lessons learnt from previous reactors having operated in France and in the world, the main French players in the nuclear sector - the CEA, Areva and EDF - defined a R&D programme in 2007 which sets out to consolidate the advantages and strengthen the weaknesses of this technology through innovative scientific developments. This made it possible to launch the design studies for a 4th-generation SFR integrated technology demonstrator called Astrid.

4. Astrid programme

The Astrid integrated technology demonstrator is first and foremost designed to demonstrate, on a sufficiently large scale, technology breakthroughs by qualifying innovative options during its operation, particularly in the fields of safety and operability.

Astrid is an integrated technology demonstrator designed to generate about 600 MWe, which is needed to demonstrate the safety case and to show that a 4th generation SFR can be operated on a pre-industrial level.

The project is aiming for commissioning sometime in the 2020s, depending on the decisions reached by the public authorities. The operation of Astrid for about ten or so years should lead to the deployment of commercial power reactors, which can be reasonably expected in the 2040s at the earliest.
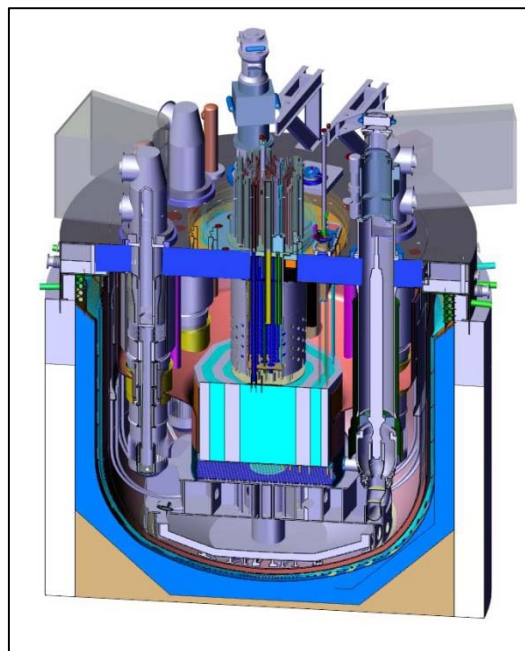
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Cross-sectional view of the reactor block for the Astrid integrated technology demonstrator

Thanks to the innovations earmarked for Astrid, it will be possible to reach a level of safety equivalent to that set for EPRs, while taking into account feedback from the Fukushima accident and specificities related to the use of sodium. The innovative options chosen for Astrid include:

- New reactor core design that is inherently resistant to accident transients

- Design of a tertiary water-free power conversion system to eliminate the risks of chemical reactions between sodium and water, or innovative measures to prevent and contain such reactions

- Core catcher integrated into the reactor vessel to catch molten core debris

- Inerting and early leak detection systems which greatly reduce the risk of sodium fires

- Integration of in-service inspection requirements into the reactor design

- Multiple, redundant decay heat removal systems. The reactor can use the atmosphere as a means of cooling in the case where the power supplies and water heat sink are lost.

Contrary to previous reactors (Phénix and Superphénix), Astrid will not be a breeder reactor, but self-sufficient without fertile radial blankets so as to stabilise the plutonium stocks without increasing them.

Astrid will also be able to qualify experimental fuels right up to the scale of full sub-assemblies: transmutation of americium, fuels designed to burn more plutonium, fertile fuels, etc. The experimental capabilities of Astrid will be vital in qualifying such fuels. The CEA has checked that the Astrid design choices - especially the CFV core concept with a low void effect - will make it possible to qualify up to a block of seven sub-assemblies, thereby enabling extrapolation to commercial-sized power reactors.

At the end of these programmes conducted in Astrid, we should have a sufficient level of industrial qualification to build - if this choice is made at the time - industrial-scale reactors capable of transmutation and of burning a greater quantity of plutonium. The design of these reactors will be optimised with respect to the tasks they need to perform.

The CEA has been the project leader of the Astrid design studies since 2010, with funding provided by a State subsidy called the "Future Investments Programme" (PIA) and various external sources of financing (European projects under the Framework Programme). Industry players involved in the design studies are providing contribution in-kind since parts of the studies under their authority are self-funded. There are currently more than six hundred people working in R&D and engineering for the Astrid project; approximately half are from the CEA while the rest are provided by the thirteen industrial partners.

The design work for the Astrid project started in 2010. This work has been broken down into three phases:

- The first phase called the pre-conceptual design (AVP1) set out to analyse the various solutions (particularly the most innovative) so as to choose the main design options. The CEA presented these solutions to the public authorities in late 2012, the deadline set by the Act of 28 June 2006.

- The second phase called the conceptual design (AVP2) started in January 2013. It aims at consolidating the design by validating the options chosen so as to achieve a finalised coherent conceptual design by late 2015. This phase will include the completion of the safety options file (DOS) which will be assessed by the Nuclear Safety Authority (ASN) to check the validity of the options in this field.

- The third phase called the basic design will take place between 2016 and 2019. This phase will focus on studies bringing together all the technical, organisational and financial data in order to decide whether Astrid should be built. The Preliminary Safety Analysis Report will be issued at the end of this phase.

A specific organisation has been set up to manage the R&D and design studies for the Astrid project. The project has been divided into study packages which have been allocated to different industrial partners, usually within the scope of bilateral agreements, i.e. part of the study is funded by the partner. The Astrid project currently boasts a total of thirteen industrial collaborations.

Most of the main nuclear-power-generating countries have investigated the SFR technology, but without necessarily focusing on the 4th-generation safety criteria. India is expected to be commissioning a 500 MWe power reactor in the ensuing months while Russia started up an 800 MWe reactor in June 2014. China remains in the background but has voiced its intentions in the field. The CEA's strategy for international collaborations consists in finding complementary skills - as well as experimental facilities where possible - in the fast reactor field, so as to enhance its research efforts to support the Astrid programme.

This report provides an overview - at the time of writing - of the design studies presently underway for the Astrid project. The end of the conceptual design in 2015 will provide another opportunity to take stock of the project work accomplished, both in engineering and in R&D (more than 2,300 documents written to date).

5. Developing technologies for the multiple recycling of plutonium

Thanks to its industrial plants at La Hague and Melox, France currently has the means to process and recycle the spent uranium oxide (UOX) fuels produced by its nuclear power reactors. Almost 1,000 tonnes are unloaded and processed each year. The 10 tonnes of plutonium recovered are fully recycled in MOX fuels (two thirds of this quantity will remain in spent MOX fuels) and almost 940 tonnes of reprocessed uranium can be re-enriched to power specific reactors in the French fleet. The remainder (about 50 tonnes of fission products and minor actinides) is embedded in glass which is remarkably resistant; this represents the main final waste containing more than 99% of the radioactivity that needs to be managed.

The technologies used achieve excellent results, whether in terms of the very high recovery and purification rates for the elements to be recycled, or the very low volume of soft housekeeping waste generated.

The possibility of multi-recycling uranium and plutonium in a fleet that would progressively incorporate fast reactors leads to:

- treatment of PWR MOX fuels - currently in interim storage - to retrieve the recoverable material (especially plutonium) from them;

- manufacturing of FR MOX fuels using the recovered plutonium;

- reprocessing and recycling of spent FR MOX fuels (repeated to achieve multiple recycling).

This does not necessarily entail using breakthrough technologies that differ completely to those used today in order to meet these objectives. The current technologies boast a number of inherent advantages (summarised below) which need to be conserved and which have proven their worth and their capacity for adaptation. Industrial-scale campaigns conducted in La Hague plant confirmed the viability of processing spent PWR MOX fuels (almost 70 tonnes were processed). The fabrication of fuels for the Phénix and Superphénix reactors resulted in the production of almost 100 tonnes of fuel pellets for fast reactors, i.e. more than 400,000 pins were manufactured. The feasibility of the processing principle for FR MOX fuels has also been demonstrated both in pilot facilities and at La Hague plant, with almost 25 tonnes of spent fuel from Phénix having been processed and recycled.

The current objective of research in this field is to prepare for the industrial-scale deployment of such operations by best adapting the technologies to high plutonium concentrations and material flows with characteristics (especially isotopes) that differ to those used in the past experiences mentioned above. The implications of certain risks, such as criticality, are to be taken into consideration in the management strategy; special innovative devices with suitable geometries may prove necessary. Some aspects related to how these technologies are operated must also be modified (such as remotely operated devices which will need to be adapted to work with plutonium isotopes that are more complex from a radiological viewpoint).

The specificities of FR MOX fuels also infer a number of changes to certain stages of the treatment process: this is the case for the dissolution operations whose standard operating conditions need to be modified to be completely efficient. One of main outcomes of research conducted in this field was the development of an additional "digestion" stage for plutonium-rich residue which may remain following the fuel dissolution phase. The process was experimentally demonstrated on real samples of spent MOX fuels. A significant increase in the plutonium recovery rate (the quantity of plutonium found in undissolved residue was reduced by a factor of 100) was measured, without this causing any problems with the downstream operations of the process (separation, purification and vitrification). The tests performed on smaller hydraulic mock-ups at the CEA also validated the transposition of this process to an industrial scale, by taking advantage of the physico-chemical and hydro-dynamic simulation means available in this field.

These studies have therefore helped define the equipment that may be used for industrial operations, such as the treatment workshop for special fuels (TCP) assessed by Areva that may supplement the current facilities and which could also possibly be used to recycle Astrid fuels.

Another important objective is to maintain or even consolidate the reliability and economic efficiency of the recycling operations, especially considering the fact that they will become more widespread leading up to the possible multiple recycling of uranium and plutonium to secure energy supplies. Current technologies have demonstrated remarkable performance levels, but that does not mean there is no room for considerably more progress: making the operations more compact (by simplifying unit operations or by consolidating their integration) is a key topic of research, while making them safer and generating less waste and effluents. Promising results have been obtained as regards optimising the current process, with the prospect of treating fuels with higher plutonium contents (MOX fuels). Otherwise, one of the main research programmes in this field concerns the development of a uranium and plutonium recovery and purification process based on a much simpler principle without involving any oxidation-reduction operations. Significant progress has been made in this field through the design of new extraction molecules, which made it possible to conduct a first test on real samples of spent fuels in late 2014. These results were very encouraging: the target separation performance levels could therefore be reached in a single operation and under implementation conditions considered more efficient on several levels (limited reagents and sub-products in particular).

Lastly, the majority of studies in this field focus on the facilities that will be required for the Astrid material cycle. At this stage of the process, efforts are concentrating on the workshop concept which should manufacture the first Astrid cores; this is evidently the most pressing need, with the recycling of Astrid spent fuels being necessary more than ten years after the reactor has reached criticality. Several options are still on the drawing board for this core fabrication workshop (AFC), with the technical & economic issues of these options having been assessed in partnership with Areva.

6. Research on the separation & transmutation of americium

Minor actinides for transmutation

Waste that is currently vitrified contains fission products and minor actinides which are residue from reprocessing operations carried out in order to recycle uranium and plutonium. The half-lives of most of the fission products are relatively short, which is not the case for minor actinides and their daughter products. Although they only represent about a thousandth of the mass of spent fuel, these minor actinides (mainly neptunium, americium and curium) are the main contributors (either themselves or their daughter products) to the radiotoxicity of ultimate waste for more than 100 years (once most of the fission products have transformed into stable isotopes). This is what underpins the ambition to eliminate these elements by separation and then transmutation in fast reactors, which offer several advantages in this respect.

This has been explained in the report published by the CEA in 2012 on the subject, with americium having been identified as the first target for a possible separation and transmutation strategy with the goal of reducing the long-term harmfulness of ultimate waste. As highlighted by results reported in late 2012 by the CEA, americium is the main contributor to radiotoxicity and decay heat for over 100 years in vitrified ultimate waste. In its 8th annual report, the French national evaluation commission recommended that "*active, structured research" on separation and transmutation be continued by focusing on americium first of all*". It is important to remember that this only makes sense within the prospect of plutonium multiple recycling; plutonium would be the main contributor well before americium, if not recovered from the waste. Minor actinides other than americium have no potential significant long-term impact, though some of them, like curium, pose an immediate radiological and thermal threat in the case of any specific processing and recycling operations. This is why we only consider americium recovery and transmutation as a first step.

The transmutation of americium would have a considerable impact on the high-level waste disposal surface area required, reducing it by an estimated factor of more than 7[1] (in addition to the three-fold reduction thanks to plutonium recycling). A complementary analysis is currently underway to assess to what extent the current disposal concepts may evolve (particularly to take into account additional thermomechanical effects) and how such changes could impact this estimation.

Research on the separation of americium

Research is thus now focussing on the possibility of recovering americium during the treatment of spent fuels. From the perspective of a mixed reactor fleet with a PWR component - transitional at the least - in which plutonium would be recycled in MOX form, americium would be recovered and transmuted in fast reactors only (since transmutation is not possible in PWRs) according to a separate cycle from that of plutonium. At this stage, we are leaning towards the "sequential separation" method by looking to develop selective extraction and complexing agents in order to - downstream of the uranium and plutonium recovery operations - recover americium while leaving curium in the fission products.

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[1] Study conducted by Andra in 2012

A process called EXAm (standing for EXtraction of Americium) was designed and tested in laboratory conditions. This process is rendered complex due to the double separation that we are trying to perform in a single phase (separating americium from the fission products and separating americium from curium). It has undergone a number of changes, whether to improve the separation performance or to integrate industrial requirements (such as being able to use sufficiently concentrated solutions to limit the volume of the solutions and the size of the equipment). We have measured americium recovery rates higher than 99% and tested various solutions to improve purification techniques that remove curium.

The next step involves testing the optimised process on real spent fuel samples: this integral test involves recovering americium from a sample of about 4 kg of spent fuel to manufacture americium pellets which could be transmuted. Such experiments are currently being performed in the ATR test reactor (Idaho, USA). The first dissolution steps and the recovery of uranium and plutonium have already been performed. The americium recovery operations should start in the second half of 2015, followed by the manufacturing of the oxide pellets so they can be loaded into the reactor in 2019.

Research on the transmutation of americium

Transmutation studies are particularly focusing on minor-actinide-bearing blankets (MABB). This concept involves the transmutation of americium in specific objects positioned on the periphery of a fast reactor (uranium oxide pins loaded with about 10% americium). This option has the advantage of dissociating the transmutation operations from the reactor operation. It provides certain benefits, particularly in terms of safety because americium is harmful when introduced in significant quantities into the core, as stated in the 8th report by the French national evaluation commission. It could also provide a progressive means of implementation in certain reactors of the French fleet, assuming that the transmutation option is chosen.

Today, research in this field is nonetheless hampered by the lack of available fast reactors to conduct fully representative experiments. In this context, we are focusing on studying the main phenomena that govern the behaviour of MABBs under irradiation by carrying out so-called 'analytical' irradiations in experimental reactors. One of the main issues concerns the behaviour of helium in the material, as helium is generated in considerable quantities during the transmutation of americium-241 and could significantly affect pellet behaviour. This issue is being investigated in the Marios and Diamino irradiation experiments performed in the HFR (Petten, Holland) and the Osiris reactor (Saclay, France) respectively. These experiments cover all the operating parameters (temperature in particular) for a blanket placed on the periphery of a fast reactor core. The first examinations are underway for the Marios experiment which has been removed from the core, whereas the Diamino experiment is still ongoing. The preliminary results for both should be available in 2016, before the above-mentioned experiment start in the ATR in the US.

Research is also exploring processes used to manufacture Am-bearing objects (fuels, targets, blankets, etc.): the stages of conversion (making americium oxide powders from solutions produced during the separation phase) and fabrication are the subject of developments and experiments in the Atalante facility at the CEA Marcoule centre. This involves transposing the processes and technologies currently used to make mixed uranium-plutonium oxide pellets so they can be used on americium, i.e. precipitation and calcination of oxalate before making MOX pellets by means of powder metallurgy. This also requires exploring new methods that are capable of minimising the production of fine particles which can be particularly harmful in the case of americium powders: a process that binds americium to resin before calcination (known as the weak acid resin (WAR) process) was tested successfully for the fabrication of several grams of americium oxide.

7. Fast reactor deployment scenarios: stakes and objectives

The report submitted to the French government in December 2012, which assessed various options for sustainably managing radioactive materials and waste, highlighted the relevance of incorporating fast reactors into the French fleet. This report recommended "*focussing on a progressive approach with scenarios that set out to first deploy a limited number of fast reactors in synergy with the water reactors comprising the French fleet.*" It recalled the reasons why it was also important to "*not delay the development of this first phase programmed for around 2040.*" Lastly, it stated that "*studies of industrial scenarios* [would] *be led with Areva and EDF to refine the approach*". The CEA, Areva and EDF have therefore set about studying different options for introducing fast reactors into the French fleet based on a realistic approach which takes into account industrial assumptions and constraints. This has shed light on the subsequent possible benefits and conditions required for success. A specific organisation in the form of a steering committee has been set up to successfully complete this work by relying on two groups of experts to orient and organise the studies. One group is responsible for recommending and elaborating scenarios while the other validates the assumptions and results by comparing them to a given industrial reality.

Rather than imagining the large-scale replacement of PWRs by fast reactors in a short time span (the assumption of some past studies), it was decided to study a progressive deployment scenario taking into account existing materials and facilities. This solution appears to be better suited to the dynamics of technical progress in the field, while providing greater flexibility to adapt to societal changes. The path for this scenario is marked by successive milestones (stages), with each one corresponding to an increased deployment of fast reactors with their own objective of increasing ambition.
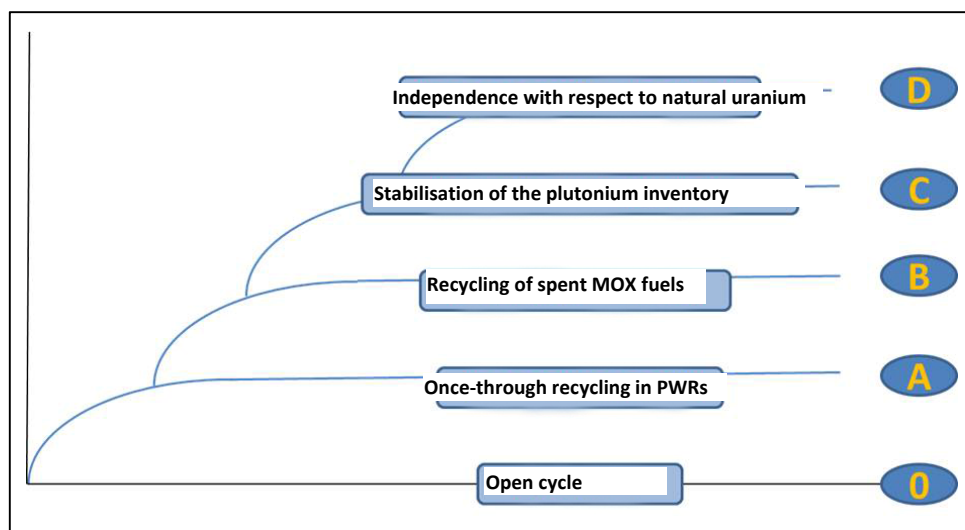
The objectives of the various milestones were identified as such:
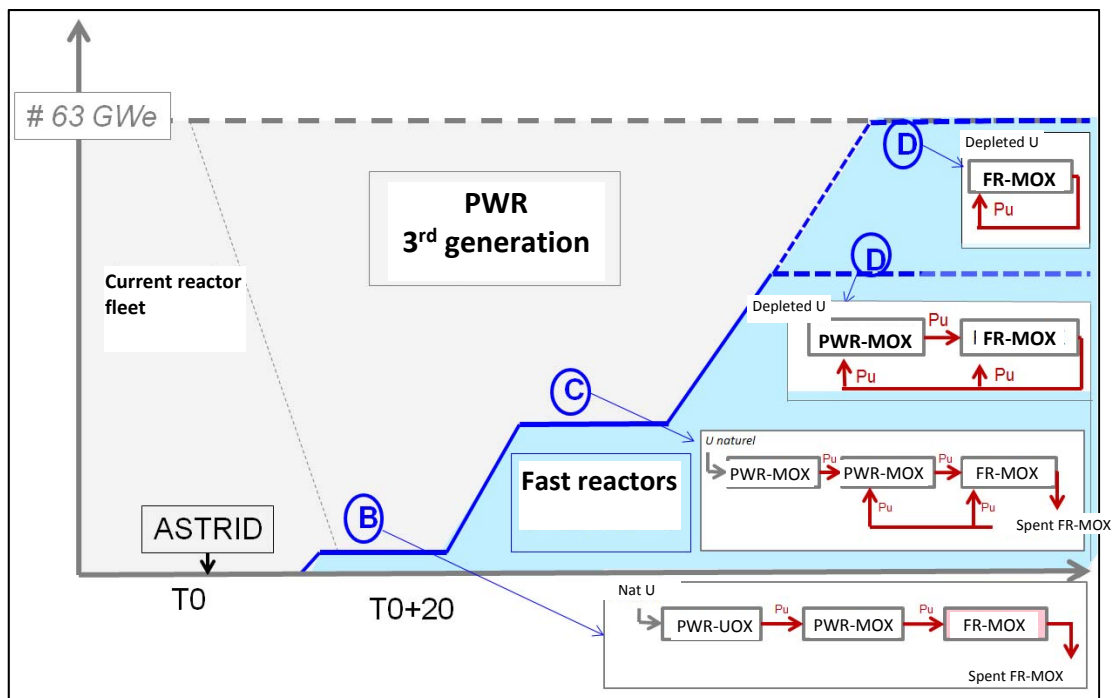
- Milestone A: once-through recycling of spent fuels in PWRs (current cycle without fast reactors)

- Milestone B: recycling of spent MOX fuels (or twice-through recycling of spent fuels) in PWRs and then in fast reactors

- Milestone C: stabilisation of the plutonium inventory through multiple recycling

- Milestone D: Independence with respect to natural uranium.

These various stages have been characterised and benchmarked against numerous criteria, particularly material & waste inventories and natural uranium consumption. At this stage, we want the deployment of fast reactors to progress as much as possible while maintaining a fraction of water reactors at each stage so it remains compatible with the milestone.

Main results of the studies on industrial scenarios

The following main conclusions were reached:

- It appears that the transition from milestone A to milestone B leads to an improvement (at each stage) in the quantities that are characteristic of the sustainable material management, particularly natural uranium consumption and the interim storage of materials with no immediate use (see figure below which is based on the assumption of a constant nuclear power generation of 420 TWh/ year). This seems to confirm the relevance of such an approach, which, as mentioned above, makes it possible to further drive the industrial maturity of fast reactors at each stage, though their incorporation in stage B remains minor (5% of the fleet, with the current nuclear reactors providing the necessary quantities of plutonium to start up these fast reactors around 2040).

	Milestone 0 open cycle (no recycling)		Milestone A once-through recycling (PWR fleet)	Milestone B twice-through recycling (PWRs then FRs)	Milestone C multiple recycling (PWRs and FRs)	Milestone D multiple recycling without natural U (FR fleet)
FR proportion of the fleet (GWe)	0		0%	5%	40%	100%
Natural U consumption	**7,600 t/year**		**6,300 t/year**	**5,800 t/year**	**2,700 t/year**	**0**
Net Pu production	**+10.5 t/year**		**+7.4 t/year**	**+7.1 t/year**	**0 (stabilisation)**	**0 (stabilisation)**
Spent fuel stocks	+960 t/year (UOX)		+160 t/year (MOX+RepU)	+100 t/year (FR MOX+RepU)	Stabilised	Stabilised

- Analysis of the waste produced during each stage confirms the advantages of a recycling policy: the transition to greater recycling stages leads to reducing the overall surface area required for deep geological disposal on the whole. The volume of high-level vitrified waste (HLW) remains constant for all stages and well below the volume represented by spent fuel within an open-cycle strategy. The integration of fast reactors leads to a reduction in operational waste - mainly low-level (LLW) and short-lived intermediate-level waste (SL-ILW) - together with an increased volume of long-lived intermediate level structural waste (LL-ILW) which requires much less disposal area than HLW.

  In the case where the phases with multi-recycling in FRs are not deployed, it is nevertheless important to remember that fuels with no use (e.g. spent PWR-MOX) would automatically become radwaste to be added to the quantities of waste already estimated as of today (analysed in terms of the required disposal surface area in the last line of the table below).

  Assessment of the impact on the disposal facilities remains to be refined, particularly by taking into account the design changes foreseen by Andra.

  The analyses were carried out and integrated the results provided by Andra. They show that:

  - Once-through recycling as practised today leads to a reduction in the disposal surface area required for HLW by a factor of about 3 compared with an open-cycle option (no recycling)

  - Most of this benefit would be lost if we decided to simply dispose of spent MOX fuels (they only represent 10% of the spent fuel flow, but their disposal surface area would be greater than that required for vitrified waste resulting from the treatment of all of the other fuels from the fleet)

  - A multiple recycling option with fast reactors makes it possible to maintain the benefits of the current strategy in the long term

  - The transmutation of americium would make it possible to once again reduce this required disposal area by a factor of about 20 compared with the open-cycle strategy.

- The transmutation of americium can only be carried out efficiently if the fleet includes a significant number of FRs. Fleets with a significant fraction of PWR fuels render this operation more complex. For the transmutation of americium, the smaller number of fast reactors in the fleet must therefore be loaded with americium in greater proportions (americium contents, number of blankets required), which hardly seems foreseeable at this stage of the project. Though it can be initiated in the previous stages, it seems that the transmutation of americium must be associated with the prospect of a fleet that is mainly composed of fast reactors.

- The implementation of plutonium recycling strategies can reduce the net production of plutonium and ultimately the overall plutonium inventory remaining when the fleet is shut down. The longer the fleet operates, the greater the plutonium reduction will be. Fast reactors can also be used in burner mode (net consumer of plutonium) to reduce inventories, but this point remains to be investigated in greater depth.